10.3 MAIN STEAM SUPPLY SYSTEM

The function of the main steam supply system is to supply steam from the steam generators to the high-pressure turbine over a range of flows and pressures covering the entire operating range from system warmup to valves wide open (VWO) turbine conditions.

The system also provides steam to the moisture separator/ reheaters, the auxiliary feedwater pump turbine drive, the steam seal system for the main turbine, the feedwater pump turbine drives, the condenser spargers, and the steam jet air ejectors. The system also dissipates heat generated by the nuclear steam supply system (NSSS) by means of steam dump valves to the condenser or to the atmosphere through power-operated atmospheric relief valves or spring-loaded main steam safety valves when either the turbine-generator or condenser is unavailable.

10.3.1 DESIGN BASES

10.3.1.1 Safety Design Bases

Pertinent safety design bases are as follows:

A. The system provides the means of dissipating residual and sensible heat generated from the NSSS during hot shutdown and cooldown even when the main condenser is not available. Power-operated atmospheric relief valves are provided to allow controlled cooldown of the steam generator and the reactor coolant system when the condenser is not available.

B. The system is provided with two main steam isolation valves (MSIVs) and two associated MSIV bypass valves in parallel with the MSIVs in each main steam line. These valves isolate the secondary side of the steam generators to deal with leakage and malfunction and to prevent the uncontrolled blowdown of two steam generators and isolate nonsafety-related portions of the system.

C. The codes and standards utilized in the design of the main steam supply system are identified in table 10.3.2-1. The following components are designed in accordance with American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code Section III, Class 2, and Seismic Category 1 requirements and are considered the safety-related portion of the system.

   1. All piping and valves from the steam generators to and including the pipe restraints provided downstream of each outboard main steam isolation valve (MSIV) to maintain piping loads upstream of the restraint in accordance with Branch Technical Position MEB 3-1.

   2. Branch lines from the above portions of the main steam lines up to and including the first valve (including a safety or relief valve) that is either normally closed or capable of automatic/remote manual closure during all modes of normal reactor operation.

   3. MSIV system media actuators.

   All other components located downstream of the outboard MSIV restraint assembly are designed in accordance with the Power Piping Code, American National Standards Institute B31.1.

D. The safety-related portion of the system is designed to withstand the effects of a safe shutdown earthquake. The safety-related portion of the system is capable of withstanding the effects of natural phenomena and of performing its intended function following postulated hazards of fire, internal missile, and pipe break.

E. The safety-related portion of the system is designed so that a single failure in the main steam supply system will have no contributory effects on:

1. Initiation of a loss-of-coolant accident.
2. Integrity of other steam lines.
3. The capability of the engineered safety features system to effect a safe reactor shutdown.
4. Transmission of excessive loading to the containment pressure boundary.

Component redundancy is provided so that safety functions can be performed assuming a single active failure coincident with loss of offsite power.

F. The portion of the main steam supply system that is constructed in accordance with ASME III, Class 2 requirements is provided with access to welds and removable insulation, as required for inservice inspection in accordance with ASME Section XI. (See paragraph 10.3.4.4.)

G. The main steam supply system provides steam to the auxiliary feedwater turbine-driven pump during emergency conditions and for shutdown operations.

H. The main steam supply system is designed to function in the normal and accident environments identified in subsection 3.11.1.

I. The main steam supply system is designed in accordance with Branch Technical Positions ASB 3-1 and MEB 3-1 with regard to high-energy pipe break location and evaluation.

10.3.1.2 Power Generation Design Basis

The following is a list of the principal power generation design bases:

A. The main steam supply system is designed to deliver steam from the steam generators to the turbine- generator for the range of flowrates, temperatures, and pressures existing from warmup to rated conditions.

B. Each main steam line is sized to provide balanced steam pressures at the turbine stop valves, using the turbine VWO (maximum calculated - not guaranteed) steam flowrate and conditions in accordance with Westinghouse Steam System Design Manual.

C. The main steam supply system provides the capacity to dump 40 percent of VWO (maximum calculated) steamflow to the condenser during plant step-load reductions.

10.3.2 DESCRIPTION

10.3.2.1 General Description

The main steam supply system, shown in drawings 1X4DB159-1, 2X4DB159-1, 1X4DB159-2, 1X4DB159-3, 1X4DB160-1, 1X4DB160-2, 1X4DB160-3, 1X4DB166, 1X4DB167-1, 1X4DB167-2, 1X4DB191, 1X4DB193, and 1X4DB196, includes the following major components:

A. Main steam piping from the steam generator outlet steam nozzles to the main turbine stop valves.

B. Two main steam isolation valves (MSIVs) and two main steam isolation valve bypass valves per main steam line.

C. Main steam safety valves.

D. Power-operated atmospheric relief valves.

Table 10.3.2-1 lists the design data covering the major components of the main steam supply system. Table 10.3.2-2 lists the design data covering the main steam safety valve design.

10.3.2.2 Component Description

10.3.2.2.1 Main Steam Piping

The main steam lines deliver a total steamflow of approximately 16.25 x 10^6 lb/h from the secondary side of the four steam generators. Each of the main steam lines from the steam generators is anchored at the containment wall and has sufficient flexibility to accommodate thermal expansion. Design of the main steam piping from the steam generators includes design considerations that incorporate the allowable steam generator nozzle loading moments and stresses with all steam generators operating.

The design of the piping and supports takes into consideration all static and dynamic loadings, stresses, and moments arising from normal operation, pressure transients, or pipe rupture. The design of Seismic Category 1 piping and supports takes into consideration the loads discussed in subsection 3.9.3.

The main steam lines between the containment penetration and the first restraint downstream of the outboard MSIV are designed to meet the no-break zone criteria of Nuclear Regulatory Commission Branch Technical Position MEB 3-1 (as described in section 3.6.) so that the piping failures need not be postulated.

A description of the main steam piping from the steam generators to the turbine stop valves is presented in table 10.3.2-3.

The sizing of the 38-in. and the 36-in. common runs hydraulically balances the steam line pressure drops from the respective pairs of steam generators to the inlet of each turbine stop valve. The 36-in. and 38-in. main steam lines are cross-connected with two 20-in. lines into a common 30-in. header, just before branching into the four 28-in. lines to each turbine stop valve, to equalize flow and pressure to the inlet of the turbine stop valves. This permits online testing of each turbine stop valve without exceeding the allowable limit on steam generator pressure differential during this testing.

Each steam generator outlet nozzle contains an internal flow restrictor arrangement of about 1.4 ft^2 to limit flow in the event of a main steam line break. A further discussion is provided in subsection 5.4.4.

Two American Society of Mechanical Engineers (ASME) standard steam sampling nozzles are installed in the nonsafety-related portion of each main steam line downstream of the outboard MSIV to sample for conductivity, pH, and steam generator outlet moisture carryover.

The power operated atmospheric relief, safety, main steam line isolation valves, and MSIV bypass valves are located outside the containment and are installed as close as possible to the containment wall. (Containment penetrations are discussed in subsection 6.2.4.)

Turbine bypass valves are provided between the MSIVs and turbine-generator stop valves, as discussed under the turbine bypass system. (Refer to subsection 10.4.4.)

Main steam line branch connections are made downstream of the isolation valves in the American National Standards Institute (ANSI) B31.1 run of piping with the exception of the takeoffs for the power-operated atmospheric relief valve, connections for main steam safety valves, lines to the auxiliary feedwater pump turbine, low point drains, and high point vents.

Branch piping downstream of the main steam line isolation valves provides steam to the single stage reheaters, steam seal system, main feedwater pump turbines, turbine bypass system, auxiliary steam system, steam jet air ejectors, and condenser spargers.

The main steam piping in the main steam pipe tunnel downstream of the anchor forging at the end of the MSIV/main feedwater isolation valve area is designed in accordance with ANSI B31.1. Anchor forgings are provided at both terminal points for main steam and feedwater piping located in the tunnel and this portion of the piping is designed to withstand the effects of a safe shutdown earthquake (SSE). In addition, pipe whip restraints are provided as necessary to mitigate the effects of postulated pipe breaks discussed in appendix 3F.

10.3.2.2.2 Main Steam Safety Valves

Main steam safety valves are provided with sufficient rated capacity to prevent the steam pressure from exceeding 110 percent of the main steam system design pressure:

A. Following a turbine trip without a reactor trip and with main feedwater flow maintained.

B. Following a turbine trip with a delayed reactor trip and with the loss of main feedwater flow.

The second situation is assumed to occur following a turbine trip resulting from high condenser pressure, which would also cause the loss of the turbine-driven main feedwater pumps. A total main steam safety valve rated capacity of 105 percent of the engineered safeguards design steamflow, at an accumulation pressure not exceeding 110 percent of the main steam system design pressure, meets this requirement. This represents a minimum relief capacity of 4,213,125 lb/h for each main steam line. At the same time, the individual safety valves are limited to the maximum allowable steam relief valve capacity of 970,000 lb/h for a main steam system design pressure of 1200 psia. This limit is imposed on safety valves connecting to the main steam system wherever a single valve failure in the open position would result in the uncontrolled release of steam from one or more steam generators. The purpose of this requirement is to limit the potential uncontrolled blowdown flowrate and the ensuing reactor transient should a single safety valve inadvertently fail or stick in the open position.

Five safety valves are provided per main steam line for the VEGP. The lowest set pressure is 1185 psig and the highest set pressure is 1235 psig. The relieving capacity per main steam line is 4,651,805 lb/h; the total relieving capacity per nuclear unit for four main steam lines is 18,607,220 lb/h. Table 10.3.2-2 lists the performance data for the VEGP main steam safety valves.

The main steam supply system safety valves are located in the safety-related portion of the main steam piping upstream of the MSIVs and outside the containment. Adequate provision is made in the steam piping for the installation and support of the valves. Particular consideration is given to the static and dynamic loads when operating or being subjected to seismic shock.

Each safety valve is connected to its vent stack by an open umbrella-type transition piece.

The vent stacks are designed to:

A. Direct the relieved steam (including steam from MSIV actuator vent, if applicable) from adjoining structures.

B. Ensure that no backflow of relieved steam can escape through the umbrella-type transition section.

C. Ensure that the total steamflow plus a small quantity of ambient air leaves the vent stack outlet.

D. Minimize the backpressure on the valve outlet in order not to restrict the valve rated capacity.

The vent stacks are not required for safety, but are structurally designed to withstand SSE and operating basis earthquake (OBE) loads in order not to jeopardize the performance of safety-related components such as the safety valves, power-operated atmospheric relief valves, or MSIVs.

10.3.2.2.3 Power-Operated Atmospheric Relief Valves

A power-operated atmospheric relief valve is installed on the outlet piping from each steam generator. The four valves are installed to provide for controlled removal of reactor decay heat during normal reactor cooldown when the MSIVs are closed or the turbine bypass system is not available. The valves will pass sufficient flow at all pressures to achieve a 50°F/h plant cooldown rate. The valves are sized to provide a minimum flow of 64,000 lb/h at an inlet pressure of 100 psia. The maximum actual capacity of the relief valve at design pressure is limited to reduce the magnitude of a reactor transient if one valve would inadvertently open and remain open.

Each power-operated relief valve is located outside the containment and upstream of the MSIV, in the safety-related portion of the main steam line associated with each steam generator, to permit valve operation following all accident conditions, including those which could result in closure of the MSIVs.

The operation of the power-operated relief valves is automatically controlled by steam line pressure during plant operations. The power-operated relief valves will automatically modulate open and exhaust to atmosphere whenever the steam line pressure exceeds a predetermined setpoint. As steam line pressure decreases, the relief valves will modulate closed, eventually reseating at a pressure at least 10 psi below the opening pressure. Typically the setpoint is selected between zero-load steam pressure and the set pressure of the lowest set safety valves.

The steam generator power-operated atmospheric relief valves provide a means for plant cooldown by discharging steam to the atmosphere when either the condenser, the condenser circulating water pumps, or the steam dump is not available. Under such circumstances, the relief valves (in conjunction with the auxiliary feedwater system) allow the plant to be cooled down at a cooldown rate of 50°F/h from the pressure setpoint of the lowest set safety valves down to the point where the residual heat removal (RHR) system can assume the burden of heat removal. Residual heat removal operations are initiated when the reactor coolant system hot leg temperature has reached 350°F and primary coolant pressure is less than or equal to 400 psig. The 350°F RHR cut-in point corresponds to a steam generator steam pressure of 125 psia if reactor coolant pumps are operating or to 100 psia if only natural circulation conditions exist in the primary system.

For their use during plant cooldown, the power-operated atmospheric relief valves are automatically controlled by steam line pressure, with remote manual adjustment of the pressure setpoint from the control room or the shutdown panels. In order to effect a plant cooldown, the operator manually adjusts the pressure setpoint downward in a step-wise fashion. As the pressure setpoint of the relief valves is adjusted downward, the relief valves will initially open wide to reduce the steam generator saturation pressure. As the pressure and temperature begin to decrease, the relief valve opening will decrease to an area sufficient to pass the decay heat load and maintain constant steam pressure until the pressure setpoint is again manually reduced. The frequency of these setpoint adjustments and the magnitude of the step reductions (together with the auxiliary feedwater flowrate) determine the average cooldown rate. The maximum cooldown rate achievable is ultimately limited; however, by the flow-passing capability of the relief valves, the number of steam generators (and hence the number of relief valves) in service, and the available auxiliary feedwater pumping capacity and by the desire to either maintain or recover steam generator water levels during the cooldown.

The power-operated atmospheric relief valves also serve to prevent operation of the safety valves during relatively mild transients and, following safety valve actuation, act to assist the safety valves to positively reseat by automatically reducing and regulating steam pressure to a value below the safety valve reseating pressure. The operation of each power-operated atmospheric relief valve is controlled from a pressure tap on the steam generator steam line with which it is associated. This piping connection is separate from the other steam piping pressure taps which are used for reactor protection, to satisfy the requirement for separation between control and protection systems.

The power-operated atmospheric relief valve consists of a Control Components, Inc., pressurized seat offset globe drag valve (8-in. inlet-offset, 10-in. outlet, 900-lb valve rating) with electrohydraulic actuator and a restrictor. The nuclear Class 2 globe drag valve is located inside the MSIV areas; the restrictor (silencer) is placed on a roof outside the control and auxiliary buildings to vent to atmosphere. The noise attenuation is divided into two stages: one by the disk stack inside the main globe valve and the second by the disk stack in the restrictor. This design approach reduces the size of the main valve and the size of the restrictor. The combined design has the capability of controlling the distant field noise as well as the local field noise. It is sized for a sound pressure level of 90 dBa at 3 ft near field noise and less than 55 dBa at 4000 ft for far field noise.

The atmospheric relief valves are electrohydraulically operated and are controlled by Class 1E sources. The capability for remote manual valve operations is provided in the main control room and at the shutdown panels. Local manual operators are provided to permit operation of the valves in the event of a complete loss of automatic or remote manual control.

The valve operator is a self-contained linear modulating electro-hydraulic valve operator for use with an 8-in. x 10-in. globe drag valve. On loss of power and/or signal, the actuator will extend the operator and close the valve. The operator is mounted on the valve by attaching the base plate to the gland of the valve. The operator rod is attached to the line valve stem.

The primary function of the operator is modulation. The actuator recognizes 4- to 20-mA command signals; four mA represents full extension (valve closed) and with increasing signal the actuator retracts (valve opens). The incoming signal is compared to the feedback signal coming from the position transducer by an onboard servo amplifier. If there is a change from the previous level greater than the deadband, the actuator is set in motion until the corresponding position is reached. Loss of command signal will extend the operator and close the valve.

The system stores energy in a pneumatic accumulator and gas bottle pressurized by an electrically driven pump which is controlled by two fluid pressure switches; one to turn pump motor on (decreasing) and the other to turn pump motor off (increasing).

Opening or closing during normal operation is accomplished by either energizing both solenoids or de-energizing both solenoids, to either retract (open valve) or extend (close valve) the operator cylinder rod. Speed or retracting (opening valve) or extending (closing valve) is controlled by a flow control valve which meters the flow of hydraulic fluid from the cylinder to return. The operator can be positioned by means of a permanently installed local hand pump in the event of pump failure.

The electrical system consists of servo valve, three pressure switches (one for gas, two for oil), and a reservoir low fluid level indicator. The system also features a servo amplifier which compares incoming command signals with actuator position feedback transducer and then energizes the corresponding solenoid valve to comply with the command.

The hydraulic system stores sufficient energy to perform its intended functions. The energy is stored in an onboard piston-type accumulator. Charging is initiated by a "fluid" pressure switch which is set to indicate the minimum system pressure necessary for proper operation. When the minimum is reached the system's hydraulic power supply is turned on to restore the system to its peak pressure. Upon reaching maximum, system pressure as indicated by another pressure switch, the power supply is turned off. As the operator modulates, the system pressure drops. When the minimum pressure switch set point is reached, the above process repeats and the actuator is recharged.

For modulation, the dual coil servo valves in the system are operated. This is automatically done by the servo amplifier when there is a change in the incoming signal. To extend the actuator (and close the valve), solenoids A and B are deenergized. Upon reaching the set point when extending, solenoid B is energized to stop the actuator. If a command signal is received to retract the actuator, the servo amplifier energizes solenoids A and B. At the set point, solenoid A is deenergized and solenoid B remains energized (as extending).

The hydraulic circuit is protected from extreme pressure transients by two thermal relief valves. These valves become active when their relief setting is exceeded. Reseat is automatic.

Stroking speed is controlled by a cartridge-type flow control valve which is preadjusted to provide the specified stroking speed.

10.3.2.2.4 Main Steam Isolation Valve System

The function of the main steam isolation system is to limit blowdown to one steam generator in the event of a steam line break in order to:

A. Limit the related effect upon the reactor core within specified fuel design limits.

B. Limit containment pressure to a value less than 90 percent of design pressure.

The main steam isolation system consists of two quick-acting gate valves in series in each main steam line and two associated quick-acting globe type MSIV bypass valves with associated actuators and instrumentation. These valves are located outside the containment building, as close to the containment as practical, and downstream of the steam generator safety valves and the atmospheric relief valve. The isolation system provides positive shutoff with minimum leakage during postulated line severance conditions either upstream or downstream of the valves.

On Unit 2, there are two different designs of main steam isolation valves on each main steam line. The outboard MSIV is hydraulically actuated and the inboard MSIV is operated by main steam system media. On Unit 1, both MSIV actuators are the hydraulic type.

Characteristics Common to both System Media Actuated and Hydraulic Actuated MSIVs

Upon receipt of the closing signal, the main steam isolation valves complete the closing cycle despite loss of normally required utility services for actuator and instrumentation. In loss of actuating power, the valves fail to the closed position. Position indication and remote manual operation of the isolation valves are provided in the control room. Separation of redundant control, power, and indication channels is provided for the isolation valves. Precautions are taken to prevent accidental closure on temporary loss of energy (e.g., electrical power, instrument air, etc.). Additionally, provisions are made for inservice inspection of the isolation valves.

Closure of the MSIVs and MSIV bypass valves is initiated by the following:

A. Low steam line pressure in one of four loops.

B. High containment pressure.

C. High steam pressure rate in one of four loops.

D. Manual actuation; one control (momentary actuation) per loop for single-loop isolation or manual actuation of either of two controls for system level isolation.

Hydraulic Actuated MSIV Characteristics

The hydraulic actuated main steam isolation valves close fully and remain fully closed within 5 s of the receipt of a manual or automatic signal. The hydraulic actuator has sufficient self-contained capacity for two full closures without restoration of utility services. Each hydraulically actuated MSIV is a bidirectional, wedge-type gate valve composed of a valve body which is welded into the system pipeline. The valve body is a straight-through T-pattern with the stem and upper works vertically upright above the flowline. The valve's flow passage is sealed by closing the gate into the seat rings. The gate is made up of two gate halves, which are tapered to match with the angle of the seat rings. The gate halves are guided throughout the stroke by tongues on the gate sides which fit into grooves in the body. The gate is positioned (open or closed) by the stem which is captured between the gate halves in a T-head slot in the gate halves. The body-to-bonnet joint is sealed against leakage by a pressure seal gasket. As the differential pressure across the disk increases, the seating load also increases, thus providing a tight seal throughout the entire range of operating differential pressures. Since the disks are completely independent of each other and the design is essentially symmetrical, positive sealing can be maintained in either direction. This feature eliminates the necessity of installing a check valve to stop reverse flow.

The hydraulic valve actuator is supported by the yoke, which is attached to the top of the body. The valve actuator consists of a hydraulic cylinder with a stored energy system to provide emergency closure of the isolation valve. The energy to operate the valve is stored in the form of compressed nitrogen contained in a spheroidal volume on one end of the actuator cylinder. The pressure of the nitrogen is 2500 psig maximum at 110°F with the piston fully retracted. The stored energy volume communicates directly with the actuator piston, which has a redundant sealing system to ensure that there is no leakage of nitrogen while the unit is in the standby mode. The piston is designed to protect the backseats of the valve. This is done by having a piston within a piston, which allows the main piston to seat on the piston stop ring. The floating piston (actually the top end of the actuator rod) regulates backseating force because its pressurized area is significantly smaller than the main piston. The lower volume of the cylinder, on the other side of the piston, is filled with hydraulic fluid. This serves three basic purposes: to be throttled through a control system to control closing speed; to provide a simple, reliable method of reopening the valve; and, most importantly, to hold the unit in the standby position. The latter is accomplished by holding the fluid under high pressure, also with the hydraulic control system. The operating pressure in the oil side of the cylinder is 4300 psig maximum at 110°F. When the actuator extends to close the valve, hydraulic fluid flows from the cylinder into the hydraulic control system through cavitating venturi orifice inserts that can limit closing speed in the event of a catastrophe which causes one or both control manifolds to be severed from the body of the cylinder. The fluid flows through the control systems into a reservoir on the front of the actuator. The pump which reopens the actuator is mounted on this reservoir and is connected directly to one of the control manifolds. There is no external piping which would prevent emergency closing of the actuator should the piping be damaged.

System Media Actuated MSIV Characteristics

Each system media actuated MSIV is a bidirectional gate valve composed of a valve body which is welded into the system pipeline. The MSIV actuator consists of a pilot assembly that is attached to the top of the body. The pilot is a part of the pressure boundary of the MSIV. Within the pilot, there is a piston that is attached to the valve stem. The valve stem is attached directly to the valve's gates. The valve is divided into three chambers; an upper piston chamber (UPC), a lower piston chamber (LPC), and the valve body. Internal ports channel system media from the valve body to the UPC. The flow of system media through these ports is controlled by a number of solenoid operated valves (SOV). When the valve is in the open position, the UPC is vented to atmosphere and the SOV isolates the UPC from system media. The hydraulic force of the system media acting on the cross section of the stem keeps the MSIV in open position. When the valve is given the command to close, the SOV directs system media into the UPC. The system media pressure acting over the area of the piston overcomes the hydraulic force acting on the stem, resulting in closure of the MSIV. Redundant SOVs are provided to ensure that single failure of one SOV does not result in inadvertent MSIV closure. The disc assembly has springs between the discs. When fluid hits the upstream disc, the upstream disc presses against the springs forcing the downstream disc into the seat. As the differential pressure across the disc increases, the seating load also increases, thus providing a tight seal throughout the entire range of operating differential pressures. A small amount of leakage is expected at low system operating pressures. Since the disks are completely independent of each other and the design is essentially symmetrical, positive sealing can be maintained in either direction. This feature eliminates the necessity of installing a check valve to stop reverse flow.

The system media MSIV actuator is operated by the system media present in the main steam line. The stroke time of the MSIV is dependent on the system pressure available. For system pressure >325 psia, the system media actuated main steam isolation valves close fully and remain fully closed within 7 s of the receipt of a manual or automatic signal. For system pressure ≤325 psia, the system media actuated main steam isolation valves close fully and remain fully closed within 40 s of the receipt of a manual or automatic signal. Both sets of closure times include a 1 second delay prior to valve stroke due to surge suppression incorporated in the actuation circuitry. Accident analyses that credit closure of the MSIV in 7 s

(Hot Zero Power, at power conditions), were determined to be limiting as compared to events initiated in Mode 3 at a SG secondary pressure ≤325psia.

The MSIV bypass valves are used when the MSIVs are closed to permit warming of the main steam lines prior to startup. The bypass valves are air-operated globe valves. For emergency closure, the valve solenoid, when deenergized, will result in valve closure. Electrical solenoids are energized from a separate Class IE source.

10.3.2.3 System Operation

10.3.2.3.1 Normal Operation

At low plant operating loads, the main steam supply system provides steam to the steam generator feedwater pump turbines, the main and steam generator feedwater pump turbines steam-seal systems, the condenser spargers, the steam jet air ejectors, the single-stage reheaters, and the auxiliary steam system. At high plant operating loads, a portion of the hot reheat steam leaving the single-stage reheater is used to drive the steam generator feedwater pump turbines. The transition from using all main steam at low loads to all hot reheat steam at high loads starts at below 10-percent load. No steam is required for the condenser spargers at high loads.

The main steam supply system is capable of accepting a ±10-percent step change in load followed by a ±5-percent/min ramp change without discharging steam to the atmosphere through the main steam safety valves or to the main condenser through the turbine bypass system. For large step change load reductions, steam is bypassed (40 percent of VWO) directly to the condenser via the turbine bypass system. The system is capable of accepting a 50-percent load rejection without reactor trip and a full-load rejection with turbine trip without lifting safety valves. If the turbine bypass system is not available, steam is vented to the atmosphere via the power-operated atmospheric relief valves and the main steam safety valves, as required.

10.3.2.3.2 Emergency Operation

In the event that the plant must be shut down and offsite power is lost, the MSIVs with associated MSIV bypass valves and other valves (except to the auxiliary feedpump turbine) associated with the main steam lines are closed. The power-operated atmospheric relief valves are then used to remove reactor decay and primary system sensible heat at a cooldown rate averaging 50°F/h from an average primary system temperature of 550°F to 350°F, whereupon the RHR system performs the remaining cooldown function to 140°F to achieve cold shutdown. If the power- operated atmospheric relief valve for an individual main steam line is unavailable because of the loss of its control or power supply, the respective safety valves will provide overpressure protection. The remaining power-operated atmospheric relief valves are sufficient to achieve cold shutdown.

In the event that a design basis accident occurs which results in a large steam line break, the MSIVs with associated MSIV bypass valves automatically close.

Steam is automatically provided to the auxiliary feedwater pump turbine from one of two steam lines upon low-low level in two steam generators or loss of offsite power. Check valves are installed in the lines to the auxiliary feedwater pump turbine to ensure that only one steam generator will feed a ruptured main steam line.

The closure of either isolation valve in each of the four pairs of MSIVs and associated MSIV bypass valves will ensure that no more than one steam generator can supply a postulated break. Coordinated operation of the auxiliary feedwater system (subsection 10.4.9) and power-operated atmospheric relief valves or safety valves may be employed to remove sensible and reactor decay heat.

10.3.3 EVALUATION

A. Each main steam line is provided with safety valves that limit the pressure in the line to preclude over-pressurization and remove stored energy. Each line is provided with a power-operated atmospheric relief valve to permit reduction of the main steam line pressure and remove stored energy to achieve an orderly shutdown. The auxiliary feedwater system, described and evaluated in subsection 10.4.9, provides makeup to the steam generators consistent with the steaming rate.

B. Redundant power supplies and power trains operate the main steam isolation valves (MSIVs) and MSIV bypass valves to isolate safety- and nonsafety-related portions of the system. Branch lines upstream of the MSIVs contain normally closed, power-operated atmospheric relief valves which modulate open and closed on steam line pressure. In the event the atmospheric relief valves fail closed, the safety valves provide the overpressure protection.

   Accidental releases of radioactivity from the main steam supply system are minimized by the negligible amount of radioactivity in the system under normal operating conditions. Additionally, the main steam isolation system provides controls for reducing accidental releases, as discussed in chapter 15, following a steam generator tube rupture.

   Detection of radioactive leakage into the system, which is characteristic of a tube leak or rupture, is facilitated by inline radiation monitors on each steam header, the condenser air ejector radiation monitor, and steam generator blowdown sampling. Two additional primary-to-secondary leak detection systems are also provided: a noble gas detector and a system utilizing N16 as the detection medium. The N16 detector is installed in the turbine building main steam pipe chase, between the two main steam pipes. The noble gas detector is located in the condenser steam jet air ejector discharge header immediately prior to the filtration unit. Remote readout is provided at the integrated plant computer in the control room. For details of these radiation monitors see subsection 11.5.2.

C. Section 3.2 delineates the quality group classification and seismic category applicable to the safety-related portion of this system and supporting systems. Table 10.3.2-1 shows that the components meet the design and fabrication codes given in section 3.2. All the power supplies and controls necessary for safety-related functions of the main steam supply system are Class 1E, as described in chapters 7 and 8.

D. The safety-related portions of the main steam supply system are located in the containment and auxiliary buildings and the main steam isolation valve (MSIV) areas. These buildings and areas are designed to withstand the effects of earthquakes, tornadoes, hurricanes, floods, external missiles, and other appropriate natural phenomena. Sections 3.3, 3.4, 3.5, 3.7, and 3.8 provide the bases for the adequacy of the structural design of these buildings and areas.

The safety-related portions of the main steam supply system are designed to remain functional after a safe shutdown earthquake. Sections 3.7 and 3.9 provide the design loading conditions that were considered. Sections 3.5, 3.6, and 9.5 provide the hazards analyses to ensure that a safe shutdown can be achieved and maintained.

E. As indicated by the failure mode and effects analysis in table 10.3.3-1, no single failure coincident with loss of offsite power compromises the system's safety functions. All vital power can be supplied from either onsite or offsite power systems, as described in chapter 8.

F. The main steam supply system is initially tested with the program given in chapter 14. Periodic inservice functional testing is done in accordance with subsection 10.3.4.

Section 6.6 provides the American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section XI requirements that are appropriate for the main steam supply system.

G. The steam line to the auxiliary feedwater pump turbine is connected to a cross-connecting header upstream of the MSIVs for steam generators 1 and 2. This arrangement ensures a supply of steam to this turbine when the steam generators are isolated. Check valves are provided in each supply line from the main steam lines to preclude potential backflow during a postulated main steam line break. The auxiliary feedwater system is described in subsection 10.4.9.

H. The components of the main steam supply system are qualified to function in normal, test, and accident environmental conditions. The results of the environmental qualification program are provided in subsection 3.11.B.3.

I. A discussion of high energy pipe break locations and evaluation of effects is provided in section 3.6.

10.3.4 INSPECTION AND TESTING REQUIREMENTS

10.3.4.1 Preservice Valve Testing

The operability and relief setpoints of the main steam safety valves will be verified at temperature in a bench test using steam as the pressurization fluid. The advantages of this approach are: the testing at temperature will reduce the probability of having to adjust the valve setpoints during hot functional testing heatup; the hot functional testing has, at other plants resulted in valves not reseating because of the combination of the booster device and limited steam flow which barely opens the valve; and the booster device test is not particularly accurate and will probably require retesting and averaging of the data to obtain acceptable results.

The lift-point of each power-operated atmospheric relief valve is checked against pressure gauges mounted in the main steam piping.

The main steam isolation valves and main steam bypass isolation valves are checked for closing time prior to initial startup.

10.3.4.2 Preservice System Testing

Preoperational testing is described in chapter 14.

The main steam supply system is designed to include the capability for testing through the full operational sequence that brings the system into operation for reactor shutdown and for main steam line break accidents. This includes operation of applicable portions of the protection system and the transfer between normal and standby power sources.

The safety-related components of the system, i.e., valves and piping, are designed and located to permit preservice and inservice inspections to the extent practicable.

10.3.4.3 Preservice Pipe Testing

Main steam piping located in the main steam tunnels adjacent to the diesel generator building is volumetrically inspected in the shop or in the field. This inspection includes a 100-percent inspection of circumferential and longitudinal welds.

The main steam lines from the five-way restraint forgings in the isolation valve areas to the main steam tunnel five-way restraint forgings are classified as non-Seismic Category 1 piping. However, these lines are designed to withstand a safe shutdown earthquake, and 100 percent volumetric inspection at installation is provided (i.e., 100 percent volumetric examination of shop and field longitudinal and circumferential welds).

10.3.4.4 Inservice Testing

The performance and structural leaktight integrity of all system components are demonstrated by continuous operation.

The redundant actuator power trains of each main steam isolation valve are subjected to the following tests:

A. Closure time; the valves are checked for closure time at each refueling.

B. While the nuclear steam supply system is in operation, the operability of the hydraulic actuator may be checked periodically by exercising the valve to approximately 90 percent of full open. This test is not applicable to the system media actuated MSIVs.

C. Leakage past the piston and piston rod rings on the MSIV system media actuators is tested periodically to confirm that the observed leakage is within allowable limits used in dose analyses.

The main steam isolation valve bypass valves may be tested for closure time at power as discussed in paragraph 7.1.2.5.C.

Additional discussion of inservice inspection of American Society of Mechanical Engineers Code Class 2 and 3 components is contained in section 6.6. Also, pump and valve testing is discussed in subsection 3.9.6.

10.3.5 WATER CHEMISTRY

The objectives of the secondary side water chemistry program are as follows:

A. Minimizing general corrosion in the steam generators, turbine, and feedwater system by maintaining proper pH control and by minimizing oxygen ingress (coupled with oxygen scavenging).

B. Minimizing localized corrosion in the steam generators, turbine, and feedwater system by minimizing chemical contaminant ingress and by controlling contaminant levels through condensate polishing and steam generator blowdown.

10.3.5.1 Chemistry Control Basis

Electric Power Research Institute (EPRI) Pressurized-Water Reactor (PWR) Secondary Water Chemistry Guidelines recommendations are considered in the secondary side water chemistry program.

Secondary side water chemistry at VEGP is optimized as shown below:

A. Design and Control Phase

1. Selection of secondary side materials to minimize corrosive species such as copper oxides.
2. Capability of deaeration in the condensate storage tank and hotwell.
3. Capability of continuous blowdown of the steam generator bulk water.
4. Post-construction cleaning of the feedwater system followed by wet layup of the feedwater system and steam generators.

B. Operation Phase

1. Early identification of contaminant ingress (salts, corrosion products, and oxygen).
2. Full flow condensate filtration and demineralization.
3. Chemical addition to establish and maintain an environment that minimizes system corrosion.
4. Identification of action levels based on chemistry conditions.

10.3.5.2 Contaminant Ingress

Contaminant may be introduced into the secondary side water system through three major mechanisms: make-up water; condenser tube leaks; atmospheric leaks at the condenser or pump seals. The following methods are used to detect the ingress of contaminants in the secondary water system.

A. Demineralized water is continuously monitored as it is being produced in the water treatment plant. Then the condenser makeup water from the condensate storage tanks and from the condensate storage tank degasifiers are monitored routinely for contaminants.

B. Ionic contaminants are detected by monitoring (either continuous process monitors or sample analysis) the condensate pump discharge.

C. Atmospheric contamination is detected by monitoring the condensate pump discharge for dissolved oxygen.

10.3.5.3 Condensate Polishing

Full-flow condensate filter/demineralizers are provided to filter suspended corrosion products from the condensate and also to remove ionic contaminants. This polishing system will not necessarily be employed during all phases of plant operation, but when the polisher is in service, provisions are made for monitoring the common inlet/outlet from each vessel, and the total outlet.

The secondary side water system has provisions for recirculating feedwater to the condenser prior to and during startup. The polisher may be used during this phase to remove corrosion products from the feedwater and thus prevent their ingress into the steam generators.

10.3.5.4 Chemical Addition

VEGP employs volatile treatment plus ammonium chloride addition to minimize general corrosion in the feedwater system, steam generators, and main steam piping. Hydrazine, alternate amine, and ammonium chloride are the chemicals to be injected into the condensate system.

Alkaline conditions reduce the general corrosion rate of ferrous alloys, so a combination of hydrazine and an alternate amine provide these conditions. The alternate amine allows for better protection of the wet steam areas of the secondary system.

Hydrazine is added to scavenge dissolved oxygen present in the feedwater system. Hydrazine also promotes the formation of a protective magnetite layer on ferrous surfaces and to keep this layer in a reduced state, further inhibiting general corrosion.

Ammonium chloride is added to the secondary system to control the steam generator sodium to chloride molar ratio. The objective in controlling molar ratio is to maintain a neutral crevice where soluble contaminants concentrate, such as the tube to tube sheet crevices. The relationship between the bulk water molar ratio and the hideout return molar ratio can then be determined to control crevice chemistry.

Polyacrylic Acid (PAA) or equivalent chemical may be added to disperse iron in the secondary system and steam generators so that it is removed via the steam generator blowdown and does not remain in the steam generators.

10.3.5.5 Action Levels for Abnormal Conditions

Prompt and appropriate responses to abnormal chemistry conditions are prudent to assure the long-term integrity of secondary cycle components. As such, three action levels have been defined for taking remedial action when control parameters are observed and confirmed to be outside of limits.

In general, these actions will be consistent with the EPRI "PWR Secondary Water Chemistry Guidelines."

10.3.5.6 Layup and Heatup

VEGP plans no long-term steam generator layup under dry conditions. When maintenance or inspection is required on the secondary side of the steam generators, the steam generators will be drained hot under nitrogen atmosphere. After cooling, the nitrogen purge will be lifted and the maintenance/inspection will begin.

Then, wet layup conditions will be established for corrosion protection during outages. Guidelines are given in EPRI, PWR Secondary Water Chemistry Guidelines.

Before heatup to full power, the bulk water in the steam generators is normally brought into power operation specifications by draining and refilling or by feeding and bleeding. Guidelines for heatup are provided in EPRI, PWR Secondary Water Chemistry Guidelines.

10.3.5.7 Chemical Analysis Basis

The chemical parameters listed below play a significant role in the corrosion of the feedwater system and the steam generators. Each parameter will be addressed as indicated below.

A. Oxygen in the presence of moisture rapidly corrodes carbon steel. These corrosion products may be carried through the feedwater system and form a sludge pile in the steam generator. This sludge pile forms an ideal environment for localized corrosion mechanisms on steam generator tubes. Thus, concentration of oxygen should be kept as low as practical in the feedwater system, and it is prudent to control dissolved oxygen at the condenser and not allow it to be transported to the feedwater system. The oxygen concentration is measured by process analyzers in which an electric current is generated at an electrode in proportion to oxygen dissolved in a flowing sample stream. Oxygen is also measured on a grab sample by color comparison after reaction with a reagent.

B. In the absence of significant impurities, the pH is controlled by the concentration of alternate amine and hydrazine. Maintaining the pH within the recommended band results in minimal corrosion rates of ferrous materials. The pH is measured in both process and bench instruments which measure the potential across an electrode sensitive to hydrogen ions and a reference electrode.

C. By passing the sample through a cation resin column, conductive anions (such as chloride, sulfate, and to a lesser extent carbonate) can be indicated in a conductivity cell. Cation conductivity is a sensitive method for indicating soluble species which have been indicated in many localized corrosion mechanisms.

D. Sodium is an effective continuous indicator of many forms of containment ingress. The ability to analyze for the species at the sub ppb level makes it a very useful chemical tracer. Increased sodium levels can be indicative of condenser leakage or makeup water contamination. Sodium is measured by process analyzers using specific ion electrodes. Grab samples are analyzed by furnace atomic absorption.

E. Silica is an effective indicator of many forms of contaminant ingress. The ability to analyze for this species at the low ppb level makes it a very useful chemical tracer.

F. Chloride is aggressive to ferrous materials at steam generator operating conditions, particularly in crevice regions. It also has been identified as an aggravant relevant to inconel 600 pitting. Grab samples are analyzed for chloride by ion chromatography.

The chemical analysis methods listed above represent current technology applicable for power plant laboratories. These methods may be changed over the lifetime of the plant as new technology is proven to be more effective.

10.3.5.8 Sampling

Sampling points are identified in table 9.3.2-3 (grab sample points and process instrumentation) and table 9.3.2-4 (grab sample points). These sample points include hotwells, condensate storage tank, condensate, feedwater, auxiliary feedwater system, steam generator blowdown, reheat steam, and heater drains. Many of these points will be sampled and analyzed routinely, others only as needed for troubleshooting and problem diagnosis.

10.3.5.9 Condenser Inspection

The secondary side water chemistry program will include a comprehensive inspection program of the condenser and will be developed to ensure condenser integrity. This program includes a visual inspection of the condenser every refueling outage, waterbox inspection for tube leaks during plant operation, and component inspection for oxygen leaks during plant operation.

These water box inspections and component inspections will be performed as necessary to diagnose and troubleshoot abnormal chemistry levels.

10.3.5.10 Data Management and Corrective Actions for Out-of-Specification Condition

All analytical data will be recorded and stored in a retrievable fashion. The technician performing the analysis will compare the analyzed value to the specifications value, where specifications are established, and promptly report any values which are out of the established specification limits stated earlier to shift supervision. Analytical data for key parameters will be routinely trended.

When an analysis indicates an out-of-specification condition, laboratory personnel will promptly investigate the problem. Investigation may include confirmation of the reported value by resampling and reanalysis, checking on-line monitoring data where appropriate, and examining values of other parameters to obtain confirmation of the analyzed value.

Once confirmed, action to correct the out-of-specification condition will be initiated. The shift supervisor, utilizing technical help from the laboratory, will initiate corrective action based on the abnormal operating procedure.

Cases may exist, at all action levels, where prompt action by on-shift personnel can rapidly correct the out-of-specification condition. If the problem is promptly corrected, notification will not be necessary. Shift personnel will receive problem-solving assistance from site technical support personnel (chemists, plant engineers, maintenance personnel, etc.) and/or off-site personnel (vendors, consultants, and/or Corporate personnel) as appropriate to diagnose the root cause of out-of-specification occurrences and to apply corrective actions. VEGP management will assist both in technical resolutions and in assuring proper problem-solving support.

Corrective actions taken will depend on the specifics of the problem. Some typical corrective actions are listed below.

- Increase blowdown levels to lower chemical concentration in the steam generator water.
- Increase/decrease treatment chemicals addition to bring parameters back into specification.
- Locate and stop contaminant ingress; special samples and/or increased sampling frequency may be used to assist this effort.
- Decrease power or shutdown to limit potential damage while corrective action is being applied.
- Direct drains to the hotwell polishing.

10.3.5.11 Conformance to Branch Technical Position MTEB 5-3

VEGP conformance to Branch Technical Position MTEB 5-3 is discussed in table 10.3.5-1.

10.3.6 STEAM AND FEEDWATER SYSTEM MATERIALS

10.3.6.1 Fracture Toughness

Compliance with fracture toughness requirements of American Society of Mechanical Engineers (ASME) Code, Section III, Articles NC-2300 and ND-2300, is discussed in section 6.1.

10.3.6.2 Material Selection and Fabrication

All pipe, flanges, fittings, valves, and other piping material conform to the referenced ASME, American Society of Testing Materials (ASTM), American National Standards Institute (ANSI), or Manufacturer Standardization Society-Standard Practice (MSS-SP) code.

The following code requirements apply.

Component	Stainless Steel	Carbon Steel
Pipe	ANSI B36.19	ANSI B36.10
Fittings	ANSI B16.9, B16.11, or B16.28	ANSI B16.9, B16.11, or B16.28
Flanges	ANSI B16.5	ANSI B16.5

The following ASME material specifications apply specifically:

- ASME SA-155, GR KCF 70 Class 1 (impact tested).
- ASME SA-155, GR KCF 70 Class 1.

- ASME SA-106, GR C (impact tested).
- ASME SA-106, GR B.
- ASME SA-106, GR B (normalized).
- ASME SA-234, GR WPB.
- ASME SA-234, GR WPBW (manufactured from grade 70 plate).
- ASME SA-516, GR70.
- ASME SA-216, GR WCC.
- ASME SA-515, GR70.
- ASME SA-106, GR B (impact tested).
- ASME SA-234, GR WPC or WPCW (impact tested).
- ASME SA-234, GR WPCW.
- ASME SA-234, GR WPC.
- ASME SA-105.
- ASME SA-193, GR B7.
- ASME SA-194, GR 2H, or ASME SA-194, GR 7
- ASME SA-216, GR WCB.
- ASME SA-333, GR 6 (impact tested).
- ASME SA-420, GR WPL6 (impact tested).
- ASME SA-508, Class 1 (impact tested).
- ASME SA-312, TP 304.
- ASME SA-403, WP-304.
- ASME SA-403, WP-304 W.
- ASME SA-182, F-304.
- ASME SA 672, GR C70.

Conformance with the following regulatory guides is discussed in section 1.9:

- Regulatory Guide 1.31, Control of Ferrite Content in Stainless Steel Weld Metal.

- Regulatory Guide 1.36, Nonmetallic Thermal Insulation of Austenitic Stainless Steel.
- Regulatory Guide 1.37, Quality Assurance Requirement for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants. (Note: Regulatory Guide 1.37 has been replaced by NQA-1-1994 as described in the SNC Quality Assurance Topical Report (QATR)).
- Regulatory Guide 1.44, Control of the Use of Sensitized Stainless Steel.
- Regulatory Guide 1.50, Control of Preheat Temperatures for Welding of Low-Alloy Steels.
- Regulatory Guide 1.71, Welder Qualification for Areas of Limited Accessibility.
- Regulatory Guide 1.85, Materials Code Case Acceptability-ASME Section III, Division I.

TABLE 10.3.2-1 (SHEET 1 OF 2)

MAIN STEAM SUPPLY SYSTEM
DESIGN DATA

Steam Flow (lb/h)	100% Turbine Guarantee	Valves Wide Open	Power Uprate 100% Power
Per steam generator	3,790,260	3,972,247	3,980,000
Total	15,161,041	15,888,986	15,920,000

Operating/Design Pressures/Temperatures

100% turbine guarantee	1000 psia/545°F
No load (hot standby)	1087 psia/547°F
Design conditions	1200 psia/600°F

Main Steam Piping: See table 10.3.2-3.

Design Pressure Drop, Steam Generator Outlet to Turbine Stop Valves	40.0 psi

Steam Generator Flow Restrictor

Number per steam generator outlet nozzle	7
Throat size	6.03 in.
Total area	1.388 ft^2

Design Codes

From steam generators to restraint downstream of outboard MSIV plus associated branch connections up to and including first valve that is normally closed or capable of automatic closure	ASME B&PV Code, Section III, Class 2, Seismic Category 1
MSIV system media actuators	ASME B&PV Code, Section III, Class 2, Seismic Category 1
Remainder	ANSI B31.1,Power Piping

TABLE 10.3.2-1 (SHEET 2 OF 2)

Power-Operated Relief Valve

Number per main steam line	1
Normal set pressure	1120 psig
Design capacity	64,000 lb/h at 100 psia inlet pressure, 596,000 lb/h at 1107 psia inlet pressure
Code	ASME B&PV Code, Section III, Class 2, Seismic Category 1
Actuator	Electrohydraulic

TABLE 10.3.2-2

DESIGN DATA FOR MAIN STEAM SAFETY VALVES

Number per main steam line	5
Total number of valves required	20
Minimum relieving capacity at 110% of main steam design pressure	16,852,500 lb/h
Available relieving capacity	18,607,220 lb/h
Valve size	6 x 10 in.
Orifice area (each)	16.0 in.^{2}
Design code	ASME B&PV Code, Section III, Class 2, Seismic Category 1

Valve Number	Set Pressure (psig)	Relieving Capacity[(a)] (lb/h)
1	1185	930,361
2	1200	930,361
3	1210	930,361
4	1220	930,361
5	1235	930,361
Total capacity, per line		4,651,805
Total capacity, 4 lines		18,607,220

a. Based on system accumulation pressure of 1272 psig per paragraph NC-7826 of ASME B&PV Code Section III Division 1 1974 Edition Subsection NC, Class 2 components including addenda through summer 1975.

TABLE 10.3.2-3

DESCRIPTION OF MAIN STEAM PIPING

Segment	Applicable Steam Generators	Description	Dimensions (in.) Nominal OD	Nominal OD	Minimum ID
Steam generator outlet to penetration	All	26-in. extruded pipe	26.0	23.956	-
Penetration to first safety valve	All	28-in. forging	28.0 $\begin{pmatrix}+0.125\\-0.031\end{pmatrix}$	23.875	23.75
First safety valve to inboard MSIV	All	29.5-in. forging	29.5 $\begin{pmatrix}+0.125\\-0.031\end{pmatrix}$	23.875	23.625
Between MSIVs	All	29.5-in. forging	29.5 $\begin{pmatrix}+0.125\\-0.031\end{pmatrix}$	23.875	23.625
Outboard MSIV to forged section	All	28-in. forging	28.0 $\begin{pmatrix}+0.125\\-0.031\end{pmatrix}$	23.875	23.75
Forged section to common run	All	26-in. extruded pipe	26.0	23.956	-
Common run to branch point	1 and 4	38-in. extruded pipe	38.0	35.138	-
Common run to branch point	2 and 3	36-in. extruded pipe	36.0	33.282	-
Branch point to turbine stop valves	All	28-in. extruded pipe	28.0	25.808	-

TABLE 10.3.3-1 (SHEET 1 OF 13)

MAIN STEAM SYSTEM
FAILURE MODES AND EFFECTS ANALYSIS

Item No.	Description of Component	Safety Function	Plant Operating Mode	Failure Mode(s)	Method of Failure Detection	Failure Effect on System Safety Function Capability	General Remarks
1.	MSIVs HV-3006A and HV-3006B, normally open, fail closed, with actuator	Isolate steam generator No. 1 in the event of a main steam line break (MSLB) to prevent blowdown of more than one steam generator	A. All but MSLB	A. Fails closed or fails to open upon command	A. Position indicator on QMCB and QPCP. If valve closes spuriously, a turbine followed by reactor trip will occur.	A. None; plant goes to or remains in a safe shutdown condition.	Two MSIVs are provided on each main steam line and operated from separate Class 1E power sources. Therefore, failure of one train, including loss of SLIS, will not impair isolation capability.
			B. MSLB (Cond. IV event)	B. Fails to close upon steam line isolation signal (SLIS)	B. Position indicator on QMCB and QPCP	B. None; closure of either MSIV will isolate steam generator No. 1 and thus prevent blowdown of more than one steam generator.	
2.	MSIVs HV-3016A and HV-3016B	Same as item 1, except for steam generator No. 2	Same as items 1A and 1B	Same as items 1A and 1B	Same as items 1A and 1B	Same as items 1A and 1B, except for steam generator No. 2	Same as item 1.
3.	MSIVs HV-3026A and HV-3026B	Same as items 1A and 1B, except for steam generator No. 3	Same as items 1A and 1B	Same as items 1A and 1B	Same as items 1A and 1B	Same as items 1A and 1B, except for steam generator No. 3	Same as item 1.
4.	MSIVs HV-3036A and HV-3036B	Same as items 1A and 1B, except for steam generator No. 4	Same as items 1A and 1B	Same as items 1A and 1B	Same as items 1A and 1B	Same as items 1A and 1B, except for steam generator No. 4	Same as item 1.

TABLE 10.3.3-1 (SHEET 2 OF 13)

Item No.	Description of Component	Safety Function	Plant Operating Mode	Failure Mode(s)	Method of Failure Detection	Failure Effect on System Safety Function Capability	General Remarks
5.	Main steam power-operated relief valve (PORV) PV-3000, normally closed, fail closed electro-hydraulic valve	Relieve steam generator No. 1 overpressure during transients to minimize lifting of safety valves and to regulate steam generator No. 1 pressure during reactor startup and cooldown	A. Normal startup and cooldown	A1. Fails closed or fails to open upon command including spurious operation	A1. Position indicator on QMCB and PSDA, plus higher pressure in steam generator No. 1 than in the other three steam generators	A1. None; manual actuation of turbine bypass valve provides steam generator pressure control. Overpressure protection is provided by main steam safety valves.	
				A2. Fails open or fails to close upon command, including spurious operation	A2. Position indicator on QMCB and PSDA, low water level and/or pressure in steam generator No. 1, plus high temperature alarm (TE-3000) in exhaust stack	A2. None; maximum valve flow less than maximum allowed by Westinghouse to prevent excessive cooldown rate. PORV can be manually isolated. Plant shutdown can be effected with only one PORV out of four operable.	
			B. Plant shutdown with loss of offsite power (LOP)	B1. Fails closed or fails to open upon command	B1. Same as item 5A1	B1. None; only one operable PORV required for safe shutdown. Overpressure provided by main steam safety valves	
				B2. Fails open	B2. Same as item5A2	B2. None; same as item 5A2	

TABLE 10.3.3-1 (SHEET 3 OF 13)

Item No.	Description of Component	Safety Function	Plant Operating Mode	Failure Mode(s)	Method of Failure Detection	Failure Effect on System Safety Function Capability	General Remarks
			C. Power operation	C. Fails open	C. Same as 5A2	C. None; plant shutdown not required. Valve can be isolated for repair. See also item 5B1.	
			D. All	D. Valve leaks	D. High temperature alarm TE-3000 in exhaust stack	D. None; same as item 5C	
6.	PORV PV-3010	Same as item 5, except for steam generator No. 2	Same as items 5A, 5B, 5C, and 5D	Same as items 5A, 5B, 5C, and 5D, respectively	Same as items 5A, 5B, 5C, and 5D, except position indicators on QMCB and PSDB and temperature alarm TE-3010	None; same as items 5A, 5B, 5C, and 5D, respectively	
7.	PORV PV-3020	Same as item 5, except for steam generator No. 3	Same as items 5A, 5B, 5C, and 5D	Same as items 5A, 5B, 5C, and 5D, respectively	Same as items 5A, 5B, 5C, and 5D, except position indicators on QMCB and PSDB and temperature alarm TE-3020	None; same as items 5A, 5B, 5C, and 5D, respectively	
8.	PORV PV-3030	Same as item 5, except for steam generator No. 4	Same as items 5A, 5B, 5C, and 5D	Same as items 5A, 5B, 5C, and 5D, respectively	Same as items 5A, 5B, 5C, and 5D, except position indicators on QMCB and PSDB and temperature alarm TE-3030	None; same as items 5A, 5B, 5C, and 5D, respectively	

TABLE 10.3.3-1 (SHEET 4 OF 13)

Item No.	Description of Component	Safety Function	Plant Operating Mode	Failure Mode(s)	Method of Failure Detection	Failure Effect on System Safety Function Capability	General Remarks
9.	Steam generator blowdown (SGBD) isolation valve HV-7603A. Normally open, fail closed pneumatic valve	Isolates blowdown from steam generator No. 1 upon auxiliary feedwater start or be remote-manual operation	All	A. Fails closed or fails to open upon command	A. Position indicator on QMCB, plus zero flow indication, FIC-1171, on PSBP	A. None; blowdown is terminated, but blowdown has no safety function.	
				B. Fails open or fails to close upon command	B. Position indicator on QMCB, plus flow indication, FIC-1171, on PSBP	B. None; isolation of 3 of 4 steam generators sufficient for safe shutdown. If coincident with SGBD line break, HV-15216A will close automatically to effect isolation and prevent room overpressurization.	
10.	SGBD isolation valve HV-7603B	Same as item 9, except for steam generator No. 2	All	Same as items 9A and 9B	Same as items 9A and 9B, except flow indication provided by FIC-1172 on PSBP	None; same as items 9A and 9B, except back up isolation provided by HV-15216B	
11.	SGBD isolation valve HV-7603C	Same as item 9, except for steam generator No. 3	All	Same as items 9A and 9B	Same as items 9A and 9B, except flow indication provided by FIC-1173 on PSBP	None; same as items 9A and 9B, except back up isolation provided by HV-15216C	
12.	SGBD isolation valve HV-7603D	Same as item 9, except for steam generator No. 4	All	Same as items 9A and 9B	Same as items 9A and 9B, except flow indication provided by FIC-1174 on PSBP	None; same as items 9A and 9B, except back up isolation provided by HV-15216D	

TABLE 10.3.3-1 (SHEET 5 OF 13)

Item No.	Description of Component	Safety Function	Plant Operating Mode	Failure Mode(s)	Method of Failure Detection	Failure Effect on System Safety Function Capability	General Remarks
13.	SGBD isolation valves HV-15216A, HV-15212A normally open, fail closed pneumatic valve	Isolates blowdown from steam generator No. 1 upon SGBD line break in auxiliary building	All	A. Fails closed	A. Position indicator on QPCP, plus zero flow indication, FIC-1171, on PSBP	A. None; same as item 9A	
				B. Fails open coincident with line break in auxiliary building	B. Position indicator on QPCP, plus high compartment pressure/temperature alarm	B. None; closure of either SGBD isolation valve will isolate SG no. 1 and prevent room high temperature and overpressurization.	
14.	SGBD isolation valves HV-15216B and HV-15212B	Same as item 13, except for steam generator No. 2	All	Same as items 13A and 13B	Same as items 13A and 13B, except flow indication provided by FIC-1172 on PSBP	None; same as items13A and 13B, except SG no. 2.	
15.	SGBD isolation valves HV-15216C and HV-15212C	Same as item 13, except for steam generator No. 3	All	Same as items 13A and 13B	Same as items 13A and 13B, except flow indication provided by FIC-1173 on PSBP	None; same as items13A and 13B, except SG no. 3	
16.	SGBD isolation valves HV-15216D and HV-15212D	Same as item 13, except for steam generator No. 4	All	Same as items 13A and 13B	Same as items 13A and 13B, except flow indication provided by FIC-1174 on PSBP	None; same as items13A and 13B, except SG no. 4	
17.	SGBD sample isolation valve HV-9451, normally open, fail closed solenoid valve	Isolates SGBD sample from steam generator No. 1 upon auxiliary feedwater start or by remote-manual control	All	A. Fails closed	A. Position indicator on QPCP	A. None; sampling terminated, but not required for safety	
				B. Fails open	B. Same as item 17A	B. None	

TABLE 10.3.3-1 (SHEET 6 OF 13)

Item No.	Description of Component	Safety Function	Plant Operating Mode	Failure Mode(s)	Method of Failure Detection	Failure Effect on System Safety Function Capability	General Remarks
18.	SGBD sample isolation valve HV-9452	Same as item 17, except for steam generator No. 2	All	Same as items 17A and 17B	Same as item 17A	None; same as items 17A and 17B	
19.	SGBD sample isolation valve HV-9453	Same as item 17, except for steam generator No. 3	All	Same as items 17A and 17B	Same as item 17A	None; same as items 17A and 17B	
20.	SGBD sample isolation valve HV-9454	Same as item 17, except for steam generator No. 4	All	Same as items 17A and 17B	Same as item 17A	None; same as items 17A and 17B	
21.	SGBD sample isolation HV-9553B, normally open, fail closed solenoid valve	Isolates SGBD sample from steam generator No. 1 upon remote-manual operation	All	A. Fails closed	A. Position indicator on PRP	A. None; same as item 17A	
				B. Fails open	B. Same as item 21A	B. None	
22.	SGBD sample isolation valve HV-9554B	Same as item 21, except for steam generator No. 2	All	Same as items 21A and 21B	Same as item 21A	None; same as items 21A and 21B	
23.	SGBD sample isolation valve HV-9555B	Same as item 21, except for steam generator No. 3	All	Same as items 21A and 21B	Same as item 21A	None; same as items 21A and 21B	
24.	SGBD sample isolation valve HV-9556B	Same as item 21, except for steam generator No. 4	All	Same as items 21A and 21B	Same as item 21A	None; same as items 21A and 21B	

TABLE 10.3.3-1 (SHEET 7 OF 13)

Item No.	Description of Component	Safety Function	Plant Operating Mode	Failure Mode(s)	Method of Failure Detection	Failure Effect on System Safety Function Capability	General Remarks
25.	SGBD sample isolation valve HV-9553A, normally closed, fail closed solenoid valve	Isolates SGDB (bottom) from steam generator No. 1 upon remote-manual actuation	All	A. Fails closed or fails to open upon command	A. Position indicator on PRP	A. None; same as item 17A	
				B. Fails open	B. Same as item 25A	B. None	
26.	SGBD sample isolation valve HV-9554A	Same as item 25, except for steam generator No. 2	All	Same as items 25A and 25B	Same as item 25A	None; same as items 25A and 25B	
27.	SGBD sample isolation valve HV-9555A	Same as item 25, except for steam generator No. 3	All	Same as items 25A and 25B	Same as item 25A	None; same as items 25A and 25B	
28.	SGBD sample isolation valve HV-9556A	Same as item 25, except for steam generator No. 4	All	Same as items 25A and 25B	Same as item 25A	None; same as items 25A and 25B	
29.	Steam generator chemical addition valve HV-5278, normally closed, fail closed pneumatic valve	Isolates chemical addition to steam generator No. 1 upon remote-manual actuation	All	A. Fails closed or fails to open upon command	A. Position indicator on QPCP	A. None; chemical addition is not possible, but this function is not required for safe shutdown	Chemical addition required only during cold shutdown.
				B. Fails open	B. Same as item 29A	B. None	
30.	Steam generator chemical addition valve HV-5279	Same as item 29, except for steam generator No. 2	All	Same as items 29A and 29B	Same as item 29A	None; same as items 29A and 29B	

TABLE 10.3.3-1 (SHEET 8 OF 13)

Item No.	Description of Component	Safety Function	Plant Operating Mode	Failure Mode(s)	Method of Failure Detection	Failure Effect on System Safety Function Capability	General Remarks
31.	Steam generator chemical addition valve HV-5280	Same as item 29, except for steam generator No. 3	All	Same as items 29A and 29A	Same as item 29A	None; same as items 29A and 29B	
32.	Steam generator chemical addition valve HV-5281	Same as item 29, except for steam generator No. 4	All	Same as items 29A and 29A	Same as item 29A	None; same as items 29A and 29B	
33.	Steam supply valve HV-3009 from steam generator No. 1 to turbine-driven auxiliary feedwater pump (TDP) turbine, normally open, dc-powered MOV	Admits steam from steam generator No. 1 to TDP turbine and also provides for isolating the downstream piping in the event of a pipe break	A. Hot standby and cooldown	A. Fails closed or fails to open upon command (remote-manual)	A. Valve position indicators on QMCB and PAFP	A. None; 100% redundant steam supply to the TDP provided via HV-3019 from steam generator No. 2. Also, two motor-driven auxiliary feedwater pumps provide 100% of auxiliary feedwater requirements.	For discussion of valve failure impact on auxiliary feedwater system, see table 10.4.9-4.
			B. All	B. Fails open coincident with break in downstream piping	B. Same as item 33A	B. None; manual closure of HV-3019 will prevent blowdown of more than one steam generator.	
34.	Steam supply valve HV-3019 from steam generator No. 2 to TDP turbine	Same as item 33, except for steam generator No. 2	Same as items 33A and 33B	Same as items 33A and 33B, respectively	Same as item 33A	Same as item 33A and 33B, except that HV-3009 from steam generator No. 1 provides redundancy	See item 33.

TABLE 10.3.3-1 (SHEET 9 OF 13)

Item No.	Description of Component	Safety Function	Plant Operating Mode	Failure Mode(s)	Method of Failure Detection	Failure Effect on System Safety Function Capability	General Remarks
35.	Check valve 006 in steam supply to TDP turbine from steam generator No. 1 (Valve internals are removed. Valve is not active.)	None. Valve internals removed.	N/A	N/A	N/A	N/A	See item 33.
36.	Check valve 008 in steam supply to TDP turbine from generator No. 2	Same as item 35, except interchange steam generators No. 1 and No. 2	All	A. Fails open coincident with MSLB from steam generator No. 2	A. Same as item 35A, except for steam generator No. 1	A. Same as item 33A, except HV-3019 provides protection of steam generator No. 1	See items 35 and 33.
37.	Main steam safety valves PSV-3001, PSV-3002, PSV-3003, PSV-3004, and PSV-3005	Protect steam generator No. 1 from overpressurization	All	A. Fails to open when required	A. Higher pressure and/or water level in steam generator. No. 1	A. None; 4 of 5 safety valves for No. 1 steam generator still available, with PORV PV-3000 available to supplement relieving capacity. Also, plant trip occurs upon high steam generator water level.	
				B. Spurious opening or failure to reset after opening	B. High temperature alarm in exhaust stack	B. None; maximum flow from one valve less than maximum specified by Westinghouse to prevent excessive cooldown rate.	

TABLE 10.3.3-1 (SHEET 10 OF 13)

Item No.	Description of Component	Safety Function	Plant Operating Mode	Failure Mode(s)	Method of Failure Detection	Failure Effect on System Safety Function Capability	General Remarks
38.	Main steam safety valves PSV-3011, PSV-3012, PSV-3013, PSV-3014, and PSV-3015	Protect steam generator No. 2 from overpressurization	All	A. Same as item 37A	A. Same as item 37A except for steam generator No. 2	A. None; same as item 37A except PORV is PV-3010	
				B. Same as item 37B	B. Same as item 37B	B. None; same as item 37B	
39.	Main steam safety valves PSV-3021, PSV-3022, PSV-3023, PSV-3024, and PSV-3025	Protect steam generator No. 3 from overpressurization	All	A. Same as item 37A	A. Same as item 37A except for steam generator No. 3	A. None; same as item 37A except PORV is PV-3020	
				B. Same as item 37B	B. Same as item 37B	B. None; same as item 37B	
40.	Main steam safety valves PSV-3031, PSV-3032, PSV-3033, PSV-3034, and PSV-3035	Protect steam generator No. 4 from overpressurization	All	A. Same as item 37A	A. Same as item 37A except for steam generator No. 4	A. None; same as item 37A except PORV is PV-3030	
				B. Same as item 37B	B. Same as item 37B	B. None; same as item 37B	
41.	Manual valves 001 and 136, normally open	Isolate PV-3000 (item 5, steam generator No. 1) for repair	All	Inadvertent closure	None; PV-3000 (item 5) incapacitated	None; same as items 5A1 and 5B1	
42.	Manual valves 002 and 137, normally open	Isolate PV-3010 (item 6, steam generator No. 2) for repair	All	Inadvertent closure	None; PV-3010 (item 6) incapacitated	None; same as items 5A1 and 5B1	

TABLE 10.3.3-1 (SHEET 11 OF 13)

Item No.	Description of Component	Safety Function	Plant Operating Mode	Failure Mode(s)	Method of Failure Detection	Failure Effect on System Safety Function Capability	General Remarks
43.	Manual valves 003 and 138, normally open	Isolates PV-3020 (item 7, steam generator No. 3) for repair	All	Inadvertent closure	None; PV-3020 (item 7) incapacitated	None; same as items 5A1 and 5B1	
44.	Manual valves 004 and 139, normally open	Isolates PV-3030 (item 8, steam generator No. 4) for repair	All	Inadvertent closure	None; PV-3030 (item 8) incapacitated	None; same as items 5A1 and 5B1	
45.	Manual valve 005, normally open	Isolates HV-3009 (item 33) for repair	All	Inadvertent closure	None; HV-3009 (item 33) incapacitated	None; same as item 33A	See table 10.4.9-4 FMEA for impact on auxiliary feedwater system. See item 33.
46.	Manual valve 007, normally open	Isolates HV-3019 (item 34) for reparation	All	Inadvertent closure	None; HV-3019 (item 33) incapacitated	None; same as item 33A	Same as item 45. See item 34.
47.	MSIV bypass valves HV-13005A and HV-13005B, normally open (N.O.), failed closed (F.C.), pneumatic valves	Valves are used when MSIV are closed to permit warming the MSS of S/G no. 1 prior to startup and to drain the portion of the MS line located upstream of MSIV during startup and normal operation	A. All but MSLB	A. Fails closed or fails to open upon command	A. Position indicator and alarm on QMCB. If valve closes spuriously, a light is turned on and operator takes necessary action to open valve	A. Inability of the operator to correct a closed MSIV bypass valve during startup can cause water to accumulate in the bypass line. This water can be drained via the drain valve when the bypass valve is open	
			B. MSLB (Cond. IV event)	B. Fails to close on steam line isolation signal (SLIS)	B. Position indicator on QMCB.	B. None; closure of either MSIV bypass valve will isolate S/G no. 1 and thus prevent blowdown of more than one S/G	Two MSIV bypass valves are provided on each main steam bypass line and operated from separate Class 1E power source; therefore, failure of one train including loss of SLIS will not impair isolation capability

TABLE 10.3.3-1 (SHEET 12 OF 13)

Item No.	Description of Component	Safety Function	Plant Operating Mode	Failure Mode(s)	Method of Failure Detection	Failure Effect on System Safety Function Capability	General Remarks
48.	MSIV bypass valves HV-13007A and HV-13007B	Same as item no. 47 except for S/G no. 2	Same as 47A and 47B	Same as items 47A and 47B	Same as items 47A and 47B	Same as 47A and 47B except for S/G no. 2	Same as 47B
49.	MSIV bypass valves HV-13008A and HV-13008B	Same as item no. 47 except for S/G no. 3	Same as 47A and 47B	Same as items 47A and 47B	Same as items 47A and 47B	Same as 47A and 47B except for S/G no. 3	Same as 47B
50.	MSIV bypass valves HV-13006A and HV-13006B	Same as item no. 47 except for S/G no. 4	Same as 47A and 47B	Same as items 47A and 47B	Same as items 47A and 47B	Same as 47A and 47B except for S/G no. 4	Same as 47B
51.	Check valve no. 404 in steam supply to TDP turbine from steam generator no. 1	Prevents cross flow between main steam lines to prevent blowdown of S/G no. 2 in event of a break in the steam line branch of AFW turbine from S/G no. 1 within the auxiliary FW pump building	All	A. Fails open coincident with branch line break within AFW pump building	A. Gradual decrease of water level in S/G no. 2	A. None; HV-3009 can be manually closed to prevent blowdown of S/G no. 2 while ensuring a steam supply to the TDP turbine	See item no. 33
				B. Fails closed	B. None	B. None; same effect as item 33A	
				C. Fails open coincident with MSLB from S/G no. 1	C. None	C. None; HV-3009 can be manually closed in the event that level in S/G no. 2 is decreasing	

TABLE 10.3.3-1 (SHEET 13 OF 13)

Item No.	Description of Component	Safety Function	Plant Operating Mode	Failure Mode(s)	Method of Failure Detection	Failure Effect on System Safety Function Capability	General Remarks
52.	Main steam isolation valve (MSIV) hydraulic reservoir desiccant type filler/ breather cap (Paul Munroe P/N 93334). This desiccant type filler/ breather cap screws into an adapter which is mounted on the MSIV's hydraulic reservoir cover plate. (not applicable to Unit 2 Train A MSIVs)	None; the desiccant type filler/ breather cap retains moisture and other particles as the MSIVs stroke open and closed. The filler/ breather cap serves as an air exhaust path.	1, 2, & 3	A. The filler/ breather's polypropelene, PVC, and desiccant components will degrade during accident conditions, when maximum temperature and radiation levels are reached. During a seismic event, the breather may fall off or its standpipe may bend.	A. Periodic surveillance checks as well as post-accident walkdowns will ensure failure detection.	A. None; the filler/breather is assembled with a stainless steel standpipe such that degraded material will not enter the hydraulic reservoir. A bent standpipe during a seismic event will cause the reservoir's displaced air to exhaust through the hydraulic reservoir cover nonpressure tight gasket.	The desiccant type filler/ breather cap is classified as nonsafety related, nonseismic Category II.

TABLE 10.3.5-1 (SHEET 1 of 4)

CONFORMANCE TO BRANCH TECHNICAL POSITION MTEB 5-3, MONITORING OF SECONDARY SIDE WATER CHEMISTRY IN PWR STEAM GENERATORS

Branch Technical Position	
1. Crevices between the tubing and the tube sheets or tubing supports should be minimized to prevent concentration of impurities or solids in these areas. To achieve this goal the tubes at the tube/tube sheet interface should be expanded for the full depth of the tube sheet. To minimize the deposition of corrosion products and sludge between the tubes and the supporting structure, the tube/tube support interface should be designed to promote high velocity water flow at the interface. This would improve the washing of this area.	1. Conform. Refer to 5.4.2.4.
2. Regulatory Guide 1.37 endorses ANSI N45.2.1 and states in part, "The surface (of components) shall appear metal clear. Scattered areas of rust are permissible provided the aggregate area of rust does not exceed two square inches in any one square foot area." Experimental work has shown that a porous packing of oxide in the tube support annulus is one of the conditions resulting in the concentration of contaminants, which leads to runaway corrosion of the tube support plate. Nuclear plant operators should start up the steam generators with metal clean surfaces. A method of confirmation such as photographing the inside of the steam generator should be undertaken after hot functional testing to confirm the metal clean condition.	2. Conform. Refer to section 1.9 and paragraph 5.4.2.4. (Note: Regulatory Guide 1.37 has been replaced by NQA-1-1994 as described in the SNC Quality Assurance Topical Report (QATR)).
3. A. In the FSAR, the applicant should describe implementation of a secondary water chemistry and monitoring program (in accordance with reference nuclear steam system supplier's recommended procedure) to inhibit steam generator corrosion and tube degradation. This program should cover the following operational modes: power operation (normal), startup, hot standby, hot shutdown, and cold shutdown/ cold wet layup Each of the above modes should be defined with regard to percent rated thermal power and approximate temperature range, °F.	3.A. Conform. Refer to paragraphs 10.3.5.5 and 10.3.5.6. Table 1.1 in chapter 16 lists the percent rated thermal power and temperatures for the appropriate operational modes.
B. The secondary water chemistry monitoring and control program should include the following:	3.B. Conform, except additives are injected into the condensate providing protection for the condensate/feedwater system. Concentrations in each steam generator can be controlled to a certain extent with blowdown.

TABLE 10.3.5-1 (SHEET 2 of 4)

	Identification of a sampling schedule for critical parameters during each mode of operation and of acceptance control criteria for these parameters. The program should include as a minimum the control of pH, cation conductivity, free sodium, and dissolved oxygen. However, other parameters such as specific conductivity, chlorine, fluorine, suspended solids, silica, total iron, copper, ammonia, and residual hydrazine merit consideration. In plants having more than one steam generator, additives to each steam generator should be controlled separately.		
C.	(1) The Nuclear Regulatory Commission will review the secondary water chemistry control and monitoring program of each individual plant. The applicant should incorporate the technical recommendations of the steam generator supplier. Any significant deviation from the supplier's recommendations should be noted and justified technically.	3.C.(1)	Conform.
	(2) Records should be made of the monitored item values, and in accordance with 10 CFR 50.71(a), they shall be made available for audit and inspection when deemed necessary.	3.C.(2)	Conform.
	(3) Each licensee as part of his annual operating report should include an evaluation of the secondary side water chemistry program with an evaluation of the trends and a summary of the total time during the reporting period the various chemistry parameters were out of specification.	3.C.(3)	Do not conform. Refer to section 1.9.16.
D.	For plants utilizing volatile chemistry		
	(1) The composition, quantities, and addition rates of additives should be recorded. Routine changes in these items should be reported under the biannual FSAR update as required by 10 CFR 50.71. However, nonconservative changes, i.e., relaxation in sample frequency or changes in impurity limits, shall be submitted to the NRC for approval before the change is implemented	3.D.(1)	Control of additives (types, amounts, impurities, sampling frequencies, etc.) is achieved in accordance with the VEGP procedure system described in section 13.5.
	(2) The electrical conductivity and the pH of the bulk steam generator water and feedwater should be measured continuously. Assurance should be provided that the sample taken at the blowdown is typical of the bulk steam generator water and that there is a minimum bypass between the feedwater inlet and the blowdown sampling point.	3.D.(2)	Conform. Refer to subsection 9.3.2.
	(3) For once-through steam generators, the pH and electrical conductivity at the coolant inlet should be measured continuously.	3.D.(3)	Not applicable to VEGP.

TABLE 10.3.5-1 (SHEET 3 of 4)

	(4) Free hydroxide concentration and impurities (particularly chloride, ammonia, and silica) in the steam generator water should be measured at least three times per week.	3.D.(4)	Ammonia is not an impurity in the secondary side system as the alternate amine added will degrade to ammonium and hydroxide. Therefore, we do not plan to monitor free hydroxide in the steam generator water. Other parameters such as pH, pH/conductivity relationships, and sodium levels will be used to minimize corrosion. Chloride will be measured at least three times per week. Silica will be analyzed as a diagnostic parameter on an as-needed basis for troubleshooting
E.	For plants utilizing phosphate treatment.	3.E.	Not applicable to VEGP.
F.	For all PWR plants:		
	(1) Condenser cooling water inleakage to the condensate has been identified as the major source of impurity ingress in the PWR secondary feedwater. The combination of impurity ingress with corrosion of copper containing alloys and corrosion product transport (Fe_30_4, $Ni0_2$, etc.) in the secondary water system produces a sludge that is difficult to remove and is reactive to steam generator materials. In reporting the program the following guidelines should be observed:		
	a. Monitor the condensate water quality at the condensate pump discharge as a minimum. Supplement as necessary by samples from the condenser hot well and condenser discharge.	3.F.(1)a.	Conform. Refer to subsection 9.3.2.
	b. Measure the cation conductivity and oxygen.	3.F.(1)b.	Conform. Refer to subsection 9.3.2.
	c. Maintain condensate impurity level at 0.1 ppm ± 0.05 ppm, oxygen at ≤ 5 ppb.	3.F.(1)c.	Condensate impurity level and oxygen are monitored and their control limits are outlined in the secondary water chemistry and monitoring program.
	d. A cation conductivity increase of 0.05 to 0.10 ˜mh o/cm justifies online investigation of possible contamination	3.F.(1)d.	An increase in control limit of cation conductivity which could initiate investigation of possible contamination is outlined in the secondary water chemistry and monitoring program.

TABLE 10.3.5-1 (SHEET 4 of 4)

	e. An increase of 0.10 to 0.20 μmho/cm is considered an indication of condenser leakage.	3.F.(1)e.	An increase of 0.10 to 0.20 μmho/cm in cation conductivity will initiate investigation of possible condenser leakage. The investigation procedure is a part of the secondary water chemistry and monitoring program.
	f. When a condenser leak is confirmed, the leak should be repaired or plugged within 96 h or before the total integrated conductivity increase reaches 20 μmho/cm h. The staff will consider other impurity-time limit proposals for limiting the quantity of impurities entering the steam generator.	3.F.(1)f.	Condenser leak identification and repair are outlined in the secondary water chemistry and monitoring program.
(2)	Identify the procedures used to measure the value of each of the critical parameters. Provide the procedure title, the applicant/licensee's procedure number, and the basis (i.e., ASTM No.).	3.F.(2)	Conform.
(3)	Identify sampling points. The program should consider sampling the steam generator blowdown, the hot well discharge, the feedwater, and the demineralizer effluent as a minimum of sampling points	3.F.(3)	Conform. Refer to subsection 9.3.2
(4)	State the procedure for recording and management of data.	3.F.(4)	Conform.
(5)	State the procedures defining corrective action for various out-of-specification parameters. The procedures should define the allowable time for correction of out-of-specification chemistry.	3.F.(5)	Conform.
(6)	Identify the authority responsible for the interpretation of the data and the sequence and timing of administrative events required to initiate corrective action.	3.F(6)	Conform.
(7)	Identify major components of the secondary water system and materials in contact with secondary water coolant.	3.F.(7)	Conform. See section 10.3.6.

10.4 OTHER FEATURES OF STEAM AND POWER CONVERSION SYSTEM

This section provides discussions of each of the principal design features of the steam and power conversion system.

10.4.1 MAIN CONDENSERS

The main condenser functions as the steam cycle heat sink, receiving and condensing exhaust steam from the main turbine, steam generator feed pump turbines, and the turbine bypass system.

10.4.1.1 Design Bases

10.4.1.1.1 Safety Design Bases

The main condenser serves no safety function and has no safety design bases.

10.4.1.1.2 Power Generation Design Bases

A. The main condenser is designed to receive and condense the full-load main steamflow exhausted from the main turbine and the feedwater pump turbines.

B. The main condenser is designed to serve as a collection point for vents and drains from various components of the steam cycle system.

C. The main condenser is designed to receive and condense steam bypass flows up to 40 percent of guaranteed maximum steamflow at the same time the remaining flow is through the turbine without exceeding 5-6 in. Hg absolute pressure at the turbine exhaust flange.

D. The main condenser hotwells shall be sized to provide surge storage for the condensate system equal to that required for 5 min of operations at maximum guaranteed load.

E. The main condenser is designed to deaerate the condensate so that dissolved oxygen content of the condensate will not exceed 7 ppb over the entire load range.

10.4.1.2 System Description

The main condenser is a three-shell, two-pass, divided water box, floor-supported unit. Each shell is located beneath its respective low-pressure turbine. The condenser is equipped with 1-in.-diameter titanium tubes. The outermost two rows of tubes in the entering steam sections are 18 Birmingham wire gauge (BWG), while the remainder are 22 BWG. The titanium material has good corrosion resisting properties. Design parameters of the condenser are listed in table 10.4.1-1.

Exhaust steam from the steam generator feedwater pump turbines is used to reheat the condensate in the condenser. Each hotwell is divided longitudinally by a vertical partition plate. The condensate pumps take suction from these hotwells.

The condenser shells are located below the turbine building operating floor and are supported above the turbine building foundation. Expansion joints are provided between each low-pressure turbine exhaust opening and the steam inlet connections of the condenser shell. The hotwells of the three shells are interconnected by steam-equalizing lines. Three low-pressure feedwater heaters are located in the steam dome of each shell. Piping is installed for hotwell level control and condensate sampling. The condensate system is shown in drawings 1X4DB168-1, 1X4DB168-2, and 1X4DB168-3.

10.4.1.2.1 System Operation

During normal power operation, exhaust steam from the low-pressure turbines is directed into the main condenser shells. The condenser also receives auxiliary system flows such as feedwater heater vents and drains, feedwater pump turbine exhaust, and gland sealing steam spillover and drains.

The hotwell level controllers provide automatic makeup or rejection of condensate to maintain a normal level in the condenser hotwells. On low level, the makeup control valves open and admit condensate, by gravity flow, to the hotwell from the condensate storage tank. On high water level in a hotwell, the condensate reject control valves open to divert water from the condensate pump discharge to the condensate storage tank. This rejection automatically stops when the hotwell level falls to within normal operating range. Rejection to the storage tank can be manually overridden upon an indication of high hotwell conductivity to prevent transfer of contaminants into the condensate storage tank in the event of a condenser tube failure.

The main condenser, with the assistance of main steam at low loads, deaerates the condensate so that dissolved condensate oxygen does not exceed 7 ppb over the entire load range. Air leakages are estimated to be no greater than 60 sf^3/min, in accordance with Table ST-4 of the Heat Exchange Institute Standards for Steam Surface Condensers. Both the air inleakage and the noncondensable gases contained in the turbine exhaust steam are collected in the condenser and removed by the main condenser evacuation system. The main condenser evacuation system is discussed in subsection 10.4.2.

To protect the condenser shells and turbine exhaust hoods from overpressurization, steam relief blowout diaphragms are provided in the main low-pressure turbine exhaust hoods and the main feedwater pump steam turbine exhausts. The diaphragms for the main feedwater pump turbines are designed to rupture at 15 psig, and the main low-pressure turbine diaphragms are designed to rupture at 5 psig.

Upon activation of the turbine bypass system, the main condenser receives up to 40 percent of full-load main steamflow from that system. The main condenser is designed to accept this without either increasing the condenser backpressure to the turbine trip setpoint or exceeding the allowable exhaust temperature during normal operational load swings. If the main condenser, because of tube failure, cooling water failure, air leakage, or other reasons, is unavailable to receive this flow, the steam bypass system controls automatically block the path to the condenser and the steam is discharged to the atmosphere through the main steam relief or safety valves.

Perforated distribution headers and internal baffles are incorporated to protect the condenser tubes and components from turbine bypass or high-temperature drains into the condenser shell.

Leakage at the connections of the tubes to the tube sheets, should it occur, can be detected at either end of each tube bundle by the collection troughs and conductivity cells (24 monitoring points are provided for the 3 shells). These conductivity measurements are indicated and alarmed in the control room. This information permits determination of which tube bundle has sustained the leakage. Steps may be taken to repair or plug the leaking tubes. Leakage occurring in tube locations other than at the tube ends is detected and alarmed in the control room by monitoring the condensate leaving each hotwell (six monitoring points altogether). Detection, isolation, and repair are performed as above. Refer to table 10.3.5-1 Branch Technical Position F.(1) for permissible cooling water inleakage and time of operation which ensures that the condensate/feedwater quality can be maintained.

A condenser tube cleaning system is provided for each shell to perform mechanical cleaning of the circulating water side of the titanium tubes. This removes fouling in order to maintain optimum performance of the condenser.

During normal operation and shutdown the main condenser has a negligible inventory of radioactive contaminants. Radioactive contaminants may enter through a steam generator tube leak. A discussion of the radiological aspects of primary-to-secondary leakage, including anticipated operating concentrations of radioactive contaminants, is included in chapter 11. No hydrogen buildup in the main condenser is anticipated.

The failure of the main condenser and the resultant flooding will not preclude operation of any essential system since no essential safety-related equipment is located in the turbine building (however, see paragraph 7.2.1.1.2.F for trip instrumentation in the turbine building), and the water cannot reach safety-related equipment located in other Category 1 plant structures. Refer to subsection 10.4.5.

10.4.1.3 Safety Evaluation

Since there are no safety design bases for the main condenser, no safety evaluation is provided.

10.4.1.4 Tests and Inspections

The main condenser water boxes are hydrostatically tested after erection. Condenser shells are tested by completely filling them with water and then testing by the fluorescent tracer method, in accordance with the American Society of Mechanical Engineers Power Test Code 19.11. All tube joints are leak tested during construction and, again, prior to startup. The main condenser is not subject to inservice inspection testing.

10.4.1.5 Instrumentation Applications

The main condenser hotwell is equipped with level control devices for automatic makeup or rejection of condensate. Remote indicators are provided for monitoring pressures and water levels in the condenser shells. Local indicators are also provided for monitoring the water levels in the condenser shells. High and low hotwell level alarms are provided in the control room.

Three pressure sensors are provided on each condenser zone to monitor condenser back pressure. An alarm for high pressure, set at 5-6 in. Hg absolute, and a turbine trip for high-high pressure, set at 7.5 in. Hg absolute, are provided.

Temperature indicators for monitoring condenser performance are provided.

10.4.2 MAIN CONDENSER EVACUATION SYSTEM

Main condenser evacuation is performed by the condenser air ejection system (CAES). The CAES removes noncondensable gases and air from the main condenser during plant startup, cooldown, and normal operation.

10.4.2.1 Design Bases

10.4.2.1.1 Safety Design Bases

The CAES serves no safety function and has no safety design bases.

10.4.2.1.2 Power Generation Design Bases

A. The CAES is designed to remove air and noncondensable gases from the condenser during plant startup, cooldown, and normal operation from the steam side of the three main condenser shells and to exhaust them to the atmosphere.

B. The CAES establishes and maintains a vacuum in the condenser during startup by the use of mechanical vacuum pumps. Steam jet air ejectors (SJAEs) are provided to remove noncondensables and hold vacuum during normal operation.

C. The CAES is designed to satisfy all applicable interface requirements and the design recommendations of both the turbine-generator and main condenser suppliers.

10.4.2.2 System Description

10.4.2.2.1 General Description

The CAES, as shown in drawing 1X4DB182, consists of two mechanical vacuum pumps and two two-stage SJAEs, which remove air and noncondensable gases from the three condenser shells during normal operation and provide condenser hogging during startup. The noncondensable gases, together with a quantity of vapor, are drawn from the condenser shell, through the air cooler section, to the suction of the air removal equipment. These noncondensables consist mainly of air, nitrogen, and ammonia. Since no hydrogen buildup is anticipated in the main condenser (as discussed in paragraph 10.4.1.2.1), likewise, no hydrogen buildup is anticipated in the CAES. Dissolved oxygen will be present in the condensate and condenser hotwell inventory. Only trace amounts of this oxygen will be released in the condenser, and the amounts are considered negligible compared to the large amounts of air being evacuated by the CAES. Therefore, the potential for explosive mixtures within the CAES does not exist. The condensate and feedwater system provide the cooling medium for the SJAE condensers, while the turbine plant cooling water system provides the cooling for the vacuum pump seal water heat exchangers. The vacuum pump seal water cooler uses turbine plant cooling water so that the seal water is kept cooler than the saturation temperature of the condenser at its operating pressure.

The noncondensable gases and vapor mixture discharged to the atmosphere from the system are not normally radioactive. However, it is possible for the mixture to become contaminated in the event of primary-to-secondary system leakage. Air inleakage and noncondensable gases are removed from the condenser during startup and normal operations by the mechanical vacuum pumps and steam jet air ejectors, respectively. The normal discharge flow path is directed to the unit vent through the charcoal adsorption train bypass. The discharge flow is monitored for radioactivity prior to discharge to the unit vent. Upon detection of radioactivity in the process stream, discharge flow will be automatically diverted through the charcoal adsorption train prior to discharge. (See subsection 9.4.4.) A discussion of the radiological aspects of primary-to-secondary leakage, including anticipated release from the system, is included in chapter 11.

As long as the CAES is functional, its operation does not affect the reactor coolant system. Should the air removal system fail completely, a gradual reduction in condenser vacuum would result from the buildup of noncondensable gases and air. This reduction in vacuum would cause a decrease in the turbine cycle efficiency necessitating an increase in reactor power to maintain a constant electrical power output. The reactor control system and the reactor protection system guarantee that reactor power is limited and maintained within safe operating limits. If the CAES remains inoperable, condenser vacuum decreases to the turbine trip setpoint and a turbine trip is initiated. A loss of condenser vacuum incident is discussed in subsection 15.2.5.

Loss of the main condenser vacuum causes a turbine trip but does not trip the main steam isolation valves. The causes of low condenser vacuum include air inleakage, poor heat transfer, and faulty air removal. Air inleakage is normally handled by the ejection system as described above. If significant air inleakage is suspected, the pump and valve seals vacuum gauge connections are checked for leaktightness. Faulty air removal by the air ejectors is checked by ensuring proper steamflow to the ejectors. In the unlikely event that air inleakage exceeds the removal capacity of the ejectors, the mechanical vacuum pumps, normally used only during startup, could be used to augment the air removal capability.

Poor heat transfer is caused by dirty tubes, air blanketing, inadequate circulating water, and high inlet circulating water temperature. As noted above, the interior of the tubes is kept clean by a tube cleaning system so dirty tubes are not likely to occur. Air blanketing is caused by insufficient venting of the circulating water system prior to operation. Normal startup procedures will eliminate this problem. The circulating water system is sized to provide adequate flow.

High inlet circulating water temperature may be caused by various atmospheric conditions. Under such conditions, power would have to be curtailed.

10.4.2.2.2 Component Description

A. Mechanical Vacuum Pumps

The mechanical vacuum pump rated capacity is 25 sf^3/min each at 1.0 in. Hg absolute with 7.5°F each subcooling. Each pump is driven with a 150-hp motor. For startup hogging service, each vacuum pump capability is 800 sf^3/min at 10 in. Hg absolute.

B. Vacuum Pump Seal Water Coolers

The vacuum pump seal water coolers are shell and tube heat exchangers. Seal water flows through the shell side of the coolers, and turbine plant cooling water flows through the tube side.

C. Steam Jet Air Ejectors

Two two-stage SJAEs are provided with each two-stage unit. They are capable of 60 sf^3/min at 1.0 in. Hg absolute with 7.5°F subcooling. The steam consumption for operating each two-stage SJAE is 4100 lb/h.

D. Steam Jet Air Ejector Condensers

One SJAE condenser, consisting of an inter- and after-condenser, services each two-stage SJAE. The steam condensed in the shell sides of the inter- and after-condensers is routed either to the main condenser hotwells or floor drain system, depending on chemistry requirements. The condensate from the condensate and feedwater system is pumped through the tube side of each SJAE inter- and after-condenser.

Piping and valves are carbon steel. All piping is designed to American National Standards Institute (ANSI) B31.1. In conformance with Regulatory Guide 1.26, the quality group and associated quality standards for the CAES are tabulated in table 3.2.2-1. The design parameters of the system are provided in table 10.4.2-1.

10.4.2.2.3 System Operation

During startup operation, air is rapidly removed from the condenser by operating the two condenser mechanical vacuum pumps.

During normal plant operation, noncondensable gases are removed from the condenser by the operation of one of the two two-stage SJAEs. Should the SJAE trip during normal holding operation, automatic switchover is provided to the vacuum pumps.

10.4.2.3 Safety Evaluation

The condenser air ejection system has no safety-related function.

10.4.2.4 Tests and Inspections

Construction quality assurance was performed by Ecolaire Heat Transfer Company quality assurance personnel. This was supplemented by an extensive surveillance program by SCS/Bechtel inspectors. Welders and weld procedures were qualified in accordance with ASME B&PV, Section IX. Tensile testing, magnaflux testing, hydro tests, and material certificates of compliance were provided by Ecolaire. Inspection prior to shipment of components was required and performed. In addition, inspections were also performed by GPC quality control personnel upon receipt of material and components. During the erection/installation of CAES components, quality assurance was performed by qualified GPC personnel.

Testing and inspection of the system is performed prior to plant operation.

Components of the system are continuously monitored during operation to ensure satisfactory operation. Periodic inservice inspections of the evacuation system are performed in conjunction with the scheduled maintenance outages.

10.4.2.5 Instrumentation Applications

Local indicating devices (e.g., pressure, temperature, and flow indicators) are provided as required for monitoring the system operation. Pressure switches are provided for automatic operation of the standby mechanical vacuum pump during normal operation.

Volumetric flow indication is provided locally to monitor the quantity of exhausted noncondensable gases.

A radiation detector is provided in the turbine building heating, ventilation, and air-conditioning system to monitor the discharge of the condenser mechanical vacuum pumps. The radiation detector is indicated and alarmed in the control room.

10.4.3 TURBINE STEAM SEALING SYSTEM

10.4.3.1 Design Bases

10.4.3.1.1 Safety Design Bases

The turbine steam sealing system serves no safety function and has no safety design bases.

10.4.3.1.2 Power Generation Design Bases

A. The turbine steam sealing system is designed to prevent air leakage into and steam leakage out of the casings of the turbine-generator and the steam generator feedwater pump turbines.

B. The turbine steam sealing system returns condensed steam to the condenser and exhausts noncondensable gases to the atmosphere.

C. The turbine steam sealing system is designed to detect the potential release of radioactive materials to the environment.

10.4.3.2 System Description

10.4.3.2.1 General Description

The turbine steam sealing system is shown in drawing 1X4DB160-3 and includes the following items and assemblies.

- Steam supply header.
- Steam exhaust header.
- Steam control panel.
- Steam packing exhauster gauge panel.

- Two steam packing exhauster blowers, motor driven.
- Associated piping, valves, and controls.

In conformance with Regulatory Guide 1.26, the quality group and associated quality standards for the turbine steam sealing system are tabulated in table 3.2.2-1.

10.4.3.2.2 System Operation

The annular space through which the turbine shaft penetrates the casing is sealed by steam supplied to shaft packings. Where the packing seals against positive pressure, the sealing steam connection acts as a leakoff. Where the packing seals against vacuum, the sealing steam either is drawn into the casing or leaks outward to a vent annulus that is maintained at a slight vacuum. The vent annulus also receives air leakage from the outside. The air-steam mixture is drawn to the steam packing exhauster.

Sealing steam is distributed to the turbine shaft seals through the steam-seal header. Steamflow to the header is controlled by the steam-seal feed valves which respond to maintain steam-seal header pressure. The DEHC provides the control signals for the steam seal feed valves. In case of high pressure, the steam packing unloading valve automatically opens to bypass excess steam directly to the main condenser.

During the startup phase of turbine-generator operation or at low turbine loads, steam is supplied to the turbine steam sealing system from the main steam piping or auxiliary steam header. During low-load operation, turbine-generator sealing steam is supplied from the main steam system through the steam-seal feed valve to maintain the necessary steamflow to the steam-seal header. As the turbine-generator load is increased, steam leakage from turbine high-pressure packings increases and enters the steam-seal header. When this leakage is sufficient to maintain steam-seal header pressure, sealing steam to all turbine seals, including the low-pressure turbine casings and the main feedwater pump turbine, is supplied entirely from these high-pressure packings. At full load, more steam leaks from the high-pressure packings than is required by vacuum packings, and excess steam is discharged directly to the main condenser. Steam leakoff from the turbine stop valves feeds into the high-pressure turbine exhaust.

The outer ends of all glands are provided with collection piping which routes the mixture of air and excess seal steam to the steam packing exhauster. The steam packing exhauster is a shell and tube heat exchanger; the steam-air mixture passes into the shell side, and service water flows through the tube side. The steam packing exhauster is maintained at a slight vacuum by a motor-operated blower, which discharges to the turbine building roof. There are two blowers mounted in parallel providing 100-percent redundancy. Condensate from the steam-air mixture drains to the main condensers, while noncondensables are exhausted to the atmosphere.

The mixture of noncondensable gases discharged from the steam packing exhauster blower is not normally radioactive; however, in the event of significant primary-to-secondary system leakage due to a steam generator tube leak, it is possible for the mixture discharged to be radioactively contaminated. The steam packing exhauster blower discharge bypasses an air filtration unit and is then headered with the condenser vacuum exhaust discharge. The headered discharge line contains a radiation monitor which diverts each discharge through its own filtration unit if radioactivity is detected in the headered discharge line. The filtered air then exhausts to the atmosphere. (See subsection 9.4.4.) A full discussion of the radiological aspects of primary-to-secondary system leakage is included in chapter 11.

Failure of the turbine steam seal system will result in no leakage of radioactivity to the atmosphere. A failure of this system would, however, result in a loss of condenser vacuum.

10.4.3.3 Safety Evaluation

The turbine steam sealing system has no safety-related function.

10.4.3.4 Tests and Inspections

Quality assurance for the design and construction of the turbine gland sealing system was provided utilizing the standard GE large steam turbine-generator quality assurance program (GEZ-4982A).

Construction phase quality assurance also included an extensive surveillance program by SCS/Bechtel inspectors. Weld procedures were reviewed by SCS stress personnel. Inspections are performed by GPC quality control personnel upon receipt of material and components and upon installation.

The system is tested, in accordance with written procedures, during the initial testing and operation program. Since the turbine steam sealing system is in constant use during normal plant operation, the satisfactory operation of the system components will be evident.

10.4.3.5 Instrumentation Applications

Automatic pressure control is provided by the digital electro-hydraulic control system (DEHC) to maintain steam seal header pressure by providing signals to the steam seal feed valves.

Local and remote indicators, as well as alarm devices, are provided for monitoring the operation of the system.

10.4.4 TURBINE BYPASS SYSTEM

The turbine bypass system has the capability to bypass main steam from the steam generators to the main condenser in a controlled manner to minimize transient effects on the reactor coolant system of startup, hot shutdown and cooldown, and the step-load reductions in generator load. The turbine bypass system is also called the steam dump system.

10.4.4.1 Design Bases

10.4.4.1.1 Safety Design Bases

The turbine bypass system serves no safety function and has no safety design basis.

10.4.4.1.2 Power Generation Design Bases

A. The turbine bypass system has the capacity to bypass 40 percent of the valves wide open (VWO) main steamflow to the main condenser.

B. The turbine bypass system is designed to bypass steam to the main condenser during plant startup and to permit a normal manual cooldown of the reactor coolant system from a hot shutdown condition to a point consistent with the initiation of residual heat removal system operation.

C. The turbine bypass system will permit a 50-percent electrical step-load reduction without reactor trip. The system under some upset conditions will also allow a turbine and reactor trip from full power without lifting the main steam relief and safety valves.

10.4.4.2 System Description

10.4.4.2.1 General Description

The turbine bypass system is shown on drawing 1X4DB160-1. The system consists of a manifold connected to the main steam lines upstream of the turbine stop valves and of lines from the manifold with regulating valves to each condenser shell. The system is designed to directly bypass 40 percent of the VWO main steamflow to the condenser.

The capacity of the system, combined with the capacity of the reactor coolant system to accept a 10-percent step-load change, provides the capability to shed 50 percent of the turbine-generator rated load without reactor trip and without the operation of relief and safety valves. A load rejection in excess of 50 percent under certain transient conditions may result in a reactor trip and may result in the operation of the main steam relief and safety valves. The operation of the main steam power-operated relief valves and spring-loaded safety valves prevents overpressurization of the main steam system.

There are 12 turbine bypass valves. Four valves discharge into each condenser shell.

The steam bypassed to the main condenser is normally not radioactive. In the event of primary-to-secondary leakage, it is possible for the bypassed steam to become radioactively contaminated. A full discussion of the radiological aspects of primary-to-secondary leakage is contained in chapters 11 and 12.

10.4.4.2.2 Component Description

The turbine bypass system contains 12 air-actuated globe valves. The valves are pilot operated, spring opposed, and fail closed upon loss of air or loss of power to the control system. Sparger piping distributes the steam within the condenser. Isolation valves permit maintenance of the bypass valves while the plant is in operation.

10.4.4.2.3 System Operation

During normal operating transients, the turbine bypass system is automatically regulated by the reactor coolant temperature control system to maintain the programmed coolant temperature. The programmed coolant temperature is derived from the high-pressure turbine first-stage pressure, which is a load reference signal. The difference between programmed reactor coolant average temperature and measured reactor coolant average temperature is used to activate the steam dump system under automatic control. The system operates in two fundamental modes.

In one mode, two groups of six valves each trip open sequentially within 3 s. This operational mode is activated during a large reactor-to-turbine power mismatch. In the other mode, four groups of three valves each modulate open sequentially. The valves will modulate with a maximum full-stroke time of 20 s over the steam pressure range of 100 to 1200 psig. A logic diagram is shown in drawing 1X6AA02-234.

When the plant is at no load and there is no turbine load reference, the system is operated in a pressure control mode. The measured main steam system pressure is compared against the pressure set by the operator in the control room. The pressure control mode is also used for plant cooldown. When the absolute pressure within the condenser shell reaches approximately 5-6 in. Hg, the valves are prevented from opening.

The turbine bypass control system can malfunction in either the open or closed mode. The effects of both these potential failure modes on the nuclear steam supply system and turbine system are addressed in chapter 15. If the bypass valves fail-open, additional heat load is placed on the condenser. If this load is great enough, the turbine is tripped on high-high condenser pressure. Ultimate overpressure protection for the condenser is provided by rupture discs. If the bypass valves fail-closed, the atmospheric relief valves permit controlled cool-down of the reactor.

10.4.4.3 Safety Evaluation

The turbine bypass system serves no safety function and has no safety design basis. There is no safety-related equipment in the vicinity of the turbine bypass system. All high-energy lines of the turbine bypass system are located in the turbine building.

The failure of a turbine bypass high-energy line will not disable the turbine speed control system. The turbine speed control system is designed in such a manner that its failure will cause a turbine trip. Additional information concerning speed control can be found in paragraphs 10.2.2.3.1.1 and 10.2.2.3.1.5.

10.4.4.4 Inspection and Testing Requirements

Before the system is placed in service, all turbine bypass valves are tested for operability. The steam lines are hydrostatically tested to confirm leaktightness. The bypass valves may be tested while the unit is in operation. All system piping and valves are accessible for inspection. The turbine bypass system falls under American National Standards Institute B31.1 rules, which do not require inservice inspection and testing.

10.4.4.5 Instrumentation Applications

The turbine bypass control system is described in section 7.7. Hand switches in the main control room are provided for selection of the system operating mode. Pressure controllers and valve position lights are also located in the main control room.

10.4.5 CIRCULATING WATER SYSTEM

10.4.5.1 Design Bases

10.4.5.1.1 Safety Design Bases

The circulating water system serves no safety function and has no safety design basis.

10.4.5.1.2 Power Generation Design Bases

The circulating water system supplies cooling water to remove heat from the main condensers and from the components of the turbine plant cooling water system under all conditions of power plant loading and design weather conditions.

10.4.5.2 System Description

10.4.5.2.1 General Description

The circulating water system consists of two 50-percent capacity circulating water pumps, one hyperbolic natural draft cooling tower, and associated piping, valves, and instrumentation. The system is shown schematically in drawings 1X4DB150 and 1X4DB150-1. Design parameters for major components are provided in table 10.4.5-1.

10.4.5.2.2 Component Description

A. Circulating Water Pumps

The two 50-percent capacity circulating water pumps are vertical, wet pit, single-stage pumps driven by solid-shaft electric motors. Each pump has a capacity of 242,300 gal/min. The pumps are mounted in an intake structure, which is connected to a cooling tower by a canal. The two-pump discharge lines connect to a common header, which connects to a three- section, six-flow-path condenser. Each pump discharge line has a motor-operated butterfly valve located between the pump discharge and the main header. This permits isolation of one pump for maintenance and allows single-pump operation.

B. Cooling Tower

The cooling tower is a hyperbolic natural draft structure employing the counterflow principal of heat transfer. The design circulating water flowrate through the tower is 509,600 gal/min. The cooling tower basin has a storage volume of 6.0×10^6 gal of water. The cooling tower is designed to withstand winds of up to 110 mph and earthquake loads of 0.15 g horizontal and 0.10 g vertical ground acceleration.

C. River Water Makeup Pumps

The circulating water system makeup is accomplished by four 22,000-gal/min river water makeup pumps. These four pumps provide the circulating water makeup to both Units 1 and 2. They are vertical, turbine-type pumps driven by electric motors and are located in the river intake structure.

D. Piping and Valves

Most of the circulating water system piping is constructed of prestressed concrete. The remainder is carbon steel and has an internal coating of corrosion preventive compound. Motor-operated butterfly valves are provided in each of the circulating water lines at their inlet to and exit from the condenser shell, to allow isolation of faulty equipment. Control valves are provided for regulation of cooling tower blowdown.

The circulating water transport system is designed to withstand the maximum operating discharge pressure of the circulating water pumps. Piping is designed for 80 psig as are the expansion joints, butterfly valves, condenser water boxes, and tube bundles.

10.4.5.2.3 System Operation

The two half-capacity circulating water pumps take suction from the circulating water intake structure and circulate the water through the tube side of the main condenser and back to the discharge network in the cooling tower. The natural draft cooling tower cools the circulating water by discharging the water over a network of baffles in the tower. The water then falls to the basin beneath the tower and, in the process, gives up some of its heat to the atmosphere. Provision is made during cold weather to direct all of the circulating waterflow to the periphery of the cooling tower. This directs the total heat load to the peripheral region. Air flowing through the peripheral spray is thus preheated which allows deicing in the central cooling tower spray region.

The flow to the cooling tower can be diverted directly to the basin, bypassing the tower internals. This is achieved by opening the motor-operated bypass valve while operating one of the two circulating pumps. The bypass is normally used only to achieve plant startup in cold weather or to maintain basic temperature above 40°F while operating at partial load during periods of cold weather.

The river water makeup pumps supply water to the circulating water system to replace water losses due to evaporation, wind drift, and blowdown. The makeup water is supplied to the cooling tower basin. Normally, only one or two of the makeup pumps are operating, depending upon the makeup demand.

During normal plant operation, biocides are added to the circulating water to control biological growth as needed. Chemical treatment is added to the flume as required to maintain a Ryznar Index to prevent scaling and reduce corrosion. A condenser tube cleaning system is installed to clean the circulating water side of the main condenser.

Blowdown from the circulating water system is taken from the cooling tower basin and is dechlorinated, as required, and discharged. Water being discharged into the river meets appropriate regulatory requirements.

The circulating water system is normally used to supply cooling water to the main condenser in order to condense the steam exhausted from the main turbine. If the circulating water pumps, the cooling tower, or the circulating water piping malfunctions such that the flowrate of the circulating water falls below the minimum allowable value, the main condenser will no longer be able to adequately perform its functions. Cooldown of the reactor in such an event would be accomplished by using the power-operated atmospheric steam relief valves rather than the turbine bypass system.

Passage of secondary system condensate from the main condenser into the circulating water system through a condenser tube leak is not possible during power generation operation, since the circulating water system operates at a greater pressure than the condenser.

Small leaks around valves and fittings would drain into the turbine building drain sump via the floor drains. The condenser pit sump is provided with sump pumps and with high-level alarms. Large leaks due to pipe failures would be indicated in the control room by a gradual loss of vacuum in the condenser shell. The effects of turbine building flooding are discussed in appendix 3F.

In the event of collapse of the cooling tower, the potential for damage to other plant structures is slight. The structure would tend to collapse inwardly. To prevent damage, the structure is located a sufficient distance from any equipment or structure important to reactor safety.

10.4.5.3 Safety Evaluation

Because the circulating water system has no safety design basis, no safety evaluation is provided.

10.4.5.4 Tests and Inspections

All active components of the circulating water system are accessible for inspection during plant power generation. The circulating water pumps and river water makeup pumps are tested in accordance with standards of the Hydraulic Institute. Performance, hydrostatic, and leakage tests associated with preinstallation and preoperational testing are performed on the circulating water system in accordance with the standards of the Hydraulic Institute and the American Water Works Association Code 504-70. The performance along with structural and leaktight integrity of all system components are demonstrated by continuous operation.

10.4.5.5 Instrumentation Applications

Indicating lights are provided in the control room to indicate open and closed positions of motor-operated butterfly valves in the circulating water piping. The motor-operated valve at each pump discharge is interlocked with the pump such that the pump will be shut down if the discharge valve fails to reach the full-open position shortly after starting the pump. The motor-operated valves are provided with position switches required for indicating lights and interlocking with the pumps.

The process sampling system discussed in subsection 9.3.2 periodically tests the circulating water quality to ensure that no harmful effects will result to the system piping and valves due to improper water chemistry.

Local pressure indicators are provided on the circulating water pump discharge lines. A local pressure indicator and a pressure transmitter are provided on one inlet branch to the condenser in the turbine building. The pressure transmitter also provides indication in the control room and provides a signal to the plant computer.

On two inlet branches to the condenser, local and remote indicators are provided for temperature. A signal for inlet temperature is also provided to the plant computer.

On each of the six outlet branches from the condenser, temperature instrumentation is provided for local indication and signal to the plant computer.

In each of the six condenser flow paths, differential pressure transmitters are provided to measure the inlet/outlet differential pressure and to feed this information to the plant computer.

Level instrumentation is provided in the circulating water pump intake structure to transmit level signals to the cooling tower makeup valves. Level instrumentation is also provided to annunciate a low water level in the pump structure and a high water level in the cooling tower basin. These annunciators are located in the main control room. Hand switches for the cooling tower makeup valves are also located in the main control room so that the valves may be aligned manually.

In order to maintain totally dissolved solids below a pre-established level, cooling tower blowdown is controlled as a function of plant load (condensate flow) and circulating water conductivity. Cooling tower blowdown also has a manual mode of operation.

10.4.6 CONDENSATE CLEANUP SYSTEM

The condensate cleanup function is performed by the condensate filter demineralizer system. The system maintains the required purity of feedwater for the steam generators by filtration to remove corrosion products and/or ion exchange to remove condenser leakage impurities.

10.4.6.1 Design Bases

10.4.6.1.1 Safety Design Bases

The condensate filter demineralizer system serves no safety function and has no safety design basis.

10.4.6.1.2 Power Generation Design Bases

The condensate filter demineralizer system has the following power generation design bases.

A. Removes dissolved and suspended solids from the condensate prior to startups.

B. Removes impurities entering the secondary cycle from condenser leaks that would otherwise concentrate in the steam generators.

C. Removes corrosion products from the condensate and any drains returned to the hotwell, so as to limit accumulation of those products which are difficult to remove from the steam generator by blowdown.

D. Limits the entry of dissolved solids into the feedwater system in the event of large condenser leaks, such as a tube break, to permit a reasonable amount of time for plant shutdown.

E. Removes impurities from the condensate, which may occur in primary-to-secondary leaks in the steam generators until the leaks can be repaired.

F. Includes a backwash recovery subsystem, which is capable of recovering the backwash water to a quality suitable for return to the condensate system. The spent resins are separated as a pumpable slurry ready for disposal.

G. Includes a spent resin dewatering system. During normal operations, resins are dewatered. If a primary to secondary leak is suspected or has occurred, resins are sampled for radioactivity and either transferred to a high integrity container for radwaste disposal or dewatered. Dewatered resins are sampled for radioactivity and then either sent to low-level radwaste storage or disposed of as nonradioactive waste.

10.4.6.2 System Description

10.4.6.2.1 General Description

The condensate cleanup system processes the full flow of condensate and has five condensate filter demineralizer vessels with 25-percent capacity each (one on standby), five holding pumps, piping, instrumentation, and controls. A backwash system cleans the filters. The filters are either operated without resin or are precoated with resins, and a backwash recovery system permits separation of spent demineralizer resins.

The condensate cleanup system filter demineralizer vessels are located at the discharge of the condensate pumps in the turbine building between el 220 and 245 ft. System arrangement is shown in drawings 1X4DB185-1, 1X4DB185-2, 1X4DB185-3 and 1X4DB185-5. Design data for major components are listed in table 10.4.6-1. Codes and standards applicable to the condensate cleanup system are listed in table 3.2.2-1. The system is designed and constructed in accordance with quality Group D specifications. The filter demineralizer system, including the backwash handling system, is designed to sustain power generating operations when condenser leaks are 5 gal/min or less, or primary-to-secondary leaks are 0.1 gal/min or less with a specified failed fuel of 0.25 percent.

Provisions are made in the building design for shielding against potential radioactivity, and space for 18-in. shielding walls around each vessel is allowed. Access is provided to the equipment for inspection and repair.

The spent powdered demineralizer resins, after backwashing from the filter demineralizer vessels, are normally dewatered, sampled for radioactivity, and then either sent to low-level radwaste disposal or disposed of as nonradioactive waste. If a primary to secondary leak is suspected or has occurred, resins are sampled for radioactivity, and either transferred to a high integrity container for radwaste disposal or dewatered.

10.4.6.2.2 Component Description

10.4.6.2.2.1 Condensate Filter Demineralizer. The condensate filter demineralizers are the precoatable type with the demineralizer vessels designed to operate at condensate pump discharge pressures and temperatures. Design specific flowrate does not exceed 4 gal/min/ft^2 of coated element surface. The demineralizer units are capable of operating with either mixed powdered resins or other approved precoat media as a precoat. In each filter vessel, 420 polypropylene-wound or nylon-wound elements, 70 in. long, are evenly coated with a total of 300 lb of mixed resins. The active surface area is 1281 ft^2 and is 1/4 to 3/8 in. thick.

The system is installed with a full-flow bypass which ensures full condensate flow and limits pressure drop across the cleanup system. The bypass can be used when water chemistry can be maintained below normal design limits by the steam generator blowdown system alone and/or during emergency conditions to maintain main feedwater pump suction pressure requirements.

For each of the condensate filter demineralizers, a holding pump is provided to recycle a minimum flow as required to retain the precoat material on the vessel element if the condensate flow should become too low or the vessel is in standby. A flow control system is provided to equalize the flows through the four vessels on line. Each vessel is also equipped with a resin trap to catch resin in the vessel effluent.

After the vessel is exhausted, the spent resin is purged out of the vessel sequentially by backwash water and service air. The backwash pump supplies the water required for backwashing from the condensate storage tank through the condensate makeup line. The backwashing air requirement is taken from the station oil free air supply system.

After all the spent resin is backwashed from the vessel, new resin is coated on the clean vessel elements. New resin is manually charged to the precoat tank, where it will flocculate and slurry with demineralized water by action of an agitator. The slurry is recirculated by a precoat pump through the filter demineralizer unit until the filter elements are completely coated. The freshly coated filter may then be returned to the standby mode.

10.4.6.2.2.2 Backwash Recovery Subsystem. The spent powdered resin or other approved precoat media is flushed to the backwash recovery tank by backwash water.

After an exhausted vessel is taken out of service and before the backwash operation of the spent resin is performed, the backwash water from the previous backwash operation stored in the backwash recovery tank can be pumped by the backwash recovery pump and processed through the exhausted vessel until the backwash water quality is good enough for recycling to the turbine cycle. If necessary, the exhausted vessel is overlaid with fresh resin for cleaning up the backwash water. An overlay tank with an agitator and an overlay pump is provided to overlay the spent vessel.

10.4.6.2.2.3 Spent Resin Dewatering Subsystem. The spent resin dewatering subsystem receives the resin slurry resulting from backwashing the spent powdered resins from the demineralizer vessels. The slurry is stored in spent resin holding tanks located within the turbine building, where it is allowed to clarify. Excess water is then removed by pumping the clarified water through cartridge-type filters to the turbine building drain system. This concentrates the resin slurry. During normal operations, resins are separated and dewatered by processing the concentrated slurry through belt-type pressure filters. The spent resin dewatering subsystem can transfer resin from the spent resin storage tanks to a high integrity container for radwaste disposal. Dewatered resins are sampled for radioactivity and then either sent to low-level radwaste storage or disposed of in the industrial waste landfill. The spent resin dewatering system is common to Units 1 and 2.

10.4.6.2.3 System Operation

The demineralizers are capable of operation in continuous or intermittent service. The system is operated to maintain the water quality requirements of the nuclear steam supply system supplier. In addition, the ion concentrations, particularly sodium and chloride, are maintained within the effluent requirements given in paragraph 10.4.6.2.4.4.

Service runs are terminated when the following occur.

A. The pressure drop from inlet to outlet reaches a preselected value between 25 and 35 psi due to buildup of suspended solids.

B. Conductivity measurements indicate that the resin bed has reached ion exchange exhaustion.

C. Sodium, silica, or other ion leakage is judged unacceptable.

The valving and controls are such that a ready standby vessel can be placed in service, or any or all of the operating vessels can be placed out of service from the local control panel.

The following functions are initiated by switch or push button at the local control panel:

A. Placing standby vessel online and exhausted vessel on hold.

B. Initiation of automatic backwash.

C. Initiation of automatic precoat operation.

D. Bypassing vessel system.

The backwash water from the previous backwashing operation can be polished through the exhausted vessel and subsequently sent to the condensate storage tank after the water quality is tested and proved suitable.

The spent resin is backwashed using condensate storage tank water and air. During normal operation, the backwash water and spent resin slurry are pumped to the spent resin dewatering subsystem for processing. After the elements are cleaned, fresh resin is coated on the elements.

Fresh resins are manually charged to the resin precoat tank, where they are slurried with demineralized water by an agitator. Flocculation of the resins occurs as the cation and anion forms agglomerate. The floc size is controlled to 4 to 8 mm. The slurry is recirculated by a slurry pump through the vessel until the filter elements are completely coated. (Leaking elements may be detected by taking filter samples from grab sample connections on each precoat system downstream of the elements.)

This vessel is then ready for service. These backwash recovery and backwash/precoat cycles are manually initiated but are carried out according to an automatic program sequence.

10.4.6.2.4 Water Quality

10.4.6.2.4.1 Influent Water Quality. Table 10.4.6-2 lists the concentrations of corrosion products and other impurities in the influent condensate used as a design basis for two operating conditions, startup or design maximum and expected operation. The concentrations listed for the expected operation are those associated with a condenser leak of 1 gal/min. The concentrations listed for design maximum or startup have been calculated from a condenser leak of 5 gal/min, and the maximum radioactivity level has been set by a steam generator tube leak of 0.1 gal/min.

10.4.6.2.4.2 Primary Coolant. The primary coolant circulating through the steam generator haas the expected analysis shown in table 10.4.6-3.

10.4.6.2.4.3 Circulating Water. The circulating water has the expected analysis after six cycles as shown in table 10.4.6-4.

10.4.6.2.4.4 Effluent Requirements. The condensate filter demineralizer is operated to produce the effluent required to meet the secondary side water chemistry specifications during power operation provided in the EPRI PWR Secondary Water Chemistry Guidelines.

10.4.6.3 Safety Evaluation

There is no safety design basis for the condensate cleanup system and no safety evaluation is provided.

10.4.6.4 Tests and Inspections

Preoperational testing of the condensate cleanup system ensures proper functioning of the equipment and instrumentation. The system is checked functionally during power generation operation.

10.4.6.5 Instrumentation Applications

In-line sensors monitor equipment performance during demineralizer operation or their restoration cycle. Local and control room alarms annunciate trouble in components of the system. Systematic analysis of local samples monitors the accuracy of the automatic equipment. At each filter, demineralizer flow and differential pressure are continually monitored. Instruments of the condensate system continuously monitor the influent and effluent streams of the filter demineralizer system. Ionic concentrations are monitored using sampling points and methods identified in table 9.3-2-3 (grab sample points and process instrumentation).

10.4.7 CONDENSATE AND FEEDWATER SYSTEM

The condensate and feedwater system provides for the condensing of high-pressure and low-pressure turbine extraction and exhaust steam and main feedwater pump turbine exhaust steam, collects the condensate in the condenser hotwell, and maintains steam generator water level by supplying preheated feedwater through all power operation modes of the plant.

10.4.7.1 Design Bases

10.4.7.1.1 Safety Design Bases

The safety-related portions of the condensate and feedwater system are protected from wind and tornado effects, as discussed in section 3.3; flood protection is discussed in section 3.4; missile protection is discussed in section 3.5; protection against dynamic effects associated with the postulated rupture of piping is discussed in section 3.6; and environmental design is discussed in section 3.11.

The portion of the feedwater system from the steam generator main and bypass feedwater inlets to the first seismic restraint upstream of the containment feedwater isolation valves are designed in accordance with the requirements of Section III of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code for Class 2 components and are designed to Seismic Category 1 requirements. This same portion of the system is also required to function following a design basis accident (DBA) and to achieve and maintain the plant in a safe shutdown condition. The failure of the condensate and feedwater system upstream of the main feedwater isolation valves (MFIVs) and the bypass feedwater isolation valves (BFIVs) would not result in either exceeding containment design pressure or a departure from nucleate boiling ratio (DNBR) below the design basis limit. The system provides isolation valving on main and bypass feedwater lines into containment. The isolation valves close within 5 s after receipt of an isolation signal. In addition, the feedwater pumps are tripped simultaneously with the signal to close the feedwater control valves.

A. The safety-related portion of the condensate and feedwater system is designed to remain functional after a safe shutdown earthquake (SSE) or to perform its intended function following postulated hazards of fire, internal missiles, or pipe break.

B. The safety-related portion of the system is protected from the effects of natural phenomena such as earth- quakes, tornadoes, hurricanes, floods, and external missiles.

C. Safety functions can be performed, assuming a single active component failure coincident with the loss of offsite power.

D. The portion of the feedwater system to be constructed in accordance with ASME III, Class 2 requirements is provided with access to welds and removable insulation for inservice inspection, in accordance with ASME XI. The condensate and feedwater system is designed so that the active components are capable of limited testing during plant operation.

E. The condensate and feedwater system is designed and fabricated to codes consistent with the quality group classification assigned by Regulatory Guide 1.26 and the seismic category assigned by Regulatory Guide 1.29. The power supply and control functions are in accordance with Regulatory Guide 1.32.

F. For a main feedwater or main steam line break (MSLB) inside the containment, the condensate and feedwater system is designed to limit high-energy fluid to the broken loop. The feedwater bypass lines provide a path for the addition of auxiliary feedwater to the three intact loops.

G. For a main feedwater line break upstream of the MFIV (outside of the containment), the condensate and feedwater system is designed to prevent the blowdown of any steam generator. The feedwater bypass lines provide a path for the addition of auxiliary feedwater.

10.4.7.1.2 Power Generation Design Bases

A. The condensate and feedwater system is designed to provide a continuous feedwater supply to the four steam generators at the required pressures and temperatures for steady-state and anticipated transient conditions.

B. The system is designed to maintain feedwater flow following a 50-percent step reduction in electrical load.

C. The system is designed to continue functioning in spite of a loss of one feedwater pump or one heater drain pump.

D. The system is designed to protect the steam generator main feedwater inlet nozzle and inlet feedwater piping during low-load and hot standby conditions.

10.4.7.2 <u>System Description</u>

10.4.7.2.1 General Description

The condensate and feedwater system is shown schematically in drawings 1X4DB168-1, 1X4DB168-2, and 1X4DB168-3. The condensate and feedwater system supplies the steam generators with heated feedwater in a closed steam cycle using regenerative feedwater heating.

The main portion of the feedwater flow is deaerated condensate pumped from the main condenser hotwells by the condensate pumps. The main condenser hotwells receive makeup from the condensate tank. (Refer to subsection 9.2.6 for a discussion of the condensate storage system.) This stream passes, in sequence, through the condensate filter demineralizer system (described in subsection 10.4.6); the three trains of low-pressure heaters, each train consisting of a number 1, 2, and 3 low-pressure heater; two trains of low-pressure heaters, each train consisting of a number 4 and 5 low-pressure heater; the parallel steam generator feedwater pumps; the two trains of high-pressure heaters, each train consisting of a number 6 high-pressure heater; regulating and isolation valves; and on into the four steam generators. The balance of the feedwater flow is provided by the drains from the moisture separators, the reheaters, and number 6, 5, and 4 heaters and is collected into a drain tank and pumped into the feedwater pump suction stream by the heater drain pumps.

To allow for feedwater and condensate system flush, cleanup, and startup recirculation, a steam generator flush line is installed upstream of the main feedwater isolation valves. This flush system is designed for approximately 50 percent of the maximum condensate-rated flow and is used in flushing operations. The flush lines may be discharged to the condensers or to the waste water retention basins. Steam may be provided to the number 5 low-pressure heater from the auxiliary steam in order to preheat the feedwater to about 200°F during the initial cleanup and startup recirculation operations. This preheating action, along with added chemicals, prevents formation of iron oxides in the condensate system.

The condensate filter demineralizer system consists of five precoat-type vessels, as described in subsection 10.4.6, that may be in service or completely bypassed.

Each of the four main 16-in. feedwater lines to the four steam generators contains a feedwater flow element, a main feedwater regulating valve, a main feedwater bypass regulating valve, a power-operated MFIV, a tilting disc check valve, and a chemical injection connection.

A 6-in. bypass feedwater line tees off each main 16-in. feedwater line just upstream of the 16-in. MFIV and contains a BFIV, one tilting disc check valve located outside containment, an auxiliary feedwater connection, followed by one tilting disc check valve inside containment before it connects to the steam generator 6-in. auxiliary feedwater (bypass feedwater) nozzle.

The condensate and feedwater chemical injection system, as shown in drawing 1X4DB157, is provided to inject hydrazine and alternate amine into the condensate pump discharge downstream of the condensate demineralizers and additional hydrazine and alternate amine into the four 6-in. bypass feedwater lines connecting with the four steam generators. Injection points are shown in drawing 1X4DB157.

During normal power operation, the continuous addition of hydrazine to the condensate system downstream of the condensate demineralizers is under automatic control, with manual control optional. As discussed in subsection 10.3.5, the addition of alternate amine and hydrazine establishes the design pH according to the condensate and feedwater system chemistry requirements and establishes a constant initial hydrazine residual in the feedwater system, so that oxygen inleakage can be scavenged.

During hot standby conditions, the wet layup chemical feed pumps are available to inject a mixture of hydrazine and alternate amine from a batch chemical feed tank into the steam generators downstream of the connection to the 6-in. bypass feedwater line when the condensate and/or steam generator feedwater pumps are not operating.

Within the condensate and feedwater system component failures which affect the final feedwater temperature or flow have a direct effect on the reactor coolant system. These are listed in table 10.4.7-1. Occurrences which produce an increase in feedwater flow or decrease in feedwater temperature result in increased heat removal from the reactor coolant system (RCS), which is compensated for by control system action, as described in section 7.7. Events which produce the opposite effect (i.e., decreased feedwater flow or increased feedwater temperature) result in reduced heat transfer in the steam generators. Normally, automatic control system action is available to adjust feedwater flow and reactor power, which prevents excess energy accumulation in the RCS, and the increasing reactor coolant temperature provides a negative reactivity feedback which tends to reduce reactor power. In the absence of normal control action, either the high-outlet temperature or the high-pressure trips of the reactor by the reactor protection system are available to ensure reactor safety. Loss of all feedwater, the most severe transient of this type, is examined in chapter 15.

Refer to section 5.4 for a discussion of steam generator design features to preclude fluid flow instabilities, such as water hammer. The main 16-in. and bypass 6-in. feedwater connections on each of the steam generators are the highest point of each feedwater line downstream of the MFIV and the BFIV. An elbow, with a short transition piece, is connected directly to the steam generator main 16-in. and bypass 6-in. feedwater nozzles, which tend to minimize the portion of feedwater piping that can drain into the steam generator and become filled with steam. The feedwater lines contain no high-point pockets that could trap steam and lead to water hammer. The horizontal pipe length from the main and bypass feedwater nozzles of each steam generator is minimized. This feedwater system configuration coupled with the steam generator design minimizes the potential for steam backleakage and bubble collapse water hammer in the main and auxiliary feedwater lines. To complement the system and steam generator design, Westinghouse recommended feedwater system maintenance and startup procedural guidelines to further preclude feedwater system water hammer conditions in the main and bypass feedwater lines. The routing of the main and bypass feedwater lines is shown in drawings 1K5-1305-062-01 and 1K5-1305-058-01.

10.4.7.2.2 Component Description

Codes and standards applicable to the condensate and feedwater system are listed in table 3.2.2-1. The condensate and feedwater system is designed in accordance with the requirements of Section III of the ASME Boiler and Pressure Vessel Code for Class 2 components and Seismic Category 1 requirements from the steam generator out to the restraint upstream of the MFIVs and BFIVs. The remaining piping of the condensate and feedwater system meets American National Standards Institute (ANSI) B31.1 requirements. Refer to tables 10.1-1 and 10.4.7-2 for design data. Safety-related feedwater piping materials are discussed in subsection 10.3.6.

10.4.7.2.2.1 Feedwater Piping. Approximately 92 percent of the feedwater is supplied to the four main steam generators by four main 16-in. carbon steel lines during normal operation. Each of the lines is anchored at the containment wall and has sufficient flexibility to provide for relative movement of the steam generators resulting from thermal expansion. The main feedwater line and associated branch lines between the containment penetration and the restraint upstream of the MFIV are designed to meet the criteria, as described in Nuclear Regulatory Commission (NRC) Branch Technical Position (BTP) MEB 3-1. (Refer to section 3.6.)

The remaining 8 percent of the feedwater is supplied to the four steam generators by four bypass 6-in. carbon steel lines. Each of the lines is anchored at the containment wall and has sufficient flexibility to provide for relative movement of the steam generators resulting from thermal expansion. The bypass feedwater line and associated branch lines are designed and constructed in accordance with Quality Group B and Seismic Category 1 requirements.

The 6-in. bypass feedwater line is provided to a 6-in. auxiliary feedwater steam generator nozzle to add feedwater during low-load or hot standby conditions when the feedwater flow and temperatures are low. During these periods, no flow is added through the main 16-in. feedwater line and nozzle by closing the MFIVs in order to reduce the potential of main feedline and nozzle cracking.

The feedwater piping system is analyzed to eliminate or minimize the potential for water hammer and subsequent detrimental effects on the following:

- Steam generators with top feed ring design (BTP ASB 10-2).
- Main and bypass feedwater check valves due to line breaks (BTP MEB 3-1).
- Spurious isolation or feedwater regulating valve trips.
- Pump trips.
- Condensate filter-demineralizer flow balancing control valve trip.
- Local feedwater piping, anchors, supports, and snubbers as applicable.

10.4.7.2.2.2 Feedwater Isolation Valves. One MFIV is installed in each of the four main feedwater lines outside the containment and downstream of the feedwater regulating valve. The MFIVs are installed to prevent uncontrolled blowdown from more than one steam generator in the event of a feedwater pipe rupture. The main feedwater check valve provides backup isolation. The MFIVs isolate the nonsafety-related portions from the safety-related portions of the system. In the event of a secondary side pipe rupture inside the containment, the MFIVs limit the quantity of high-energy fluid that enters the containment through the broken loop and provides a pressure boundary for the controlled addition of auxiliary feedwater to the three intact loops.

The MFIVs are also provided to prevent feedwater from entering the main steam generator inlet nozzle to cause line cracking during low-load and/or hot standby conditions when the feedwater flow and temperatures are low. During these periods feedwater enters the steam generator through the 6-in. bypass line and 6-in. auxiliary feedwater nozzle. The valves are bidirectional, double disc, parallel slide gate valves. Stored energy for closing is supplied by accumulators which contain a fixed mass of high-pressure nitrogen and a variable mass of high-pressure hydraulic fluid. For emergency closure, a solenoid is deenergized, which causes the high-pressure hydraulic fluid to be admitted to the top of the valve stem driving piston and also causes the fluid stored below the piston to be dumped to the fluid reservoir. Two separate pneumatic/hydraulic power trains are provided for each MFIV. Electrical solenoids are energized from separate Class 1E sources.

One BFIV is installed in each of the four bypass feedwater lines outside containment. The BFIVs are installed to prevent uncontrolled blowdown from more than one steam generator in the event of a feedwater line rupture. Bypass feedwater check valves provide backup isolation. The BFIVs isolate the nonsafety-related portions from the safety-related portions of the system. The BFIVs provide the same function as do the MFIVs. The valve body configuration and type of BFIVs are similar to the MFIVs. Air-operated piston actuators are provided to close the valves. Electrical solenoids are energized from separate Class 1E sources.

10.4.7.2.2.3 Main Feedwater Regulating Valves and Regulating Bypass Valves. The 16-in. main feedwater regulating valves are air-operated angle valves which control feedwater flow between approximately 20 percent and full power. The bypass control valves are air-operated globe valves, which are used during startup to approximately 40-percent power.

The feedwater regulating and bypass valves automatically maintain the water level in the steam generators during all phases of operation, in accordance with the steam generator manufacturer's prescribed normal water level. Positioning of the main feedwater regulating valve during normal operation is the function of a three-element feedwater level control system. The bypass feedwater regulating valve is also positioned during normal operation by the three-element feedwater level control system. The three-element control system maintains feedwater flow equal to the steamflow, and steam generator water level is used as an input to trim feedwater flow and maintain programmed water level. During startup operations from no-load to about 40-percent power, a 4-in. bypass valve is provided to automatically maintain the water level in each steam generator.

In the event of a secondary side pipe rupture inside the containment, the main feedwater regulating valve (and associated bypass valve) provides a secondary backup to the MFIV to limit the quantity of high-energy fluid that enters the containment through the broken loop. For emergency closure of the main feedwater regulating valves, both of the two separate solenoids, when deenergized, will result in valve closure in 5 s or less. For emergency closure of the regulating bypass valves, either of two separate solenoids, when deenergized, will result in valve closure in 5 s or less. Electrical solenoids are energized from separate Class 1E sources.

10.4.7.2.2.4 Feedwater Check Valves. The main feedwater tilting disc check valves are located immediately downstream of the MFIVs. The check valve provides a diverse backup to the MFIV for preventing blowdown of the steam generators during a main feedline break outside the containment.

Two tilting disc check valves are located downstream of the BFIVs. To prevent a possible backflow or spillage of auxiliary feedwater, the first check valve is located upstream of the auxiliary feedwater line connection to the bypass feedline. The check valve in the bypass line inside containment reduce the possibility of the steam generator blowing down through the 6-in. nozzles simultaneously following an auxiliary and bypass feedline rupture outside containment.

10.4.7.2.2.5 Chemical Addition Line Check Valves and Isolation Valves. The check valves are located downstream of the chemical feedline isolation valves in the chemical addition lines. The check valves provide a secondary backup to the isolation valves to ensure the pressure boundary. The normally closed isolation valves are air-operated valves which fail-closed.

10.4.7.2.2.6 Condensate Pumps. The three condensate pumps are motor-driven and operate in parallel. Valving is provided to allow individual pumps to be removed from service. Pump capacity is sufficient to meet full-power requirements with two of the three pumps in operation.

10.4.7.2.2.7 Low-Pressure Feedwater Heaters. Parallel strings of closed, low-pressure feedwater heaters number 1, 2, and 3 are located in each of three condenser necks. The number 1, 2, and 3 heaters have integral drain coolers, and their drains cascade to the next lower stage feedwater heater. The drains from the number 1 heaters are dumped to their respective condenser. Feedwater leaving the three trains of number 3 heaters is headered and enters the two trains of low-pressure feedwater heaters number 4 and 5. Heater number 4 has no integral drain cooler and drains directly into the heater drain tank. Heater number 5 has an integral drain cooler, and its drains cascade into the number 4 heater shell. The discharges from the two trains of number 5 heaters enter the suction of the two steam generator feedwater pumps.

10.4.7.2.2.8 High-Pressure Feedwater Heaters. The steam generator feedwater pumps discharge into two trains of number 6 high-pressure feedwater heaters which have integral drain coolers. The number 6 heaters drain directly into the shell side of low-pressure heaters number 5.

Isolation valves and bypasses are provided which allow each string of feedwater heaters to be removed from service.

Provisions are made in all heater drain lines, except number 4, which drains via the heater drain tank to allow direct discharge to the condenser in the event the normal drain path is blocked or flooding occurs in the heater.

The low-pressure and high-pressure feedwater heater shells are carbon steel, and the tubes are stainless steel.

10.4.7.2.2.9 Heater Drain Tanks. Two heater drain tanks, each serving one train of heaters number 4, 5, and 6, receive the total drains from these heaters and provide a reservoir capacity for the two heater drain pumps. Each heater drain tank is installed below the number 4 heater and receives the total drain flow by gravity flow. The drain level is maintained within the tank by a level controller in conjunction with a heater drain pump.

Each heater drain tank is provided with an alternate drain line to the main condenser for automatic dumping upon high-liquid level. The alternate drain line is also used during startup and shutdown when it is desirable to bypass the drain flow to the condenser for feedwater quality purposes.

10.4.7.2.2.10 Heater Drain Pumps. Each motor-driven heater drain pump takes suction from its respective heater drain tank and discharges it into the condensate system between the no. 4 and no. 5 feedwater heaters.

10.4.7.2.2.11 Steam Generator Feedwater Pumps. The steam generator feedwater pumps operate in parallel and discharge to the number 6 high-pressure feedwater heaters. The pumps take suction from the number 5 low-pressure feedwater heaters and discharge through the high-pressure feedwater heaters. Each pump is turbine driven. The turbine drivers are regulated by a master speed controller for both pumps and one slave speed controller for each pump. Steam for the turbines is supplied from the main steam header at low loads and from the moisture separator/reheater steam outlet during normal operation.

Isolation valves are provided which allow each steam generator feedwater pump to be individually removed from service, while continuing operations at reduced capacity with the parallel pump.

10.4.7.2.2.12 Pump Recirculation Systems. Minimum flow control systems are provided to allow all pumps in the main condensate and feedwater trains to pump at the manufacturer's recommended minimum flowrate to prevent pump damage.

10.4.7.2.3 System Operation

10.4.7.2.3.1 Normal Operation. For normal operating conditions between approximately 3- to 100-percent load, system operation is automatic. Automatic level control systems control the water levels in all feedwater heaters, the heater drain tank, and the condenser hotwell. Feedwater heater water levels are controlled by modulating flow control valves. Control valves in the discharges of the heater drain pumps control heater drain pump flows in relation to the level in the heater drain tank. Level control valves in the makeup line to the condenser from the condensate storage tank and in the return line to the condensate storage tank control the level in the condenser hotwell.

At very low power levels, feedwater is available from the auxiliary feedwater system. (Refer to subsection 10.4.9.) Once sufficient steam pressure has been established, a steam generator feedwater pump turbine is started, and from this low power level to approximately 20-percent power, feedwater flow is controlled by the feedwater bypass regulating valves. During this phase of operation, the variable-speed steam generator feedwater pump turbine may be placed on manual or automatic control at a low pump speed. Open and shut valve position indicators are provided in the control room for the main and bypass feedwater regulating valves. Feedwater flow control is transitioned to include both the bypass regulating valves and the main feedwater regulating valves. This operation may be done entirely in automatic control or with the bypass and main regulating valves in manual control. The plant load may be increased and the plant placed on completely automatic control with both the main and bypass feedwater valves in operation.

Ten-percent step load and 5-percent/min ramp changes are accommodated without major effect to the condensate and feedwater system. The system is capable of accepting a 50-percent step-load rejection without tripping of reactor or turbine systems. Under this transient, heater drain pump flow may be lost and the feedwater heater drain flows may be subsequently dumped to the condenser via the heater drain tank. The condensate pumps furnish full feedwater flow until heater drain pump flow is restored.

There are basically two modes of operation involved in supplying feedwater into the steam generators: (1) zero and low powers and (2) moderate and high powers. For the first, the required feedwater is supplied through the feedwater bypass line; i.e., BFIV is open and MFIV is closed. At moderate and high powers, the steam generator is supplied simultaneously through the bypass and main lines; i.e., both BFIV and MFIV are open.

The MFIV should not be opened until the resulting flowrate through the main nozzle exceeds that where stratification and striping can occur. This is 500 gal/min per steam generator, which corresponds to a 6-percent power level.

With both BFIV and MFIV open, approximately 8 percent of the total flow will be through the bypass line and 92 percent through the main line. The simultaneous flow in the bypass line serves two purposes. It serves as tempering flow, maintaining the 6-in. nozzle at a reduced temperature, and it ensures that steam generator water or steam does not reverse flow in the bypass line into the auxiliary feedwater system.

During the plant loading operation, the MFIV will be opened when the power level is 12 to 20 percent, so that the required 500 gal/min per steam generator is significantly exceeded. To achieve the 1- to 2-percent tempering flow recommendation would require opening the MFIV at about 25-percent power. At this power level, sufficient differential pressure across the bypass line is available to provide the recommended flow.

During the operations of heatup, cooldown, and hot standby, the necessary steam generator makeup will be provided by the auxiliary feedwater system through the auxiliary feedwater nozzle.

10.4.7.2.3.2 Emergency Operation. In the event that the plant must be shut down and offsite power is lost or a DBA occurs which results in a feedwater isolation signal, the MFIV, BFIV, and the main feedwater control and bypass valves are automatically closed. Coordinated operation of the auxiliary feedwater system (refer to subsection 10.4.9) and the main steam supply system (refer to section 10.3) is employed to remove the primary loop sensible heat and reactor decay heat.

10.4.7.3 Safety Evaluation

A. The safety-related portions of the condensate and feedwater system are located in the control and auxiliary buildings. These buildings are designed to withstand the effects of earthquakes, tornadoes, hurricanes, floods, external missiles, and other appropriate natural phenomena. Sections 3.3, 3.4, 3.5, 3.7, and 3.8 provide the bases for the adequacy of the structural design of these buildings.

B. The safety-related portions of the condensate and feedwater system are designed to remain functional after an SSE. Subsection 3.7.2 and section 3.9 provide the design loading conditions that were considered. Sections 3.5 and 3.6 and subsection 9.5.1 provide the hazards analyses to ensure that a safe shutdown, as outlined in section 7.4, can be achieved and maintained.

C. The condensate and feedwater system safety functions are accomplished by redundant means. A single active component failure of the safety-related portion of the system will not compromise the safety function of the system. As indicated by table 10.4.7-1, no single failure of nonessential portions of the system will compromise the system's safety functions. All vital power can be supplied from either onsite or offsite power systems, as described in chapter 8.

D. Preoperational testing of the condensate and feedwater system is performed as described in chapter 14. Periodic inservice functional testing is done in accordance with paragraph 10.4.7.4.

Section 6.6 provides the ASME Boiler and Pressure Vessel Code Section XI requirements that are appropriate for the condensate and feedwater system.

E. Section 3.2 delineates the quality group classification and seismic category applicable to the safety-related portion of this system and supporting systems. Table 10.4.7-2 shows that the components meet the design and fabrication codes given in section 3.2. All the power supplies and controls necessary for the safety-related functions of the condensate and feedwater system are Class 1E, as described in chapters 7 and 8.

F. For a main feedwater line break inside the containment or a MSLB, the MFIVs, BFIVs, and the main feedwater regulating valves (and associated bypass valves) located in the control and auxiliary buildings (main feedwater isolation valve areas) are automatically closed upon receipt of a feedwater isolation signal, which occurs on a safety injection initiation, steam generator high-high water level, or a reactor trip coincident with a low T_{avg} condition. For each intact loop, the main and bypass feedwater check valves, MFIV, BFIV, and associated redundant isolation of the chemical addition line will close, forming a pressure boundary to permit auxiliary feedwater addition. The auxiliary feedwater system is described in subsection 10.4.9.

G. For a main feedwater line break upstream of the MFIV, the MFIVs and BFIVs are supplied with redundant power supplies and power trains to ensure their closure to isolate safety- and nonsafety-related portions of the system. Branch lines downtream of the MFIVs and BFIVs contain normally-closed, fail closed with power locked out, or locked closed manual valves. These valves separate the safety-related and nonsafety-related portions of the system.

Releases of radioactivity from the condensate and feedwater system resulting from the main feedwater line break are minimal because of the negligible amount of radioactivity in the system under normal operating conditions. Additionally, following a steam generator tube rupture, the main steam isolation system provides controls for reducing accidental releases, as discussed in section 10.3 and chapter 15. Detection of radioactive leakage into and out of the system is facilitated by area radiation monitoring (discussed in subsection 12.3.4), process radiation monitoring (discussed in section 11.5), and steam generator blowdown sampling (discussed in subsection 10.4.8).

H. In the event of loss of offsite power, loss of the steam generator feedwater pumps, or other situations that may result in a loss of main feedwater, the feedwater isolation signal will automatically isolate the feedwater system and permit the addition of auxiliary feedwater to allow a controlled reactor cooldown under emergency shutdown conditions. The auxiliary feedwater system is described in subsection 10.4.9.

I. Hydraulic instabilities are minimized from occurring during the various modes of operation. The feed ring, with J nozzles and thermal sleeve in each steam generator (model F), is designed so that the feed discharge nozzles are located on the top of the feed ring. The feedwater piping into the steam generators has been arranged so that it does not drain into the steam generators. This was accomplished by sloping the piping upward. These two features prevent the formation of a steam pocket in the feedwater piping which, when collapsed, creates the hydraulic instability.

10.4.7.4 Tests and Inspections

10.4.7.4.1 Preservice Valve Testing

The MFIVs, BFIVs, and feedwater regulating valves are checked for closing time prior to initial startup.

10.4.7.4.2 Preservice Pipe Testing

The main feedwater lines from the five-way restraint forgings in the isolation valve areas to the main steam tunnel five-way restraint forgings are classified as non-Seismic Category 1 piping. However, these lines are designed to withstand a safe shutdown earthquake, and 100 percent volumetric inspection at installation is provided (i.e., 100 percent volumetric examination of shop and field longitudinal and circumferential welds).

10.4.7.4.3 Preoperational System Testing

Preoperational testing of the condensate and feedwater system is performed as described in chapter 14.

10.4.7.4.4 Inservice Inspections

The performance and structural and leaktight integrity of system components are demonstrated by continuous operation.

The redundant actuator power trains of each MFIV and BFIV are subjected to the following tests:

A. Closure Time

The valves are checked for closure time at each refueling.

B. Online Operability

While the condensate and feedwater system is in operation, the operability of the actuator may be checked periodically by exercising the MFIV to approximately 90 percent of full open.

Additional discussion of inservice inspection of ASME Code Class 2 and 3 components is presented in section 6.6.

10.4.7.5 <u>Instrumentation Applications</u>

The main feedwater instrumentation, as given in table 10.4.7-3, is designed to facilitate automatic operation, remote control, and continuous indication of system parameters. As described in chapter 7, certain devices are involved in the secondary cycle pipe rupture protection system.

The feedwater flow to each steam generator is controlled by a three-element feedwater flow control system to maintain a programmed water level in the steam generator. The three-element feedwater controllers regulate the feedwater regulating valves by continuously comparing the feedwater flow and steam generator water level with the programmed level and the pressure-compensated steamflow signal. (Refer to section 7.7.) The steam generator feedwater pump turbine speed is automatically varied to maintain sufficient discharge pressure to support feedwater flow demand. The pump speed is increased or decreased, in accordance with the speed signal by modulating the steam pressure at the inlet of the pump turbine drivers.

Both steam generator feedwater pump turbines are tripped upon either one of the following:

A. High-high level in any one steam generator.

B. Actuation of safety injection. (Refer to section 7.3.)

One steam generator feedwater pump turbine trips when any one of the following directly affects it:

A. Turbine low bearing oil pressure.

B. Pump low bearing oil pressure.

C. Turbine overspeed.

D. Low vacuum (median).

E. Low suction pressure (median).

F. Thrust bearing wear.

G. Manual.

Yarway automatic recirculation valves are provided in the discharge lines from each condensate and heater drain pump. The purpose of the valves is to protect the respective pumps from becoming vapor bound, with subsequent internal pump damage and/or failure to pumps resulting from pumping rates below the minimum pump flow conditions stipulated by the pump manufacturer.

A flow element is installed in the suction of each of the steam generator feedwater pumps to provide the control signal to open the minimum recirculation flow valves, in order to protect the steam generator feedwater pumps.

A flow element with two flow transmitters is located on the inlet to each of the four steam generators to provide signals for the three-element feedwater control system.

In the event that a feedwater heater experiences a sizable number of tube leaks or a feedwater heater water level control valve fails closed, the main turbine is protected from failure resulting from flooding on the shell side of a feedwater heater and subsequent water induction into the moving turbine blades. This is done by the steam extraction lines to that heater by closing the respective bleeder trip and isolation valves and opening the high-level dump control valves that dump the heater excess drains to the condenser.

The total water volume in the condensate and feedwater system is maintained through automatic makeup and rejection of condensate to the condensate storage tank. The system makeup and rejection are controlled by the condenser hotwell level controller.

The system water quality requirements are maintained through the injection of hydrazine and alternate amine into the condensate system. The alternate amine and hydrazine injection is controlled by pH and the hydrazine residual in the system, which is continuously monitored by the process sampling system.

A temperature sensor is installed just below each elbow in the main and bypass feedwater piping welded to the Westinghouse steam generator inlet feedwater nozzles. These sensors may be used to provide a backup source for plant calorimetric data. These RTDs may also be used as inputs in the normal plant calorimetric data.

Instrumentation, including pressure indicators, flow indicators, and temperature indicators, required for monitoring the system is provided in the control room.

10.4.8 STEAM GENERATOR BLOWDOWN PROCESSING SYSTEM

The steam generator blowdown processing system (SGBPS) accepts water from each steam generator blowdown line, processes the water as may be required, and delivers the processed water either to the condensate system or to the waste water retention basin. Process steps include cooling with heat recovery, pressure reduction, filtration, and ion exchange. The purpose of the SGBPS is to remove impurities resulting from primary coolant or circulating water inleakage which are concentrated in the steam generator by the evaporative process.

10.4.8.1 Design Bases

The SGBPS is designed to maintain optimum secondary side water chemistry during normal operation and during anticipated operational occurrences of main condenser inleakage or primary to secondary leakage.

The ion removal capacity provided by the two series-operated mixed bed demineralizers is sufficient to allow the blowdown water to be recycled to the condensate system or discharged via the waste water retention basin. The demineralizers are designed for the full SGBPS design flow of 360 gal/min, although the blowdown rate normally is much less.

Design radioactivity in the untreated blowdown water assumes a primary-to-secondary side leakage of 1.0 gal/min (12,000 lb/day) coincident with operation with 1.0-percent failed fuel; i.e., clad leakage in the fuel rods providing 1 percent of full power. Additional assumptions are provided in table 10.4.8-1. The secondary side water isotopic concentrations under these assumptions are provided in table 10.4.8-2. This design basis radioactivity is used for in-plant design information such as shielding and radiation area evaluation.

Expected radioactivity levels in the secondary side water are determined based on a primary-to-secondary side leakage of 100 lb/day and reactor coolant activities based on American National Standards Institute (ANSI) Standard N237-1976. Specific activities for secondary side water using a realistic basis are provided in section 11.1.

The demineralizer resins are temperature sensitive, and if the process flow temperature exceeds 140°F, the resins will begin to degrade, releasing some of the stored ions. Process temperatures significantly greater than 140°F could potentially cause an unacceptable excursion in secondary side water chemistry and activity. To prevent this and to maintain the purification function of the demineralizers, the SGBPS is designed to reduce the blowdown liquid temperature to 130°F before it enters the demineralizers. The design includes instrumentation to actuate an alarm if the water going to the demineralizers exceeds 140°F and to halt blowdown if the water temperature reaches 147.5°F.

Because portions of the SGBPS are classified as high energy lines, design features are provided to rapidly isolate the blowdown path should a rupture occur in the system piping outside of the containment.

Other instrumentation is provided in the SGBPS design to ensure that the operation is maintained in established parameter ranges. Description of this instrumentation is provided in paragraph 10.4.8.2.

While the design flow capacity of the SGBPS is 360 gal/min, the flow from each steam generator can be varied from 5 to 90 gal/min at the discretion of the operator. Estimated average total blowdown flow is 240 gal/min. Continuous blowdown from each steam generator is necessary to counteract the evaporative concentration occurring in each steam generator. The SGBPS may be isolated for maintenance and repair provided that the secondary side water chemistry does not go beyond specifications (subsection 10.3.5). The SGBPS is capable of purging the steam generators during all modes of operation, including shutdown conditions.

Process design parameters and equipment design capacities are provided in tables 10.4.8-3 and 10.4.8-4, respectively.

The design of the SGBPS conforms with the recommendations of Regulatory Guide 1.143, Revision 1. Those portions of the SGBPS extending from the steam generators to the outermost containment isolation valves are Safety Class 2 and Seismic Category 1. The remainder of the system is classified Non-Nuclear Safety Class since it performs no function related to safety. Seismic and safety classifications and design codes of individual pieces of equipment are provided in table 3.2.2-1.

10.4.8.2 System Description

The SGBPS piping and instrumentation diagram is shown in drawings 1X4DB159-1, 2X4DB159-1, 1X4DB159-2, 1X4DB159-3, 1X4DB179-1, and 1X4DB179-2. Location of SGBPS major components is indicated on general arrangement drawings provided in section 1.2. The SGBPS process flow diagram in figure 10.4.8-1 includes process parameters for a selected number of points in the flow hydraulic path. These process parameters are based on a design basis maximum blowdown rate of 90 gal/min per steam generator, realistic secondary side water activity levels (see section 11.1 for values and assumptions), and demineralizer decontamination factors consistent with reference 1.

Blowdown is drawn from each of the four steam generators at 5 to 90 gal/min each. The blowdown flow is controlled administratively to permit blowdown to match the variable nature of water quality. The blowdown water is extracted from each steam generator through a sparger pipe located just above the tube sheet where impurities are expected to accumulate. A sample flow is continuously diverted from each steam generator blowdown flow. These sample flows are analyzed to ascertain water chemistry conditions for each of the steam generators.

The blowdown from each steam generator passes through a separate blowdown heat exchanger where it is cooled with condensate. The condensate outlet temperature from the heat exchanger is controlled to a value consistent with its disposition in the feedwater train and which optimizes heat recovery. After leaving the blowdown heat exchanger, each separate blowdown flow passes through a flow control valve. The four blowdown flows are then combined, and if the temperature of the blowdown was not sufficiently reduced by the blowdown heat exchangers due to high-condensate temperatures, the blowdown flow is directed to the trim heat exchanger where it is cooled to below the demineralizer resin temperature limits. Otherwise, the trim heat exchanger is bypassed. Downstream of the trim heat exchanger is a temperature element that measures the temperature of the flow entering the demineralizer. A high-temperature signal (140°F) actuates an alarm and a high-high signal (147.5° F) isolates blowdown. The blowdown next flows through a pressure control valve that, with varying blowdown rates, maintains a constant upstream pressure at a high level to limit flashing of blowdown upstream of the blowdown heat exchangers. This valve is the boundary between the high- and low-pressure portions of the SGBPS. The low-pressure portion of the SGBPS is protected from overpressurization by a relief valve and by automatic isolation of blowdown on high pressure.

The blowdown now normally passes through the steam generator blowdown backflushable prefilter, a cartridge type filter, and through a cation demineralizer and a mixed bed demineralizer, connected in series. An increase in conductivity, at the outlet of each demineralizer, signals resin bed exhaustion. When the upstream cation bed is exhausted, flow is directed around both beds to the waste water retention building. Spent resin is transferred to the spent resin storage tank with the aid of the spent resin sluice pump or demineralized water header pressure, and new resin is charged to the demineralizer.

From the demineralizers, the blowdown passes through the two steam generator blowdown filters in parallel and the discharge radiation monitor. It is then normally returned to the condensate system or it can be discharged to the environment via the waste water retention basin. The discharge radiation monitor is provided downstream of the demineralizers to isolate the blowdown flow and to terminate the discharge to the waste water retention basin upon a high radiation signal.

Another radiation monitor is provided downstream of the pressure control valve and alarms on high radiation.

Although blowdown normally is processed through the demineralizers and is recycled, the prefilter, a cartridge type filter, and demineralizers may be bypassed, especially during plant startup and shutdown operations when the quality of the blowdown water is most likely to be poorest. This water is discharged to the environment.

The spent resin handling portion of the SGBPS serves to sluice spent resin out of the steam generator blowdown demineralizers and to a storage tank or directly to the solid waste disposal system. If the spent resin is stored in the spent resin storage tank, it is eventually transferred to the solid waste disposal system by pressurizing the tank with nitrogen.

10.4.8.2.1 Component Descriptions

A summary of principal component data is given in table 10.4.8-4. Component design codes are given in section 3.2.

A. Steam Generator Blowdown Heat Exchangers

The blowdown heat exchangers are provided to subcool the blowdown fluid streams from steam generator exit conditions to approximately 130°F, 10° below the recommended maximum operating temperature for the resin in the demineralizers. Direct cooling of the blowdown with condensate recovers some of the heat loss due to blowdown.

B. Trim Heat Exchanger

A trim heat exchanger cooled by turbine plant cooling water is used when condensate temperature significantly exceeds the design temperature for the blowdown heat exchanger. When condensate temperatures as high as 130°F are expected for significant periods of time, the trim heat exchanger, located downstream of the blowdown heat exchangers, can be valved into operation.

C. Steam Generator Blowdown Prefilter

The prefilter is of the backflushable type. It is designed to prevent release of radioactive particulates and to prevent plugging of the resin beds.

D. Steam Generator Blowdown Cartridge Filter

The cartridge filter is of the cartridge type. It is designed to prevent release of radioactive particulates and to prevent plugging of the resin beds.

E. Steam Generator Blowdown Outlet Filter

The outlet filters are of the cartridge type. They operate in parallel to filter the demineralizer train effluent, eliminating the possibility of recycling or discharging radioactive resin particles.

F. Steam Generator Blowdown Demineralizers

A cation bed and a mixed bed demineralizer are provided to remove ionic contaminants, including radioisotopes if present, from the blowdown fluid. These beds prepare the blowdown fluid for discharge or recycling.

G. Spent Resin Storage Tank

The spent resin storage tank provides a collection point for spent ion exchange resins from the steam generator blowdown mixed bed demineralizers. The tank is a vertical, cylindrical type and is designed to withstand an internal pressure sufficient to push the resins to the resin disposal area.

H. Steam Generator Blowdown Spent Resin Sluice Pump

This centrifugal pump is designed to sluice resin in a 3-in. pipe at 110 gal/min.

I. Steam Generator Blowdown Spent Resin Sluice Filter

This disposable cartridge filter catches any resin fines contained in the sluice water.

J. Steam Generator Drain Pump

This centrifugal pump is designed to drain the steam generator. The drain pump is rated at 200 gal/min at 265 ft head.

10.4.8.2.2 Instrumentation Application

The instrumentation for the SGBPS is primarily provided on the steam generator blowdown process panel (PSBP), which is located on level B of the auxiliary building. Alarms and control functions are provided as noted. There is also a common alarm on the main control board, which indicates any alarms on the PSBP.

10.4.8.2.2.1 <u>Temperature.</u>

A. Blowdown Heat Exchanger Inlet Temperature

Instruments provide local indication of the temperature of the blowdown fluid entering each blowdown heat exchanger.

B. Blowdown Heat Exchanger Coolant Outlet Temperature

Instruments maintain the outlet temperature of the condensate used for cooling in the blowdown heat exchangers and control the condensate flowrate. Temperature indication, a variable control setpoint, and high and low alarms are provided on the PSBP for each heat exchanger.

C. Blowdown Heat Exchanger Outlet Temperature

An instrument monitors the temperature of the blowdown stream downstream of the trim heat exchanger. Indication and a high-temperature alarm are provided on the PSBP. A high-temperature interlock terminates blowdown flow to protect the demineralizer resin.

D. Trim Heat Exchanger Inlet Temperature

An instrument monitors the temperature of the blowdown entering the trim heat exchanger. Indication is provided on the PSBP.

10.4.8.2.2.2 Pressure.

A. Blowdown Heat Exchanger Outlet Pressure

An instrument maintains a constant backpressure on the high-pressure section of the SGBPS by modulating valve PV-1151. The instrument's setpoint can be manually set by the operator at the PSBP. The set pressure and the position of the individual flow control valves establish the flowrate from each steam generator.

Pressure indication, a variable control setpoint, and a variable high alarm are provided on the PSBP.

B. Prefilter Inlet Pressure

An instrument monitors the inlet pressure to the prefilter. Indication is provided locally and on the PSBP. A high-pressure signal sounds an alarm on the PSBP and a high-high-pressure signal terminates blowdown flow.

C. Prefilter Outlet Pressure

An instrument locally indicates the prefilter outlet pressure. This instrument, used in conjunction with the prefilter inlet pressure instrument, is used to determine the pressure drop across the prefilter.

D. Cartridge Filter Differential Pressure

Instrument locally indicates differential pressure across the filter.

E. Outlet Filter Differential Pressure

Instruments locally indicate differential pressure across each outlet filter.

F. Spent Resin Storage Tank Pressure

An instrument provides indication of the spent resin storage tank pressure on the PSBP. A high-pressure alarm is furnished on the PSBP. This alarm indicates that the setpoint on the sluice pump discharge relief valve may be reached if the sluice pump is started.

G. Spent Resin Sluice Filter Differential Pressure

An instrument locally indicates the differential pressure across the sluice filter.

10.4.8.2.2.3 Level.

A. Spent Resin Storage Tank Level

An instrument indicates the total level in the resin storage tank; i.e., combined resin slurry and sluice water level.

High- and low-level alarms are provided, as well as a low-level interlock which shuts off the resin sluice pump. Level indication and alarms are located on the PSBP. Level indication is also provided on the PSPP.

10.4.8.2.2.4 Flow.

A. Steam Generator Blowdown Flow

Instruments monitor the blowdown flowrate from each steam generator with varying steam generator pressure (load). The instruments modulate flow control valves to maintain the desired blowdown flowrate for each steam generator. The instruments' setpoints are set at the miscellaneous systems equipment panel in the main control room. Flowrate indication is provided on the miscellaneous systems equipment panel and on the PSBP. A high-flow alarm is provided on the PSBP.

B. Total Flow

An instrument monitors the total blowdown flowrate through the SGBPS. A high-flow alarm is provided on the PSBP. Indication is provided both on the miscellaneous systems equipment panel and on the PSBP.

C. Discharge Flow

An instrument provides indication on the PSBP of the flowrate of blowdown being discharged to the environment.

D. Spent Resin Sluice Pump Flow

An instrument provides indication of spent resin sluice pump flow. A low-flow or low-level interlock shuts off the spent resin sluice pump. Indication and a low-flow alarm are given at the PSBP.

10.4.8.2.2.5 Discharge Radiation. An instrument monitors the activity of the fluid leaving the SGBPS. A high-activity signal from this instrument automatically terminates blowdown flow and isolates the discharge line to the environment. Indication and alarm are given in the main control room. This instrument is discussed in more detail in section 11.5.

10.4.8.2.3 System Operation

10.4.8.2.3.1 Plant Startup. Plant startup encompasses the operations that bring the reactor plant from cold shutdown to no-load power operating temperature and pressure.

During startup operations the steam generators are brought into chemical specification as rapidly as possible. High-blowdown rates (90 gal/min per steam generator) are the most effective means the operator has to reduce the solids content of the steam generators. The filters and demineralizers are normally bypassed during this operation with the blowdown discharged to the environment.

10.4.8.2.3.2 Normal Operation. Normal operation includes operation at steady power level, load follow operation, and hot standby.

During normal operation blowdown rates range from 5 to 90 gal/min per steam generator, as required. The typical, average blowdown rate is 240 gal/min (60 gal/min per steam generator). The cooled, filtered, demineralized effluent is normally recycled to the condensate system although it may be discharged if desired.

If there is any significant primary-to-secondary side leakage or main condenser inleakage, the blowdown demineralizers provide continuous cleanup to counteract the inflow of contaminants.

10.4.8.2.3.3 Plant Shutdown. Plant shutdown is defined as the operations that bring the reactor plant from normal operating temperature and pressure to cold shutdown for maintenance or refueling.

High blowdown rates can again be used to clean up the steam generators. As during startup operations, if secondary side activity levels permit, the filters and demineralizers are bypassed and the blowdown is discharged to the environment.

10.4.8.2.3.4 Resin Handling.

A. Resin Fluffing

The demineralizer is valved out of service, and the flow path is aligned from the spent resin sluice pump or the demineralized water header, through the process line of the demineralizer, through the Johnson screen at the top of the demineralizer, and back to the spent resin storage tank. The resin bed is backflushed for about 10 min to loosen it for sluicing. This operation may also be used to recover pressure drop due to bed fouling by backwashing particulates from the top layer of the resin into the spent resin storage tank. Such a recovery is useful when the resin is not ionically depleted.

B. Resin Sluicing

After the resin is loosened by backflushing, the resin outlet isolation valve is opened and the resin flows to the spent resin storage tank.

When resin activity in a demineralizer is low, the spent resin can be sluiced to a bulk disposal unit for low-level radioactive waste instead of to the spent resin storage tank. Demineralized water is added to the spent resin storage tank to replace any sluice water lost to the bulk disposal facility.

C. Resin Fill

After sluicing is completed, fresh resin must be added. The path to the drain header from the demineralizer is opened to allow overflow. The resin fill line is opened and resin added. After fresh resin is added, the fill line valve is closed and the flow path is realigned for normal demineralizer operation.

D. Resin Disposal

The resin in the spent resin storage tank is loosened before disposal by sending pressurized nitrogen or sluice water through the six sparging nozzles in the tank. The valves in the resin transfer line are opened to direct the spent resin to a container in the radwaste processing facility.

The tank is then pressurized with nitrogen to force the resin up through the resin transfer line to the disposal area. A single nozzle in the spent resin storage tank is provided to allow local fluidization with sluice water at the opening of the discharge pipe. During resin transfer, this nozzle is used to ease the flow of the resin slurry. After resin transfer is complete, the tank is vented to the plant vent and returned to atmospheric pressure. The resin transfer line is then backflushed to the spent resin storage tank to clear it of resin.

Since a certain amount of resin remains in the tank after a disposal operation, it may hinder the backflush operation. Therefore, the fluidizing nozzle is again utilized to facilitate the backwash operation.

10.4.8.3 Design Evaluation

10.4.8.3.1 Discharge of Radioactivity to the Environment

Tritium is not treated by this system, and during discharge operation, it is released to the environment unaffected by the SGBPS. It is assumed that the noble gases are stripped out of the steam generator so that the blowdown water is free from isotopes of krypton and xenon.

A. Design Discharge

When operating without steam generator leakage or when the system effluent is being recycled back to the main condenser, there will be essentially zero release of radioactivity from this system. When the system effluent is discharged to the environment while operating with concurrent fuel defects and steam generator leakage, radioactivity discharge rates will depend on the combinations of parameters such as percent of fuel defects, steam generator leakage rates, blowdown rate, and quantity of dilution flow. The system is designed to limit average discharge concentrations under these various combinations of conditions to isotopic concentrations consistent with meeting the maximum permissible site boundary concentrations for intermittent or off-normal releases (Appendix B, Table II of 10 CFR 20.1 - 20.601) and the offsite dose criteria outlined in 10 CFR 50, Appendix I. The total decontamination factor provided by the two blowdown demineralizers is 100 for cesium and rubidium and 1000 for all other isotopes. These decontamination factors are based on reference 1.

B. Expected Discharge

A determination of expected annual releases from the SGBPS is included in table 11.2.3-1.

10.4.8.3.2 Failure Analysis

If a failure in the SGBPS blocks the blowdown flow or prevents necessary processing of the blowdown water, reactor operation can continue without blowdown until the required maintenance is performed or until plant shutdown is necessitated because of inability to maintain secondary side water chemistry specification or radioactivity level requirements. While the SGBPS is not a safety-related system, its interfaces with other plant systems can adversely affect safety-related portions of the plant following a postulated pipe rupture in the high-energy portion of the system outside of containment. For this reason, safety-related flow sensors are installed in each blowdown line inside containment and temperature sensors are installed in areas where SGBPS high-energy piping is routed. These sensors automatically terminate blowdown if the blowdown flow in a line or ambient temperature in the affected area exceeds predetermined setpoints. The single-failure adequacy of the safety-related portion of the SGBPS is addressed in table 10.3.3-1.

10.4.8.4 Tests and Inspections

The SGBPS is in continuous operation during normal reactor operation. Periodic visual inspection and preventive maintenance are conducted using normal industry practice. The Flow-Accelerated Corrosion Program is credited as a license renewal aging management program for carbon steel components which may be affected by acidic effluent from the SGBPS demineralizers (see subsection 19.2.10).

Periodic tests and recalibration are performed on the radiation monitor in the SGBPS.

Periodic tests of the SGBPS containment isolation valves are performed to check operability, as described in the Technical Specifications.

10.4.8.5 REFERENCE

1. U.S. Nuclear Regulatory Commission, "Calculation of Releases from Pressurized Water Reactors," NUREG-0017, April 1976.

10.4.9 AUXILIARY FEEDWATER SYSTEM

The auxiliary feedwater system (AFW) is designed to supply feedwater to the steam generators whenever the reactor coolant temperature is above 350°F and the main feedwater system is not in operation; i.e., during startup, cooldown, or emergency conditions resulting in a loss of main feedwater. The system is a safety grade, Seismic Category 1, redundant system with Class 1E electric components as indicated in table 3.2.2-1. The AFW meets the additional requirements of II.E.1.1 and II.E.1.2 of NUREG-0737 as described in the paragraphs below.

10.4.9.1 Design Bases

Protection of the AFW from wind and tornado effects is discussed in section 3.3; flood protection is discussed in section 3.4; missile protection is discussed in section 3.5; protection against dynamic effects associated with the postulated rupture of piping is discussed in section 3.6; and environmental design is discussed in section 3.11.

10.4.9.1.1 Safety Design Bases

The AFW automatically provides feedwater for the removal of reactor core decay heat in order that there is no damage to the reactor core following a loss of main feedwater from a condition of full power to where the reactor coolant temperature is brought to the point at which the residual heat removal system (RHRS) may be placed in operation. The automatic initiating circuits are powered from the emergency buses. The motor-driven pumps are automatically sequenced on the emergency diesel generators.

The AFW is protected from the effects of natural phenomena. This system is designed to remain functional after a safe shutdown earthquake (SSE) or following a postulated hazard such as fire, internal missile, or high-energy line break (HELB).

The AFW and supporting systems ensure the required flow to the steam generators in the event of a single active failure. In the unlikely event that the control room must be evacuated, the AFW can be operated from the shutdown panels and the auxiliary feedwater panel as the primary means of feedwater supply. The AFW includes condensate storage tanks (CSTs) as described in subsection 9.2.6.

The AFW is designed with sufficient diversity to remain operable for a limited duration with neither offsite nor onsite ac power available. The system is designed to avoid the effects of hydraulic instability (water hammer). The AFW is also designed to permit testing of the pumps during normal plant operation.

The AFW is provided with flow limiting orifices to limit the auxiliary feedwater flow to a depressurized steam generator and to ensure that the required minimum flow is directed to the remaining three intact steam generators.

10.4.9.1.2 Power Generation Design Basis

The AFW supplies feedwater to the steam generators at a sufficient flowrate to support normal low-power transients such as startup, cooldown, and hot standby. Flowrate is varied as necessary to maintain stable plant conditions by throttling the motor-operated discharge isolation/flow control valves.

10.4.9.1.3 Codes and Standards

Codes and standards applicable to the AFW are listed in table 3.2.2-1. The AFW is designed and constructed in accordance with Safety Class 3 requirements up to the motor-operated valves in each branch line located upstream of the stop-check valves, which serve as the containment isolation valves; the remaining valves and piping to the main feedwater system are Safety Class 2.

10.4.9.2 System Description

10.4.9.2.1 General Description

The system consists of two motor-driven pumps, one steam turbine-driven pump, and associated piping, valves, instruments, and controls as shown on drawings 1X4DB161-2, 1X4DB161-3, and 1X4DB168-3. Table 10.4.9-1 provides data for the various active components in the AFW. Table 10.4.9-2 presents minimum system flow requirements for various modes of operation. A plan view of the AFW pumphouse is provided in drawing 1X4DE316.

Each motor-driven auxiliary feedwater pump is sized to supply the feedwater flow required for removal of 100 percent of the decay heat from the reactor. The turbine-driven pump is sized to supply up to twice the capacity of a motor-driven pump. The system capacity is sufficient to remove decay heat and to provide adequate feedwater for cooldown of the reactor coolant system (RCS) at an average of 50°F per hour but not to exceed 100°F in any 1 hour. Following transients or accidents, cooldown may begin within 1 to 2 hours at zero load hot standby.

For a transient or accident condition, the minimum flow is delivered within 1 min of any automatic auxiliary feedwater actuation signal to at least two effective steam generators.

Normal flow is from the CST to the auxiliary feedwater pumps. The design of the condensate storage tank provides for cold shutdown capability for a period of 9 hours; 4 hours at hot standby, followed by a 5 hours cooldown of the primary system at an average rate of 50°F per hour but not to exceed 100°F in any 1 hour to a temperature of 350°F. For Unit 2, a combined safety-related capacity in two CSTs supports an additional 3 h hot standby/cooldown for a total of 12 h prior to placing RHR in service. Table 10.4.9-2 provides the nuclear steam supply system (NSSS) required makeup rates to the steam generators for various situations. Refer to subsection 9.2.6 for a description of the condensate storage system.

The design of the AFW provides that auxiliary feedwater can automatically be supplied to the steam generators in approximately 1 min in the event that the normal source of feedwater is lost.

Provisions are incorporated in the AFW design to allow for periodic operation to demonstrate performance and structural and leaktight integrity. Leak detection is provided by visual examination and in the floor drain system described in subsection 9.3.3.

Refer to section 5.4 for a discussion of steam generator design features to preclude fluid flow instabilities, such as water hammer. The main 16-in. and bypass 6-in. feedwater connections on each steam generator are the highest point of each feedwater line downstream of the MFIV and the BFIV. An elbow, with a short transition piece, is connected directly to the steam generator main 16-in. and bypass 6-in. feedwater nozzles, which tend to minimize the portion of feedwater piping that can drain into the steam generator and become filled with steam. The feedwater lines contain no high-point pockets that could trap steam and lead to water hammer. The horizontal pipe length from the main and bypass feedwater nozzles of each steam generator is minimized. This feedwater system configuration coupled with the steam generator design minimizes the potential for steam backleakage and bubble collapse water hammer in the main and auxiliary feedwater lines. To complement the system and steam generator design, Westinghouse recommended feedwater system maintenance and startup procedural guidelines further preclude feedwater system water hammer conditions in the main and bypass feedwater lines. The routing of the main and bypass feedwater lines is shown in drawings 1K5-1305-062-01 and 1K5-1305-058-01.

10.4.9.2.2 Component Description

10.4.9.2.2.1 Motor-Driven Pumps. Two of the three auxiliary feedwater pumps are horizontal, centrifugal pumps driven by electric motors which are supplied with power from independent, Class 1E essential switchgear buses. Each pump has a capacity of 630 gal/min. Each motor-driven auxiliary feedwater pump is normally aligned to feed two steam generators. The pumps can take suction from either of two Seismic Category 1 CSTs, and their discharge piping is directed to each of the four steam generators.

10.4.9.2.2.2 Turbine-Driven Pump. A turbine-driven auxiliary feedwater pump provides system redundancy of auxiliary feedwater supply and diversity of motive pumping power. The pump is a horizontal, centrifugal unit with a capacity of 1175 gal/min. The turbine-driven auxiliary pump is normally aligned to feed four steam generators. Steam supply piping to the turbine driver is taken from two of the four main steam lines between the containment penetrations and the main steam isolation valves. Each of the steam supply lines to the turbine driver is equipped with a check valve and a normally open, dc motor- operated gate valve. These steam supply lines join to form a header which leads to the turbine through a normally closed steam admission valve and normally open trip/throttle valve, both of which are dc motor-operated, and a normally open electro-hydraulically operated speed governing valve. The steam lines contain provisions to prevent the accumulation of condensate in the lines. The turbine driver can operate with steam inlet pressures ranging from 90 to 1130 psig. Exhaust steam from the turbine driver is vented to the atmosphere above the auxiliary feedwater pumphouse roof. In the unlikely event that neither offsite nor onsite ac power is available, the turbine-driven pump can function normally for up to 4 h, which is the time that the batteries can sustain dc power during a station blackout. Station blackout coping is discussed in section 8.4. Each auxiliary feedwater pump is located in a separate room in the auxiliary feedwater pumphouse.

10.4.9.2.2.3 Pump Discharge Isolation/Flow Control Valves. The motor-operated isolation/flow control valves in the branch lines from each motor-driven auxiliary feedwater pump that serves an individual steam generator are powered from the same Class 1E bus as the respective motor-driven auxiliary feedwater pump and associated controls. The motor-operated isolation/flow control valves in the branch lines from the turbine-driven auxiliary feedwater pump that serve an individual steam generator are powered from the train C Class 1E dc distribution system. The motor-operated isolation/flow control valves can be controlled from either the control room or local panels to modulate the auxiliary feedwater flow to maintain the required steam generator water level. The valves are normally open during power generation, but can be throttled or closed as necessary. The valves receive an open signal upon receipt of the same signal which starts the respective auxiliary feedwater pump. The valves can be manually closed to isolate the flow to a faulted steam generator and thus minimize spillage losses.

During normal operation, the isolation/flow control valve operability status is continuously indicated in the control room, thus informing the operators that a valve is closed or the automatic opening function is overridden. Also, as shown on the failure modes and effects analysis (table 10.4.9-4), inadvertent closure of any one valve terminates only the auxiliary feedwater flow from one pump to one steam generator, with all other flow paths from the three pumps to the four steam generators unaffected. Thus, adequate auxiliary feedwater flow is ensured during postulated accident conditions.

10.4.9.2.2.4 Piping and Valves. The piping from the CSTs to the pump suction and from the pump discharge to the pump discharge locked open manual gate valve is seamless stainless steel. The remainder of the discharge piping is seamless carbon steel. Welded joints are used throughout the system, except at flanged equipment connections. The auxiliary feedwater pump suction, recirculation, and discharge lines outside the auxiliary feedwater pumphouse are heat traced. The discharge line from each pump is equipped with a check valve, and manually operated isolation valve. Each branch auxiliary feedwater line from the pump discharge that serves an individual steam generator is equipped with a motor-operated isolation/flow control valve, a manually operated isolation gate valve, and two stop-check valves. The motor-operated steam supply valves for the turbine-driven pump and the turbine-driven pump discharge motor-operated isolation/flow control valves are supplied with Class 1E dc power. The motor-driven pump, motor-operated isolation/flow control valves are supplied with Class 1E ac power.

The lines that supply water to the auxiliary feedwater pumps from the No. 1 CST are equipped with two locked open, manual butterfly valves; the lines from the No. 2 CST have one locked open manual and one normally shut motor-operated butterfly valve. All motor-operated valve positions are monitored in the control room to provide the operators with the status of system alignment at all times.

Auxiliary feedwater is provided to each of the four steam generators by a 4-in. carbon steel line that ties into the 6-in. bypass feedwater line downstream of the 6-in. bypass feedwater check valve as shown in drawing 1X4DB168-3. The 6-in. bypass feedwater line connects to the steam generator 6-in. auxiliary feedwater nozzle, which is provided with an internal upward- inclined pipe extension that ends in a horizontal section so that the incoming feedwater enters below the steam generator normal water level in a stream horizontal to the steam generator deck plates. This flow path is provided at low load or hot standby conditions when auxiliary feedwater flowrate and temperatures are low and during normal power operation from 15-percent to 100-percent load.

10.4.9.2.2.5 Pump Miniflow. Auxiliary feedwater pumps are provided with miniflow protection. Each motor-driven pump has a miniflow return line with a flow control valve which isolates when flow to the generators exceeds a preset minimum. The turbine-driven pump miniflow return line has a manual valve which may be closed when flow to the generators exceeds a preset minimum. The motor-driven pump miniflow and turbine-driven pump miniflow may be routed to CST 1 or CST 2.

10.4.9.2.3 System Operation

10.4.9.2.3.1 Plant Startup. During startup, the auxiliary feedwater pumps are used under manual control to supply feedwater from the CST to the steam generators until sufficient steam is available to operate the turbine-driven main feedwater pumps.

10.4.9.2.3.2 Normal Plant Operation. The AFW is not required during normal power generation. The pumps are placed in the automatic mode, lined up with the CST, and are available if needed.

10.4.9.2.3.3 Normal Plant Cooldown. During cooldown, the auxiliary feedwater pumps are used under manual control to supply water from the CST to the steam generators. Auxiliary feedwater flow to each steam generator is regulated by the motor-operated isolation/flow control valves. The auxiliary feedwater pumps are used until reactor coolant temperature drops to 350°F, at which point the RHRS is placed in service and further cools the reactor.

10.4.9.2.3.4 Emergency Operation. In addition to manual actuation capabilities, the AFW is aligned to be placed in service automatically in the event of an emergency. Upon the occurrence of a safety injection signal, low-low steam generator water level in any one steam generator, or trip of both main feedwater pumps, or an Anticipated Transient Without Scram (ATWS) Mitigation System Actuation Circuitry (AMSAC) actuation, the auxiliary feedwater isolation/flow control valves go to the full-open position and the two motor-driven auxiliary feedwater pumps are started and begin to pump the contents of the CST into the steam generators to remove heat from the RCS. The turbine-driven pump is actuated automatically on low-low water level in any two steam generators or an AMSAC actuation. A blackout sequence starts the turbine-driven pump and both motor-driven pumps. To start the turbine-driven pump, a dc motor-operated turbine steam admission valve in the steam supply line to the turbine driver is opened. The turbine is designed to start immediately without warmup. The operator can remotely manipulate the position of the auxiliary feedwater motor-driven isolation/flow control valves in order to control steam generator water level. Provision for local manual operation of the isolation/flow control valves is also provided.

Two seismic Category 1 CSTs are provided for each unit. The CSTs are redundant because the motor-driven pump miniflow and turbine-driven pump miniflow may be routed to either CST 1 or CST 2. For Unit 1, a safety-related volume of 340,000 gal in either CST is sufficient to supply the auxiliary feedwater pumps. For Unit 2, a combined safety-related volume of 378,000 gal in two CSTs is sufficient to support an additional 3 h hot standby/cooldown for a total of 12 h prior to placing RHR in service. In addition to the 340,000 gal, the CSTs contain an additional capacity exceeding the equivalent heat generated by the remaining three reactor coolant pumps for 9 hours after reactor trip.

Decay heat is removed from the reactor by boiling the feedwater in the steam generators and venting steam to the atmosphere through the power-operated atmospheric relief valves or the main steam safety valves. If the main condenser is available, the steam may be discharged via the turbine bypass system to the main condensers. When reactor coolant temperature drops to 350°F, cooldown is shifted to the RHRS.

A summary of system performance for various accident conditions is provided in table 10.4.9-3. As noted, the table includes flows to both the faulted and intact steam generators. Comparing these data with those in table 10.4.9-2, it is seen that minimum flow requirements for the intact steam generators are satisfied under all failure modes.

10.4.9.3 Safety Evaluation

The AFW, in conjunction with two Seismic Category 1 CSTs, provides a means of pumping sufficient feedwater to prevent damage to the reactor following a loss-of-main-feedwater incident as well as to cool down the RCS at a rate not to exceed 100°F in any 1 hour to a temperature of 350°F, at which point the RHRS can operate. At a minimum, the 340,000 gal of water in the CSTs for the exclusive use of the AFW is sufficient to supply water as follows:

- 226,500 gal of water to hold the plant for 4 hours at hot standby conditions followed by an additional 5 hours during plant cooldown to 350°F.
- 19,800 gal of water to dissipate the heat generated by one reactor coolant pump for 9 hours after the reactor trip.
- 72,000 gal of water to compensate the turbine-driven auxiliary feedwater pump runout flow for 30 min (single failure until operator action isolates failure).
- 17,790 gal of water to compensate the auxiliary feedwater flow spilled through a pipe break for 30 min (initiating event).

For Unit 2, an additional 38,000 gal of water is required to compensate for an additional 3 h of hot standby/cooldown prior to placing RHR in service. Therefore, two CSTs with a combined safety-related volume of 378,000 gal are sufficient to meet the safety-related AFW requirements for Unit 2.

Pump capacities and start times are calculated by the NSSS vendor such that these objectives can be realized. At minimum water levels in the CSTs, there is sufficient net positive suction head (NPSH) to the motor- and turbine-driven auxiliary feedwater pumps.

Pump discharge head is sufficient to establish the minimum necessary flowrate against a steam generator pressure corresponding to the accumulation pressure of the lowest main steam safety valve. The maximum times to start the electric motors and the steam turbine which drive the auxiliary feedwater pumps are chosen so that sufficient flowrates are established before the loss of feedwater can result in damage to the reactor core. Minimum flow requirements are satisfied by one motor-driven pump assuming a HELB in the turbine-driven pump circuit and a single active failure.

In accordance with Branch Technical Position (BTP) ASB 10-1, the AFW provides redundant and diverse means of supplying feedwater to the steam generators for cooling the RCS under emergency conditions. Either of the motor-driven pumps has the capability of supplying a minimum of 100 percent of the feedwater requirements for safe cooldown of the RCS. The turbine-driven auxiliary feedwater train is capable of providing 200 percent of the minimum required feedwater flow for residual heat removal.

The three-train configuration can thus perform its safety-related function, assuming a postulated failure in the discharge piping of one train concurrent with a single active component failure in another train.

In the event the steam generators become contaminated, manual action is required to secure steamflow from the affected steam generators in order to prevent spread of contamination to the turbine-driven pump.

The motor-driven pump discharges are cross-connected through two normally closed, manually-actuated isolation valves.

Complete physical and electrical separation is maintained throughout the pump controls, control signals, electrical power supplies, and instrumentation for each auxiliary feedwater pump. Each motor-driven pump and associated motor-operated isolation/flow control valves are powered from the corresponding diesel generator, while the auxiliary feedwater turbine-driven pump components are powered from the train C, Class 1E dc distribution system.

Instrumentation and controls for the operation of the AFW are provided at the shutdown panels and the auxiliary feedwater panel as well as in the control room. Instrumentation controls are discussed in paragraph 10.4.9.5.

The AFW, including the CSTs, is designed to Seismic Category 1 requirements. The components (and supporting structures) of any system, equipment, or structure which are not Seismic Category 1 and whose collapse could result in loss of a required function of the AFW through either impact or flooding are analytically checked to determine that they will not collapse when subjected to seismic loading resulting from the SSE or are upgraded to Seismic Category 1.

Diversity is provided in the type and number of pumps, power supplies, piping arrangements, and pump and valve controls so that any single failure will not negate the AFW's ability to perform its safety function. The features which provide this capability are shown in the system schematic, drawings 1X4DB161-2 and 1X4DB161-3. In the event of a total loss of all ac power, the AFW can fulfill its safety function with any coincident, single active mechanical failure except the failure of the turbine-driven pump itself and its steam inlet valve.

An AFW analysis was performed in accordance with Action Items II.E.1.1 and II.E.1.2 of NUREG-0737 using the methodology defined by Appendix III and Annex 1 of Appendix X of NUREG-0611 and NUREG-0635 to determine the system reliability and major contributions to AFW failure under various loss-of-main-feedwater transients. This reliability analysis is presented in appendix 10A.

Auxiliary feedwater capabilities to cope during a station blackout (SBO) are discussed in section 8.4. An SBO coping analysis was performed as required by 10 CFR 50, Section 50.63.

A failure modes and effects analysis is provided in table 10.4.9-4.

10.4.9.4 Inspection and Testing Requirements

Each of the auxiliary feedwater pumps is hydrostatically tested in accordance with American Society of Mechanical Engineers (ASME) Section III, Class 3. The turbine driver is given a hydrostatic test, a mechanical running test, and an overspeed trip test prior to shipment. Inservice inspection and testing is performed as discussed in section 3.9.

The wall thickness of pressure boundary castings of the pumps, turbines, trip valve, throttle valve, and governor valve are checked and recorded by the manufacturer. The entire AFW is hydrostatically tested after assembly is completed. The system, its initiating signals, and its circuits are capable of being tested periodically while the plant is at power, in accordance with the frequency specified in the Technical Specifications. In addition, there will be double verification of proper valve alignment after system maintenance.

10.4.9.5 Instrumentation Requirements

Safety-related display instrumentation related to the AFW is discussed in section 7.5. Information indicative of the readiness of the AFW prior to operation and the status of active components during system operation is displayed for the operator in the main control room, at the shutdown panels, and at the auxiliary feedwater panel.

Equipment normally relied upon to be operable is monitored by the bypass/inoperable status monitoring system. Should the component be bypassed (e.g., by operation of a control switch, loss of control power, etc.), it is indicated at the system level on the bypass/inoperable status board. This indication satisfies the intent of the bypass indication requirement of Institute of Electrical and Electronics Engineers (IEEE) 279-1971 and Regulatory Guide 1.47 and the control power surveillance requirement of IEEE 308-1973.

The indication and controls provided for the AFW are summarized in table 10.4.9-5.

The logic for auxiliary feedwater initiation upon loss of both main feedwater pumps is shown in drawing 1X6AA02-239. Hydraulic fluid pressure for the feedwater pump turbine controls is sensed by instrumentation in each turbine. Upon loss of hydraulic fluid pressure (indicative of loss of main feedwater pumps) in both turbines, the pressure switches actuate relays which transmit the auxiliary feedwater pump start signal. The pressure switches are nonsafety grade; the actuating relays serve as isolation devices, with the remainder of the circuitry being safety grade.

The AFW can be manually initiated from the control room and the remote shutdown panels. A single failure in the manual circuits will not result in the loss of system function as described in section 7.3. The motor-driven auxiliary feedwater pumps and valves are automatically actuated and sequenced to the emergency bus as described in drawings 1X3D-AA-K02A, 2X3D-AA-K02A, 1X3D-AA-K02B, and 2X3D-AA-K02B. The automatic initiation signals and circuits are designed so that their failure will not result in the loss of manual initiation from the control room in accordance with Regulatory Guide 1.62.

Two channels of safety grade auxiliary feedwater flow instrumentation are provided per steam generator. Diversity is provided by steam generator water level for each steam generator.

10.4.10 CONDENSATE AND FEEDWATER CHEMICAL INJECTION SYSTEM

10.4.10.1 Design Bases

10.4.10.1.1 Safety Design Bases

The condensate and feedwater chemical injection system serves no safety function and has no safety design basis.

10.4.10.1.2 Power Generation Design Basis

The system is designed to control the dissolved oxygen content and pH in the turbine cycle and the steam generators.

10.4.10.2 System Description

10.4.10.2.1 General Description

The condensate and feedwater chemical injection system is shown in drawing 1X4DB157. The system is designed and constructed in accordance with Quality Group D specifications. Design parameters for the major components are provided in table 10.4.10-1.

10.4.10.2.2 Component Description

A. Methoxypropylamine (MPA) Storage Tank

Stainless steel, 12,789-gal-capacity tank used for bulk storage of MPA.

B. MPA Storage Transfer Pumps

Vertical, centrifugal type pump mounted atop the MPA storage tank (single pump per unit).

C. MPA and Hydrazine Day Tanks

Stainless steel, 250-gal-capacity tanks furnished with gauge glass level indicator, low-level switch, and 10-liter metering cylinder for Hydrazine day tanks (meter cylinder removed from Unit 1 and Unit 2 MPA tanks).

D. Batch Day Tank

Stainless steel, 250-gal-capacity tank furnished with gauge glass level indicator, and 10-liter metering cylinder.

E. Condensate Chemical Feed Pump

Simplex unit, positive displacement, disc diaphragm type pump, with autoelectric stroke adjustment control. The pump is driven by an electric motor.

F. Steam Generator Layup Chemical Pump

Duplex unit, positive displacement, disc diaphragm type pump, with manual stroke adjustment control. The pump is driven by an electric motor.

G. Auxiliary Feedwater Layup Chemical Pump

Electric motor-driven, simplex unit, positive displacement, disc diaphragm type pump with manual stroke adjustment control.

H. Hydrazine Dispensing Pump/High Volume

Electric motor-driven, magnetic drive, portable drum pump.

I. Tank Mixing Pump

Electric motor-driven, horizontal centrifugal type pump.

J. Hydrazine Storage Tank

Stainless steel, 6,000-gal-capacity tank used for bulk storage of hydrazine.

K. Hydrazine Storage Transfer Pump

Vertical, centrifugal type pump mounted atop the hydrazine storage tank.

L. Deleted

M. Deleted

N. Deleted

10.4.10.2.3 System Operation

The methoxypropylamine (MPA) consumed by the VEGP is brought to the plant by truck or rail and is stored in the 12,789-gal storage tank in the yard. Two MPA transfer pumps, one for each unit, send the MPA to the unit MPA day tank and the unit batch tank. The hydrazine is brought to the plant in two ways: (1) bulk delivery by truck or rail and stored in the 6644-gal storage tank in the yard and (2) purchased and stored in 55-gal drums. Hydrazine is transferred to the hydrazine day tank and the batch day tank of each unit by using the hydrazine transfer pump or hydrazine dispensing pump.

The chemical injection systems for both units are located in the turbine building at el 195 ft 0 in. Each injection system consists of two subsystems. The first subsystem (for normal operation) has two day tanks, one for MPA and one for hydrazine. The tanks are connected to two autoelectric-controlled simplex metering pumps. One pump is capable of supplying the MPA and hydrazine required for one unit; the second pump is a 100-percent spare. Each hydrazine simplex pump has a capacity of 11.7 gal/h at a rated head of 650 psig. Each MPA simplex pump has a capacity of 0.5 gal/h at a rated head of 650 psig. Two separate lines, one for hydrazine and one for MPA, and individual automatic controls are provided for injecting chemicals to the condensate and feedwater system during normal operation and the hot standby condition.

The second subsystem is principally for wet layup operation. It consists of one batch tank for storing mixed chemicals in predetermined concentrations and one duplex metering pump for sending the mixed chemicals to the four steam generators during wet layup conditions. Two chemical lines run from the chemical injection area in the turbine building to the steam generators, one line to steam generators 1 and 4 and the other to steam generators 2 and 3. One simplex metering pump is also provided to send the mixed chemicals to lay up the condensate and feedwater system during cold shutdown, as well as the auxiliary feedwater system whenever necessary. The duplex metering pump and the simplex metering pump are rated at 100 gal/h and a head of 1200 psig and are interconnected to serve as spare for each other.

For the hydrazine day tank, MPA day tank, and batch tank a condensate supply, chemical metering cylinder, and mixing pump are provided. The chemical is measured in the metering cylinder before being drained to the day tank or batch day. The condensate supply provides water from the condensate pump discharge to the day tanks and the batch tank to dilute the concentrated chemicals to predetermined concentrations. The mixing pump provides the agitation required for the thorough mixing of the chemicals and the condensate.

10.4.10.3 Safety Evaluation

The condensate and feedwater chemical injection system is not a safety-related system.

10.4.10.4 Tests and Inspections

The condensate and feedwater chemical injection system is operationally checked before initial plant startup to ensure proper functioning of the feed systems and chemical makeup sensors.

10.4.10.5 Instrumentation Applications

Instrumentation is provided for manual and automatic control of the chemical injection system; a condensate flow signal and hydrazine and specific conductivity analyzers provide indication of water quality and provide inputs to the automatic mode chemical injectors. Local and remote hydrazine and conductivity alarms are provided to monitor the performance and protect components of the system. Wet layup operations are manually controlled, relying on the results of grab samples and local indications.

TABLE 10.4.1-1 (SHEET 1 OF 2)

MAIN CONDENSER DESIGN DATA

Condenser Data	
Condenser type	Ecolaire Model 165-TSMC2-45, two-pass, single-pressure, vertically divided surface condenser unit in three shells
Condenser overall dimensions per shell	73 ft 11 in. x 27 ft 0 in. x 61 ft 8 in. high
Hotwell storage capacity	8600 ft^3/3 shells
Heat transfer capability	7.9514 x 10^9 Btu/h
Surface area	825,000 ft^2
Steam load	8,575,935 lb/h
Shell pressure (design)	Full vacuum to 27 psig (hydrostatic head)
Circulating water	509,600 gal/min
Water box pressure (design)	80 psig
Tube-side inlet temperature	89°F
Tube-side temperature rise	33.01°F
Condenser friction (ft of H_2O)	18 ft at 89°F
Condenser outlet temperature	120.6°F
Water box material	Steel (ASTM A283, Grade C)
Condenser Tube Data	
Condenser tubes (main section)	1 in. OD 22 BWG
Tube material	Titanium (ASTM B338-76, Grade B)
Quantity	22,264/condenser shell

TABLE 10.4.1-1 (SHEET 2 OF 2)

Condenser tubes (periphery)	1 in. OD 18 BWG
Tube material	Titanium (ASTM B338-76, Grade B)
Quantity	1168/condenser shell
Total number of tubes	23,432/condenser shell
Effective tube length	44 ft 9 1/4 in.
Tube sheet material	Aluminum bronze (ASTM B171, alloy 614)
Support plates	15

TABLE 10.4.2-1

CONDENSER AIR EJECTION SYSTEM
DESIGN DATA

<u>Steam Jet Air Ejectors</u>	
Quantity	Two per unit
Holding requirements	3.5 in. Hg absolute at full load (VWO)
Available steam	930 psig and 1191.7 Btu/lb (normal operation)
Auxiliary steam	170 psig and 1238 Btu/lb
Applicable standards	ANSI B31.1 HEI Standards
<u>Condenser Exhausters</u>	
Quantity	Two per unit
Hogging requirements	Evacuate total condenser volume from atmospheric pressure to less than 5 in. Hg absolute in 2 h or less.
Holding requirements	Maintain 36 sf^3/min exhaust flow at 1.0 in. Hg absolute based on an initial temperature difference of 25°F.

TABLE 10.4.5-1

DESIGN PARAMETERS FOR MAJOR COMPONENTS
OF CIRCULATING WATER SYSTEM

Circulating Water Pump

Quantity	Two per unit
Flowrate (gal/min)	242,300
Head (ft)	95

Natural Draft Cooling Tower

Quantity	One per unit
Design wet bulb temperature (°F)	78
Design relative humidity (%)	51
Design inlet temperature (°F)	122
Design outlet temperature (°F)	89
Design temperature rise (°F)	33
Design flowrate (gal/min)	509,600
Design bypass flowrate (gal/min)	300,000

TABLE 10.4.6-1 (SHEET 1 OF 3)

CONDENSATE FILTER/DEMINERALIZER SYSTEM
MAJOR COMPONENTS

1. Powdex Filter/Demineralizer Vessel

   Qty.: 5 per unit
   Dia.: 72 in. (x 96 in. straight side)
   Code: ASME Boiler and Pressure Vessel Code, Section VIII
   Design press.: 700 psig
   Internal filter elements: qty. 420 per vessel, polypropylene-wound or nylon-wound elements with stainless steel support hardware
   Design flow: 5340 gal/min per vessel

2. Backwash Recovery Pump

   Qty.: 1 per unit
   Size: 6 x 8-15, single-stage, vertically split casing
   Type: horizontal centrifugal pump
   Rating: 1280 gal/min at 200 ft TDH, 150°F
   Motor: 125 hp, 1800 rpm

3. Spent Resin Pump

   Qty.: 1 per unit
   Size: 1 x 1 1/2-8, single-stage, vertically split casing
   Type: horizontal centrifugal pump
   Rating: 108 gal/min at 83 ft TDH, 150°F
   Motor: 10 hp at 3600 rpm

4. Hold Pump

   Qty.: 5 per unit
   Size: 1 1/2 x 2-9, single-stage, vertically split casing
   Type: horizontal centrifugal
   Rating: 70 gal/min at 60 ft TDH, 150°F
   Motor: 5 hp at 1800 rpm

TABLE 10.4.6-1 (SHEET 2 OF 3)

5. Precoat Pump

   Qty.: 1 per unit
   Size: 6 x 8-13
   Type: horizontal centrifugal pump
   Rating: 1680 gal/min of 5% slurry at 150°F and 100 ft TDH
   Motor: 60 hp at 1800 rpm

6. Backwash Pump

   Qty.: 1 per unit
   Size: 3 x 4-10
   Type: horizontal centrifugal
   Rating: 500 gal/min at 100 ft TDH
   Motor: 25 hp at 3550 rpm

7. Precoat Tank

   Dia.: 60 in. (x 5 ft high), lined with Plasite No. 7155 HHB

8. Overlay Tank

   Dia.: 36 in. (x 4 ft high), lined with Plasite No. 7155 HHB

9. Backwash Recovery Tank

   Dia.: 12 ft (x 10 ft high), lined with Plasite No. 7155 HHB

10. Overlay Pump

   Qty.: 1 per unit
   Size: 1 x 1 1/2-6
   Type: horizontal centrifugal
   Rating: 25 gal/min at 100 ft TDH, 10% resin slurry
   Motor: 5 hp at 3600 rpm

11. Resin Trap

   Qty.: 5 per unit
   Type: Basket, stainless steel

TABLE 10.4.6-1 (SHEET 3 OF 3)

12. Dirty Spent Resin Holding Tank

    Dia: 18 ft (x 10.5-ft high) lined with Plasite No. 7122

13. Clean Spent Resin Holding Tank

    Dia.: 18 ft (x 10.5-ft high) lined with Plasite No. 7122

14. Recirculation Pump

    Qty.: 2 shared
    Size: 3 x 4-13
    Type: horizontal centrifugal
    Rating: 360 gal/min at 118-ft TDH
    Motor: 20 hp at 1780 rpm

15. Dewatering Pump

    Qty.: 2 shared
    Size: 2 x 2
    Type: double diaphram, air powered
    Rating: 120 gal/min at 70-ft TDH for 100 psi air

16. Dewatering Filter

    Qty.: 4, shared
    Dia.: 14 1/8 in. (x 45 7/8-in. high)
    Design press.: 150 psi
    Internal filter elements: 10-micron cotton matrix
    Design flow: 120 gal/min

17. Pressure Filter Skid

    Qty.: 2 shared
    Pump rating: 120 gal/min at 70-ft TDH for 100 psi air

TABLE 10.4.6-2 (SHEET 1 OF 2)

IMPURITY CONCENTRATIONS IN INFLUENT CONDENSATE

Impurity		Concentration: Design Max. or Startup	Concentration: Expected Full-Load Operation at 1-gal/min Condenser Leak[(a)]
Iron (as Fe)	Soluble	40 ppb	4 ppb
	Insoluble	1000 ppb[(b)]	6 ppb
Copper (as Cu)	(Total)	Trace	Trace[(c)]
Chloride (as Cl)	Soluble	10 ppb[(d)]	1.7 ppb
Sodium (as Na)	Soluble	10.6 ppb[(d)]	2.5 ppb
Silica (as SO_2)	Soluble	10 ppb	0.4 ppb
Suspended solids	Insoluble	1400 ppb[(b)]	Equivalent to the insoluble iron
Influent pH at 25°C		7 to 9.6	8.8 to 9.2 during leak (9.2 to 9.6 if operated full time)
Influent conductivity at 25°C		~10 μmho	10 μmho at pH 9.6
Influent cation conductivity at 25°C		0.5 μmho	0.2 μmho
Radioactivity level (μ/Ci per ml condensate)		<5.7 (10^{-5})	Negligible

a. The maximum condenser leak for allowable water quality maintenance that can be accommodated continuously at full-load operation with continuous recoating (no 2-h-lag allowance) and with a 0.1-gal/min steam generator leak of primary coolant is 61 gal/min.

b. This might be as much as 4000 ppb for several hours at initial plant startup. This value is used for equipment specification purposes, since without condenser tube leaks, sodium and chloride concentrations are much lower as a result of cleanup by the steam generator blowdown demineralizer. The maximum sodium and chloride input is that which occurs during a condenser tube leak. The maximum allowable concentration is determined by the rate at which the demineralizer units can be kept continuously in service without breakthrough.

TABLE 10.4.6-2 (SHEET 2 OF 2)

c. The secondary system does not use copper or copper nickel.

d. The design maximum influent impurity level is operationally accommodated by the powdered resin units at normal coating coverage. The time for exhaustion is not shorter than the time required to backwash, precoat, and apply resins to a unit, allowing 2 additional hours for these operations to start. The allowable influent impurity level depends on the pounds of resin, exchange capacity of the resin, backwash time, backwash handling time (1 h), and the circulating water analysis. Paragraph 10.4.6.2.4.3 gives a typical analysis of the condenser leakage circulating water. The maximum condenser leak under these conditions is 27 gal/min.

TABLE 10.4.6-3

WATER CHEMISTRY FOR REACTOR COOLANT SYSTEM

Item	Value
Electrical conductivity	Determined by the concentration of boric acid and alkali present. Expected range is < 1 to 40 μmho/cm at 25°C.
Solution pH	Determined by the concentration of boric acid and alkali present. Expected values range between 4.2 (high boric acid concentration) to 10.5 (low boric acid concentration) at 25°C. Values will be 5.0 or greater at normal operating temperatures.
Oxygen	0.1 ppm max.
Chloride	0.15 ppm max.
Fluoride	0.15 ppm max.
Hydrogen	25 to 50 cm^3 (STP)/kg H_2O
Total suspended solids	0.2 ppm max.
pH control agent (Li^7/OH)	Lithium is coordinated with boron in accordance with the principles of the EPRI PWR Primary Water Chemistry Guidelines, Volume 1.
Boric acid	Variable from 0 to 4000 ppm boron

TABLE 10.4.6-4 (SHEET 1 OF 2)

WATER CHEMISTRY FOR CIRCULATING WATER SYSTEM

Constituent	Concentration
Hardness (ppm as $CaCO_3$)	184
Total dissolved solids (ppm)	407
Suspended matter (ppm)	Estimated to vary from 20-100
Ammonia (ppm as N)	0.22
Total sulfides (ppm as S)	0.0
Silica (ppm as SiO_2)	45.0
Iron (ppm as Fe)	1.8
Manganese (ppm as Mn)	0.0
Calcium (ppm as Ca)	39.0
Magnesium (ppm as Mg)	21.0
Sodium (ppm as Na)	43.8
Potassium (ppm as K)	11.4
Bicarbonate (ppm as HCO_3)	76.9
Sulfate (ppm as SO_4)	137.01[a]
Chloride (ppm as Cl)	28.8
Nitrate (ppm as NO_3)	1.7
Fluoride (ppm as F)	0.2
Free chlorine (intermittent ppm)	0.2
Phosphate (ppm as PO_4)	0.5
pH value	7.0-7.5[b]
Free CO_2 (ppm)	3

TABLE 10.4.6-4 (SHEET 2 OF 2)

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a. This sulfate concentration results from the sulfuric acid treatment for scale control when adjusting the pH to the 7.0 to 7.5 range.

b. The VEGP will optimize the water stability, adjusting acid feed toward a Ryzner index in the range of 7.0 to 7.5. The pH will be higher (maximum 8.5), the sulfate lower, and the bicarbonates higher than shown.

TABLE 10.4.7-1

CONDENSATE AND FEEDWATER SYSTEM COMPONENT FAILURE ANALYSIS

Component	Failure Effect on Train	Failure Effect on System	Failure Effect on RCS
Condensate pump	None. Condenser hotwells are interconnected into a common condensate pump suction header.	Operation continues at full capacity, using standby pumps.	None.
No. 1, 2, or 3 feedwater heater	One train of No. 1, 2, and 3 feedwater heaters is isolated. Remaining trains continue to operate.	Operation continues at reduced capacity, using parallel feedwater heater train and bypass line. Load must not exceed 91 percent of VWO rating to protect the low-pressure turbines from excessive exhaust flow.	None. No. 4 feedwater heater will maintain normal outlet feedwater temperature under this condition.
Heater drain tank	Extraction steam to corresponding No. 4 feedwater heater must be isolated. Drains from respective No. 6 and 5 feedwater heaters are dumped to condenser.	Operation continues at reduced capacity.	RCS reduces reactor power to compensate for reduced feedwater temperature.
Heater drain pump	None. Parallel pump with condensate pumps have sufficient capacity to handle full load.	No. 6, 5, and 4 heater drains of affected train are dumped to condenser.	RCS reduces reactor power to compensate for reduced feedwater temperature.
No. 4 or 5 feedwater heater	One train is isolated.	Condensate and feedwater system operation continues at full capacity, using parallel train and bypass line.	RCS reduces reactor and generator output power to compensate for reduced feedwater temperature.
Steam generator feedwater pump	None. Two parallel trains are interconnected.	Operations may continue at reduced capacity, using parallel pump if the reactor does not trip.	RCS reduces reactor power to compensate for reduced feedwater flow.
No. 6 feedwater heater	One train is isolated.	Condensate and feedwater system operation continues at full capacity, using parallel train and bypass line.	RCS reduces reactor and generator output power to compensate for reduced feedwater temperature.

TABLE 10.4.7-2 (SHEET 1 OF 2)

CONDENSATE AND FEEDWATER SYSTEM
DESIGN DATA FOR MAJOR COMPONENTS

Component	Data
Condensate Pump	
Type	Vertical, 34APKD - 8-stage
Driver	4500 hp, 1180 rpm motor
Quantity	Three per unit
Capacity	10,640 gal/min at 1250 ft head
Condenser	
Type	Two-pass, single-pressure, vertically-divided surface condenser
Size	825,000 ft^2 condensing surface in three shells
Tube Sheet	Aluminum bronze
Tubes	1-in. outside diameter - 22 BWG and 18 BWG titanium
Design Pressure	Water box - 80 psig; shell - 30 psig and vacuum
Feedwater Heaters (Low Pressure)	
Type	Horizontal, U-tube, shell and tube
Quantity	First through third stage - three each mounted in condenser neck; fourth and fifth stage - two each
Tubes	20 BWG - AW stainless steel
Shell	Carbon steel

Heater No.	Tube Design Pressure/Temperature	Shell Design Pressure/temperature
1	700 psig/300°F	50 psig and vacuum/300°F
2	700 psig/335°F	50 psig and vacuum/370°F
3	700 psig/375°F	100 psig and vacuum/460°F
4	700 psig/410°F	225 psig and vacuum/410°F
5	700 psig/450°F	350 psig and vacuum/450°F

TABLE 10.4.7-2 (SHEET 2 OF 2)

Component	Data
Feedwater Heater (High Pressure)	
Type	Vertical head down, U-tube, shell and tube
Quantity	Two - sixth stage
Tubes	18 BWG - AW stainless steel
Shell	Carbon steel

Heater No.	Tube Design Pressure/Temperature	Shell Design Pressure/temperature
6	2000 psig/490°F	525 psig/490°F

Feedwater Pump	
Type	Single-stage, centrifugal 20 x 17 HVF, variable speed
Driver	15170 BHP steam drive turbine
Quantity	Two per unit
Capacity	20,400 gal/min at 3000 ft NTDH at 5950 rpm

TABLE 10.4.7-3

CONDENSATE AND FEEDWATER SYSTEM CONTROLS AND INDICATION

Control/Indication	Control Room	Local
Steam generator feedwater control valves	X	
Steam generator startup control valves	X	
Steam generator feedwater isolation valves	X	
Hotwell temperature	X (computer)	X
Hotwell level	X	X
Condensate pump discharge temperature	X	X
Condensate pump discharge pressure	X	X
Condensate pump recirculation pressure		X
Condensate to steam jet air ejector (SJAE) condenser temperature	X (computer)	X
SJAE condenser outlet temperature		X
Steam packing exhauster condenser outlet temperature		X
Condensate to heater 1 flowrate	X	
Heater 1 condensate inlet temperature	X (computer)	X
Heater 2 condensate inlet temperature	X (computer)	X
Heater 3 condensate inlet temperature	X (computer)	X
Heater 3 condensate outlet temperature	X (computer)	X
Heater 4 condensate inlet temperature	X (computer)	X
Heater 4 condensate outlet temperature	X (computer)	X
Heater 5 condensate outlet temperature	X (computer)	X
Steam generator feedwater pump condensate inlet flowrate	X	
Steam generator feedwater pump discharge pressure	X	X
Heater 6 discharge header temperature	X	
Steam generator feedwater flow	X	
Feedwater bypass flow		X

TABLE 10.4.8-1

PARAMETERS USED TO CALCULATE DESIGN BASIS SGBPS SOURCES

Steam generator type	Model F
Total secondary side water mass (lb)	4.7×10^5
Steam generator steam fraction	0.0548
Total steam flowrate (lb/h)	1.6×10^7
Moisture carryover (percent)	0.25
Total makeup water feedrate (lb/h)	9.9×10^2
Total blowdown rate (gal/min)	108
Total primary-to-secondary leakrate (gal/min)	1.0
Iodine partition factor (mass basis)	100

TABLE 10.4.8-2

STEAM GENERATOR SECONDARY SIDE WATER SPECIFIC ACTIVITY

Nuclide	Activity (μCi/g)	Nuclide	Activity (μCi/g)
Br-83	2.5×10^{-4}	Y-90	5.3×10^{-7}
Br-84	3.5×10^{-5}	Y-91m	2.9×10^{-5}
Br-85	4.3×10^{-7}	Y-91	6.1×10^{-6}
I-129	4.2×10^{-10}	Y-92	4.0×10^{-6}
I-130	1.3×10^{-4}	Y-93	2.3×10^{-6}
I-131	2.7×10^{-2}	Zr-95	6.6×10^{-6}
I-132	9.3×10^{-3}	Nb-95	6.6×10^{-6}
I-133	3.1×10^{-2}	Mo-99	6.9×10^{-3}
I-134	7.0×10^{-4}	Tc-99m	6.5×10^{-3}
I-135	1.1×10^{-2}	Ru-103	5.8×10^{-6}
Rb-86	3.9×10^{-4}	Ru-106	1.4×10^{-6}
Rb-88	2.1×10^{-3}	Rh-103m	5.8×10^{-6}
Rb-89	8.0×10^{-5}	Rh-106	1.4×10^{-6}
Cs-134	4.1×10^{-2}	Ag-110m	1.4×10^{-5}
Cs-136	5.2×10^{-2}	Te-125m	2.8×10^{-6}
Cs-137	2.7×10^{-2}	Te-127m	3.0×10^{-5}
Ba-137m	2.5×10^{-2}	Te-127	8.1×10^{-5}
Cs-138	7.5×10^{-4}	Te-129m	1.9×10^{-4}
H-3	1.8	Te-129	1.3×10^{-4}
Cr-51	5.6×10^{-5}	Te-131m	2.2×10^{-4}
Mn-54	4.5×10^{-6}	Te-131	4.4×10^{-5}
Mn-56	5.7×10^{-5}	Te-132	2.8×10^{-3}
Fe-55	2.0×10^{-5}	Te-134	2.9×10^{-5}
Fe-59	5.3×10^{-6}	Ba-140	4.2×10^{-5}
Co-58	1.5×10^{-4}	La-140	1.8×10^{-5}
Co-60	1.9×10^{-5}	Ce-141	6.4×10^{-6}
Sr-89	7.8×10^{-5}	Ce-143	4.3×10^{-6}
Sr-90	2.2×10^{-6}	Ce-144	3.9×10^{-6}
Sr-91	5.0×10^{-5}	Pr-143	6.4×10^{-6}
Sr-92	4.3×10^{-6}	Pr-144	3.9×10^{-6}

TABLE 10.4.8-3

STEAM GENERATOR BLOWDOWN PROCESSING SYSTEM PARAMETERS

Design blowdown flowrate, per steam generator (gal/min)	90
Normal temperature of blowdown entering system (°F)	539-567
Normal temperature of blowdown leaving system (°F)	130
Normal temperature of condensate water entering system (°F)	100
Normal temperature of condensate water leaving system (°F)	350
Design pressure of high-pressure section (psig)	1185
Design temperature of high-pressure section (°F)	600
Design pressure of low-pressure section (psig)	300
Design temperature of low-pressure section (°F)	200
Design pressure of spent resin sluice section (psig)	150/220
Design temperature of spent resin sluice section (°F)	200
Design pressure of condensate section (psig)	600
Design temperature of condensate section (°F)	400

TABLE 10.4.8-4 (SHEET 1 OF 3)

STEAM GENERATOR BLOWDOWN PROCESSING SYSTEM
MAJOR COMPONENT PARAMETERS

Steam Generator Blowdown Heat Exchanger

Number	4
Design heat transfer rate (Btu/h)	2.22×10^7

	Process Side	Coolant Side
Design pressure (psig)	1300	600
Design temperature (°F)	600	400
Design flow (lb/h)	4000 (steam)/ 41,000 (liquid)	88,000
Inlet temperature (°F)	539	100
Outlet temperature (°F)	130	350
Fluid	Blowdown	Condensate
Material		
Tubes	Stainless steel	
Shell	Carbon steel	

Trim Heat Exchanger

Number	1
Design heat transfer rate (Btu/h)	5.4×10^6

	Process Side	Coolant Side
Design pressure (psig)	1300	150
Design temperature (°F)	250	250
Design flow (gal/min)	360	540
Inlet temperature (°F)	160	95
Outlet temperature (°F)	130	115
Fluid	Blowdown	Turbine plant cooling water
Material		
Tubes	Stainless steel	
Shell	Stainless steel	

TABLE 10.4.8-4 (SHEET 2 OF 3)

<u>Steam Generator Blowdown Prefilter</u>

Number	1
Type	Backflusable
Design pressure (psig)	375
Design temperature (°F)	200
Design flow (gal/min)	360
Particle retention	98 percent of 2-mm size
Material (vessel)	Stainless steel

<u>Steam Generator Blowdown Cartridge Filter</u>

Number	1
Type	Cartridge
Design pressure (psig)	300
Design temperature (°F)	150
Design flow (gal/min)	360
Particle retention	99 percent of 2-mm size
Material (vessel)	Carbon steel

<u>Steam Generator Blowdown Outlet Filters</u>

Number	2
Type	Cartridge
Design pressure (psig)	300
Design temperature (°F)	250
Design flow (gal/min)	250
Particle retention	98 percent of 25-mm size
Material (vessel)	Stainless steel

<u>Steam Generator Blowdown Demineralizers</u>

Number	2
Design pressure (psig)	300
Design temperature (°F)	250
Design flow (gal/min)	360
Resin volume (ft^3)	75
Material (vessel)	Stainless steel

TABLE 10.4.8-4 (SHEET 3 OF 3)

<u>Steam Generator Blowdown Spent Resin Storage Tank</u>

Number	1
Capacity (ft^3)	550
Design pressure (psig)	150
Design temperature (°F)	200
Material	Stainless steel

<u>Steam Generator Blowdown Spent Resin Sluice Pump</u>

Number	1
Design pressure (psig)	220
Design temperature (°F)	200
Design flow (gal/min)	110
Design head (ft)	165
Material	Stainless steel
Motor horsepower	15

<u>Steam Generator Blowdown Spent Resin Sluice Filter</u>

Number	1
Design pressure (psig)	300
Design temperature (°F)	250
Design flow (gal/min)	250
Particle retention	98 percent of 25-mm size
Material (vessel)	Stainless steel

<u>Steam Generator Drain Pump</u>

Number	1
Design pressure (psig)	255
Design temperature (°F)	140
Design flow (gal/min)	200
Design head (ft)	265
Material	Stainless steel
Rotational speed (rev/min)	3600
Motor horsepower	40

TABLE 10.4.9-1 (SHEET 1 OF 2)

AUXILIARY FEEDWATER SYSTEM COMPONENT DATA

Motor-Driven Auxiliary Feedwater Pump (per pump)

Quantity	2
Type	Horizontal, centrifugal, multistage, split case with packing or mechanical seals
Capacity, each (gal/min)	630
Minimum flow - design	175
- allowable	150
TDH (ft) -	3500
NPSH required, maximum (ft)	19
NPSH available, minimum (ft)	30
Material	
Case	Cast Steel
Impellers	Cast steel
Shaft	Forged steel
Design code	ASME Section III, Class 3
Seismic design	Category 1
Driver	
Type	Electric motor
Horsepower (hp)	900
Rpm	3600
Power supply	4160 V, 60 Hz, 3-phase, Class 1E
Source of power	Train A 1AA02 Train B 1BA03
Design code	National Electrical Manufacturers Association (NEMA)
Seismic design	Category 1

Turbine-Driven Auxiliary Feedwater Pump

Quantity	1
Type	Horizontal, centrifugal, multistage, split case with packing or mechanical seals
Capacity (gal/min)	1175
Minimum flow (gal/min)	145
TDH (ft)	3500
NPSH required, maximum (ft)	19
NPSH available, minimum (ft)	30

TABLE 10.4.9-1 (SHEET 2 OF 2)

Material	
Case	Cast Steel
Impellers	Cast steel
Shaft	Forged steel
Design code	ASME Section III, Class 3
Seismic Design	Category 1
Driver	
Type	Noncondensing, single stage, mechanical-drive steam turbine
Rpm	4200
Horsepower (hp)	1590
Design code	NEMA
Seismic design	Category 1
Motor-Driven Pump Isolation/Flow Control Valves	
Quantity	4 (2 per pump)
Type	ac-powered, motor-operated globe valve
Size (in.)	4
Cv	145
Design pressure (psig)	1800
Design temperature (°F)	120
Material	Carbon steel
Design code	ASME Section III, Class 2
Seismic design	Category 1
Turbine-Driven Pump Isolation/Flow Control Valves	
Quantity	4
Type	dc-powered, motor-operated globe valve
Size (in.)	4
Cv	145
Design pressure (psig)	1975
Design temperature (°F)	120
Material	Carbon steel
Design code	ASME Section III, Class 2
Seismic design	Category 1

TABLE 10.4.9-2

NSSS STEAM GENERATOR MAKEUP REQUIREMENTS

Power Level (MWt)	Plant Condition/Situation	Flow Requirements (gal/min)
3579	Main Feedline Rupture	510 to intact SGs
	Loss of Normal Feedwater	510 to intact SGs
	Plant Cooldown	470
	HELB in Turbine-Driven Pump System	510 to intact SGs
	Station Blackout	510 to intact SGs
	Non-LOCA Steam Line Break (Hot Zero Power)	Max. total flow < 3300
	Steam Generator Tube Rupture (Margin to Overfill)	Max. to a single SG < 576.5
	Steam Generator Tube Rupture (Dose Analysis)	Min. total flow > 1,600
	LOCA Mass and Energy Release	> 127.5 to each SG
	Steam Line Break Mass and Energy Release	Max. to faulted SG < 1,108 & Max. total flow < 2,430
	Small Break LOCA	510 to intact SGs

Note:
The main steam line rupture (hot shutdown) and main steam system failure (depressurization of secondary side) cases have been removed from table 10.4.9-3. It was determined that the Non-LOCA Steam Line Break (Hot Zero Power) case bounds these cases. Therefore, these cases are not included in this table. UFSAR subsections 15.1.4 and 12.1.5, table 15.1.2-1, and figures 15.1.5-9 and 15.1.5-10 are revised to retain these cases for historical purposes.

TABLE 10.4.9-3 (SHEET 1 OF 4)

UNIT 1 AUXILIARY FEEDWATER FLOW INFORMATION FOR VARIOUS POSTULATED EVENTS

	Accident Description				Flow to the Intact Steam Generator (gal/min)					AFW Flow Information					
	Pressure (psia)									Pump Discharge Flow (gal/min)			Pump Recirculation Flow (gal/min)		
Case	Faulted SG	Intact SG	Temp. Water (°F)	Number of Pumps in Operation	SG1	SG2	SG3	Total	Flow to Faulted SG (gal/min)	Turbine-Driven Pump	Motor-Driven Pump A	Motor-Driven Pump B	Turbine-Driven Pump	Motor-Driven Pump A	Motor-Driven Pump B
Section 1.3-1															
Main feedwater line rupture[c]	14.7	1229	120	Turbine-driven pump and the motor-driven pump A are running	178.3	172.1	171.3	521.7	912.9	965.7	621.6	0	138.7	0[b]	0
Section 1.3-2															
Main feedwater line rupture[c]	14.7	1229	120	Only the two motor-driven pumps are running	0	264.0	260.8	524.8	739.2	0	742.8	532.0	0	0[b]	0[b]

TABLE 10.4.9-3 (SHEET 2 OF 4)

Case	Accident Description: Pressure (psia): Faulted SG	Accident Description: Pressure (psia): Intact SG	Accident Description: Temp. Water (°F)	Accident Description: Number of Pumps in Operation	Flow to the Steam Generator (gal/min): Intact SG1	Flow to the Steam Generator (gal/min): Intact SG2	Flow to the Steam Generator (gal/min): Intact SG3	Flow to the Steam Generator (gal/min): Intact SG4	Total Flow to SG (ga//min)	AFW Flow Information: Pump Discharge Flow (gal/min): Turbine-Driven Pump	AFW Flow Information: Pump Discharge Flow (gal/min): Motor-Driven Pump A	AFW Flow Information: Pump Discharge Flow (gal/min): Motor-Driven Pump B	AFW Flow Information: Pump Recirculation Flow (gal/min): Turbine-Driven Pump	AFW Flow Information: Pump Recirculation Flow (gal/min): Motor-Driven Pump A	AFW Flow Information: Pump Recirculation Flow (gal/min): Motor-Driven Pump B
Section 1.4 B[d]	NA	1239	120	Only motor-driven pump A running	259.7	0	0	258.9	518.6	0	525.8	0	0	0	0
Section 1.4 C[d]	NA	1239	120	Only motor-driven pump B running	0	260.4	257.1	0	517.5	0	0	524.6	0	0	0
Section 1.5 C[e]	NA	1239	120	Only turbine-driven pump running	176.9	175.9	175.0	177.3	705.1	858.7	0	0	139.6	0	0

a. Not used.

b. As shown in the description columns (see fifth column), three different cases were considered.

c. The motor-driven valves in the motor-driven pump recirculation lines are intended to close within a minute when discharging to the main discharge line and flow is above the minimum required. Thus, the motor-driven pump circulation flow was not considered.

d. Sections 1.4A, 1.4B, and 1.4C are for accident case involving loss of normal feedwater with loss of offsite power and with different pumps operating.

e. Section 1.5 is accident case involving station blackout with loss of normal feedwater and with only the turbine-driven pump operating.

TABLE 10.4.9-3 (SHEET 3 OF 4)

UNIT 2 AUXILIARY FEEDWATER FLOW INFORMATION FOR VARIOUS POSTULATED EVENTS

	Accident Description				Flow to the Intact Steam Generator (gal/min)					AFW Flow Information					
	Pressure (psia)									Pump Discharge Flow (gal/min)			Pump Recirculation Flow (gal/min)		
Case	Faulted SG	Intact SG	Temp. Water (°F)	Number of Pumps in Operation	SG1	SG2	SG3	Total	Flow to Faulted SG (gal/min)	Turbine-Driven Pump	Motor-Driven Pump A	Motor-Driven Pump B	Turbine-Driven Pump	Motor-Driven Pump A	Motor-Driven Pump B
Section 1.3-1															
Main feedwater line rupture[c]	14.7	1229	120	Turbine-driven pump and the motor-driven pump A are running	204.1	177.6	176.8	558.6	916.0	974.1	648.6	0	134.2	0[b]	0
Section 1.3-2															
Main feedwater line rupture[c]	14.7	1229	120	Only the two motor-driven pumps are running	0	263.1	259.8	522.9	760.7	0	764.3	530.1	0	0[b]	0[b]

TABLE 10.4.9-3 (SHEET 4 OF 4)

	Accident Description				Flow to the Steam Generator (gal/min)					AFW Flow Information					
	Pressure (psia)									Pump Discharge Flow (gal/min)			Pump Recirculation Flow (gal/min)		
Case	Faulted SG	Intact SG	Temp. Water (°F)	Number of Pumps in Operation	Intact SG1	Intact SG2	Intact SG3	Intact SG4	Total Flow to SG (gal//min)	Turbine-Driven Pump	Motor-Driven Pump A	Motor-Driven Pump B	Turbine-Driven Pump	Motor-Driven Pump A	Motor-Driven Pump B
Section 1.4 B[d]	NA	1239	120	Only motor-driven pump A running	259.9	0	0	259.1	519.0	0	526.2	0	0	0	0
Section 1.4 C[d]	NA	1239	120	Only motor-driven pump B running	0	259.2	256.0	0	515.2	0	0	522.4	0	0	0
Section 1.5 C[e]	NA	1239	120	Only turbine-driven pump running	179.9	178.9	178.1	180.2	717.1	865.9	0	0	134.8	0	0

a. Not used.

b. As shown in the description columns (see fifth column), three different cases were considered.

c. The motor-driven valves in the motor-driven pump recirculation lines are intended to close within a minute when discharging to the main discharge line and flow is above the minimum required. Thus, the motor-driven pump circulation flow was not considered.

d. Sections 1.4A, 1.4B, and 1.4C are for accident case involving loss of normal feedwater with loss of offsite power and with different pumps operating.

e. Section 1.5 is accident case involving station blackout with loss of normal feedwater and with only the turbine-driven pump operating.

TABLE 10.4.9-4 (SHEET 1 OF 31)

AUXILIARY FEEDWATER SYSTEM FAILURE MODES AND EFFECTS ANALYSIS

Item No.	Description of Component[(a)]	Safety Function	Plant Operating Mode[(b)]	Failure Mode(s)	Method of Failure Detection[(c)]	Failure Effect on System Safety Function Capability	General Remarks
1	Condensate tank makeup valve LV-5158, normally closed, fail closed air-operated valve	Maintains inventory or condensate storage tank V4-001 at nominal level of 480,000 ± 22,750 gal; also prevents inventory loss post-SSE	All	1A. Fails closed or fails to open upon command	1A. Position lights on QPCP coincident with low water level alarm LSL-5146 on QMCB	1A. None - One tank contains sufficient reserve without makeup to satisfy Regulatory Guide 1.139 with regard to 4 h at hot standby followed by a 5-h cooldown to RHR cut-in at RCS temperature of 350°F.	Total capacity of both tanks is 960,000 ± 45,500 gal.
				1B. Fails open or fails to close upon command	1B. Position lights on QPCP coincident with high water level alarm LSH-5146 on QMCB	1B. None - Tank may overflow, but surrounding dike holds more than 30-min makeup at maximum flow thus permitting operator corrective action.	
2	Condensate tank makeup valve LV-5162, normally closed, fail closed air-operated valve	Same as item 1, except tank is V4-002	All	Same as item 1	Same as item 1, except water level alarms are LSL-5147 and LSH-5147 on QMCB	None - Same as item 1	See item 1.
3	Degasifier feed pump suction valve HV-5087, normally open, fail closed, air-operated valve	Isolates condensate storage tank V4-001 from degasifier manually or upon low pump suction pressure	All	3A. Fails closed or fails to open upon command	3A. Position lights on QPCP	3A. None - Degasification not required for safe shutdown. Also, tank diaphragm limits oxygen buildup.	The degasifier is used periodically to control concentration of dissolved oxygen in the tanks to reduce potential for corrosion of the steam generators and the AFW piping (see also item 1).

TABLE 10.4.9-4 (SHEET 2 OF 31)

Item No.	Description of Component[(a)]	Safety Function	Plant Operating Mode[(b)]	Failure Mode(s)	Method of Failure Detection[(c)]	Failure Effect on System Safety Function Capability	General Remarks
				3B. Fails open or fails to close upon command	3B. Position lights on QPCP coincident with low pump suction pressure alarm PSL-5087 on QPCP	3B. None - Possible loss of tank inventory, but connection located above minimum level required for AFW reserve.	
4	Degasifier feed pump suction valve HV-5088, normally open, fail closed, air-operated valve	Same as item 3, except for condensate storage tank V4-002	All	Same as item 3	Same as item 3, except low suction pressure alarm is PSL-5088 on QPCP	None - Same as item 3	See item 3.
5	Not used						
6	Not used						
7	AFW pump suction valve HV-5119, normally closed MOV, train A	Admits water from condensate storage tank V4-002 to suction of motor-driven AFW pump P4-003 upon depletion of condensate storage tank V4-001 (valve actuated by remote-manual operation from either the control room or shutdown panel PSDA)	7A. Modes 1 and 2	7A1. Fails closed or fails to open upon command with condensate storage tank V4-001 depleted	7A1. Position lights on PSDA or QMCB, plus low suction pressure (PI-5129A on QMCB or PI-5129B on PSDA), plus open indication for pump miniflow valve FV-5155 on QMCB and PSDA due to low pump flow	7A1. None - AFW pumps P4-001 and P4-002 available to satisfy AFW requirements taking suction from tank V4-002 via HV-5113 and HV-5118, respectively. (Tank V4-001 presumed depleted. See item 21.)	For discussion of the ability of the AFW pumps to satisfy system requirements during various failure modes, see items 10, 11, and 12. Irrespective of source, pump miniflow returned to tank V4-002 (see also item 2). For the effects of loss of all ac power, see item 67.
				7A2. Spurious opening while tank V4-001 not depleted and pump running	7A2. Position lights on PSDA or QMCB and reduced tank V4-002 inventory (LI-5116A and LI-5104 on QMCB, LI-5116B on PSDA, and LI-5104A on PSDB)	7A2. None - Both tanks supplying AFW simultaneously but total consumption not affected. Also, makeup (see item 1) presumed available during modes 1 and 2.	

TABLE 10.4.9-4 (SHEET 3 OF 31)

Item No.	Description of Component[a]	Safety Function	Plant Operating Mode[b]	Failure Mode(s)	Method of Failure Detection[c]	Failure Effect on System Safety Function Capability	General Remarks
			7B. Modes 3 and 4	7B1. Same as item 7A1	7B1. Same as item 7A1 7	7B1. None - Same as 7A1	
				7B2. Same as item 7A2	7B2. Same as item 7A2	7B2. None - Same as item 7A2, except makeup not operable, but not required (see item 1A)	
			7C. Mode 6	7C1. Same as item 7A1	7C1. Same as item 7A1	7C1. None - Same as item 7A1	
				7C2. Same as item 7A2	7C2. Same as item 7A1	7C2. None - Same as item 7B2	
			7D. Mode 7	7D. Spurious opening	7D. Position indicator on QMCB or PSDA	7D. None - No loss of tank V4-002 inventory, since pump P4-003 is not operating.	
			7E. Mode 5	7E1. Same as item 7A1	7E1. Same as item 7A1	7E1. None - Same as item 7A1, except only pump P4-002 available, but this is sufficient for safe shutdown.	
				7E2. Same as item 7A2	7E2. Same as item 7A2	7E2. None - Same as item 7B2	
8	AFW pump suction valve HV-5118, normally closed MOV, train B	Same as item 7, except to suction of motor-driven AFW pump P4-002 and shutdown panel is PSDB	8A. Modes 1 and 2	8A1. Same as item 7A1	8A1. Same as item 7A1, except pressure indicators are PI-5128A on QMCB or PI-5128B on PSDB, and miniflow valve FV-5154 position indicated on QMCB and PSDB.	8A1. None - AFW pumps P4-001 and P4-003 available to satisfy AFW requirements taking suction from tank V4-002 via HV-5113 and HV-5119, respectively.	Same as item 7
				8A2. Same as item 7A2	8A2. Same as item 7A2	8A2. None - Same as item 7A2	

TABLE 10.4.9-4 (SHEET 4 OF 31)

Item No.	Description of Component[(a)]	Safety Function	Plant Operating Mode[(b)]	Failure Mode(s)	Method of Failure Detection[(c)]	Failure Effect on System Safety Function Capability	General Remarks
			8B. Modes 3 and 4	8B. Same as items 7A1 and 7A2	8B. Same as items 8A1 and 8A2, respectively	8B. None - Same as items 7B1 and 7B2, respectively	
			8C. Mode 6	8C. Same as items 7A1 and 7A2	8C. Same as items 8A1 and 8A2, respectively	8C. None - Same as items 7C1 and 7B2, respectively	
			8D. Mode 7	8D. Spurious opening	8D. Position lights on QMCB or PSDB	8D. None - No loss of tank V4-002 inventory, since pump P4-002 is not operating.	
			8E. Mode 5	8E1. Same as item 7A1	8E1. Same as item 8A1	8E1. None - Same as item 8A1, except only pump P4-003 available, but this is sufficient for safe shutdown.	
				8E2. Same as item 7A2	8E2. Same as item 7A2	8E2. None - Same as item 7B2	
9	AFW suction valve HV-5113, normally closed, dc-operated MOV, train C	Same as item 7, except to suction of turbine-driven AFW pump P4-001, and controlled from the control room and AFW panel PAFP	9A. Modes 3 and 4	9A1. Fails closed or fails to open upon command with condensate storage tank V4-001 depleted	9A1. Position lights on QMCB or PAFP, plus low pump suction pressure PI-5110A or PI-5110B on QMCB or PAFP, respectively	9A1. None - AFW pumps P4-002 and P4-003 available to satisfy AFW requirements taking suction from tank V4-002 via HV-5118 and HV-5119, respectively.	Turbine-driven AFW pump not ordinarily used during modes 1 and 2. (For discussion of the ability of the AFW pumps to satisfy system requirements during various failure modes, see items 10, 11, and 12.) The condensate storage facility is sized to accommodate 30-min spillage at turbine-driven pump (TDP) runout to allow for operator action to isolate the turbine and/or the TDP (for mode 5,
				9A2. Spurious opening while tank V4-001 not depleted and TDP pump running	9A2. Same as item 7A2, except position indicator on QMCB or PAFP	9A2. None - Both tanks supplying AFW simultaneously, but total consumption not effected. Makeup not operating but not required.	

TABLE 10.4.9-4 (SHEET 5 OF 31)

Item No.	Description of Component[a]	Safety Function	Plant Operating Mode[b]	Failure Mode(s)	Method of Failure Detection[c]	Failure Effect on System Safety Function Capability	General Remarks
							see item 9B). For the TDP, irrespective of source, the miniflow is returned to tank V4-001 via fixed flow orifice FO-5109.
			9B. Mode 5	9B. Fails open or fails to close upon command	9B. Same as item 9A2	9B. None - Manually operated valve HV-5097 can be closed to isolate pump, and HV-3009 (item 14) and HV-3019 (item 15) closed to stop turbine.	
			9C. Mode 6	9C. Same as items 9A1 and 9A2	9C. Same as items 9A1 and 9A2, respectively	9C. None - Same as items 9A1 and 9A2, respectively	
			9D. Mode 7	9D. Spurious opening	9D. Position indicator on QMCB or PAFP	9D. None - No loss of tank V4-002 inventory, since pump P4-001 is not operating.	
10	Motor-driven AFW pump P4-003 (train A)	Provides AFW to SGs 1 and 4 automatically as required or by remote manual actuation from the control room or shutdown panel	10A. Modes 1, 2, and 3	10A. Fails to start or stops running when required	10A. Pump status light shows green on QMCB or PSDA, plus low discharge pressure indicators PI-5141A and PI-5141B on QMCB or PSDA, respectively. Also miniflow valve FV-5155 (item 27) is open.	10A. None - Motordriven AFW pump (MDP) P4-002 available to satisfy 100% of AFW requirements. TDP P4-001 also available if required.	For HELB in TDP discharge (mode 5, items 10D and 11D), only one MDP is required to satisfy AFW requirements. (For the effects of loss of all ac power, see item 67.)
			10B. Mode 4, SG 1 or 4	10B. Spurious start or failure to stop upon command	10B. Pump status light shows red on QMCB or PSDA with same low pressure indicators	10B. None - MDP P4-002 and TDP P4-001 operate automatically to satisfy AFW	

TABLE 10.4.9-4 (SHEET 6 OF 31)

Item No.	Description of Component[a]	Safety Function	Plant Operating Mode[b]	Failure Mode(s)	Method of Failure Detection[c]	Failure Effect on System Safety Function Capability	General Remarks
					as for item 10A, plus low discharge pressure alarm PSL-5149 on QMCB if break upstream of flow limiting orifices.	requirements. Thirty-minute spillage through break accounted for in condensate tank sizing.	
			10C. Mode 4, SG 2 or 3	10C. Fails to start or stops running	10C. Same as item 10A	10C. None - TDP P4-001 operates automatically to satisfy AFW requirements.	
			10D. Mode 5	10D. Fails to start or stops running	10D. Same as item 10A	10D. None - MDP P4-002 operates automatically to satisfy AFW requirements.	
			10E. Mode 6	10E. Fails to start or stops running	10E. Same as item 10A	10E. None - MDP P4-002 and TDP P4-001 available.	
11	Motor-driven AFW pump P4-002 (train B)	Same as item 10, except provides AFW to SGs 2 and 3 and shutdown panel is PSDB	11A. Modes 1, 2, and 3	11A. Fails to start or stops running when required	11A. Pump status light shows green on QMCB or PSDB, plus low discharge pressure indicators PI-5140A or PI-1541B on QMCB and PSDB, respectively. Also, miniflow valve FV-5154 (item 28) is open.	11A. None - Same as item 10A, except MDP P4-003 provides backup AFW capability along with TDP P4-001.	See item 10.
			11B. Mode 4, SG 2 or 3	11B. Spurious start or failure to stop upon command	11B. Pump status light shows red on QMCB or PSDB, plus same low pressure indicators as item 11A, plus low discharge pressure alarm PSL-5148 on QMCB	11B. None - Same as item 10B, except backup MDP is P4-003.	

TABLE 10.4.9-4 (SHEET 7 OF 31)

Item No.	Description of Component[a]	Safety Function	Plant Operating Mode[b]	Failure Mode(s)	Method of Failure Detection[c]	Failure Effect on System Safety Function Capability	General Remarks
			11C. Mode 4, SG 1 or 4	11C. Fails to start or stops running when required	11C. Same as item 11A	11C. None - Same as item 10C	
			11D. Mode 5	11D. Fails to start or stops running	11D. Same as item 11A	11D. None - MDP P4-003 operates automatically to satisfy AFW requirements.	
			11E. Mode 6	11E. Fails to start or stops running	11E. Same as item 11A	11E. None - Same as item 10E, except backup MDP is P4-003.	
12	Turbine-driven AFW pump P4-001 (train C, with dc-powered controls)	Same as item 10, except provides AFW to all four SGs	12A. Mode 3	12A. Fails to start or stops running with steam admission valve HV-5106 (item 13) open	12A. Low pump discharge pressure alarm PSL-5108, plus low speed indication SI-15109A on QMCB, SI-15109B on PAFP, SC-15109 on PAFT, and SI-15109.	12A. None - MDPs P4-002 and P4-003 provide 100% of AFW requirements.	The turbine-driven AFW pump not ordinarily used during modes 1 and 2. (For FMEA of turbine steam admission valve, see item 13; and for main steam supply valves, see items 14 and 15.)
			12B. Mode 4 any SG	12B. Same as item 12A	12B. Same as item 12A	12B. None - Same as item 12A. Also, the MDPs satisfy minimum AFW requirements with allowance for 30-min spillage from faulted feedline.	
			12C. Mode 5	12C. Continues to run or fails to stop upon command	12C. Same as item 12A, except high pump speed indication SI-15109A on QMCB, SI-15109B on PAFP, SC-15109 on PAFT, and SI-15109; HV-5106 position	12C. None - MDPs P4-002 and P4-003 provide 100% of AFW requirements. The condensate storage tank capacity accounts for 30-min spillage at pump runout conditions. Main	

TABLE 10.4.9-4 (SHEET 8 OF 31)

Item No.	Description of Component[(a)]	Safety Function	Plant Operating Mode[(b)]	Failure Mode(s)	Method of Failure Detection[(c)]	Failure Effect on System Safety Function Capability	General Remarks
					lights on PAFP or QMCB show valve open	steam turbine supply valves HV-3009 (item 14) and HV-3019 (item 15) can be closed to terminate turbine steam supply.	
			12D. Mode 6	12D. Fails to start or stops running (HV-5106 open)	12D. Same as item 12A	12D. None - MDPs P4-002 and P6-003 available.	
13	AFW TDP turbine steam admission valve HV-5106, normally closed dc-powered MOV, train C	Controls steam admission to TDP turbine driver	13A. Modes 3, 4, and 6	13A. Fails closed or fails to open upon command	13A. Position lights on QMCB or PAFP coincident with low pump discharge pressure alarm PSL-5108 on QMCB	13A. None - MDPs P4-002 and P4-003 provide 100% of AFW requirements. (See also item 12B, mode 4.)	The turbine-driven AFW pump not ordinarily used for modes, 1, 2, or 7. (For FMEA of TDP, see item 12; and for main steam supply valves, see items 14 and 15.)
			13B. Mode 5	13B. Fails open or fails to close upon command	13B. Same as item 13A	13B. None - Same as item 12C	
14	Steam supply valve HV-3009 from SG 1 to TDP P4-001 turbine, normally open dc-powered MOV, train B	Admits steam from SG 1 to TDP turbine and also provides for isolation of the downstream piping in the event of a line break	14A. Modes 3, 4, and 6	14A. Fails closed or fails to open upon command	14A. Valve position lights on QMCB or PAFP	14A. None - 100% redundant steam supply to the TDP provided via HV-3019 from SG 2. Also, MDPs P4-002 and P4-003 provide 100% of AFW requirements.	See also table 10.3.3-1, item 33.
			14B. Mode 5	14B. Fails open or fails to close upon command	14B. Same as item 14A	14B. None - TDP steam admission valve can be closed to terminate steam supply. (See also item 12 for pump redundancy and design of CST capacity.)	

TABLE 10.4.9-4 (SHEET 9 OF 31)

Item No.	Description of Component[(a)]	Safety Function	Plant Operating Mode[(b)]	Failure Mode(s)	Method of Failure Detection[(c)]	Failure Effect on System Safety Function Capability	General Remarks
15	Steam supply valve HV-3019 from SG 2 to TDP P4-001 turbine, normally open dc-powered MOV, train A	Same as item 14A, but from SG 2	15A. Modes 3, 4, and 6	15A. Fails closed or fails to open upon command	15A. Valve position lights on QMCB or PAFP	15A. None - Same as item 14A, except redundant steam supply from SG 1 via HV-3009.	See also table 10.3.3-1, item 34. For containment isolation function, see table 6.2.4-1, item 19.
			15B. Mode 5	15B. Fails open or fails to close upon command	15B. Same as item 15A	15B. None - Same as item 14B	
16	Check valve 1301-U4-404 in steam supply line to TDP turbine from SG 1	Prevents crossflow between main steam lines to prevent blowdown of SG 2 in the event of MSLB in the No. 1 main steam line	16A. Mode 6 (MSLB in SG 1)	16A. Fails open	16A. Gradual decrease of water level in SG 2	16A. None - TDP performance degraded but MDPs available. Degrading upon break location, HV-3009 can be closed to terminate blowdown from SG 2 and permit continued TDP operation.	See also table 10.3.3-1, item 35.
			16B. Modes 3, 4, and 5	16B. Fails closed	16B. None	16B. None - Same as item 14A	
17	Check valve 1301-U4-008 in steam supply line to TDP turbine from SG 2	Same as item 16, except SG designations reversed	17A. Mode 6 (MSLB in SG 2)	17A. Fails open	17A. Gradual decrease of water level in SG 1	17A. None - Same as item 16A, except HV-3019 can be closed to terminate blowdown from SG 1 and permit continuous TDP operation.	See also table 10.3.3-1, item 36.
			17B. Modes 3, 4, and 5	17B. Fails closed	17B. None	17B. None - Same as item 15A	
18	Check valve 013 in supply line from condensate storage tank V4-001 to TDP P4-001 suction	Prevents backflow to tank V4-001 from V4-002 when former is depleted	Modes 1 through 6	18A. Fails open with HV-5113 open	18A. Water level equalized in tanks V4-001 and V4-002	18A. None - No loss of inventory also, HV-5090 and HV-5093 can be closed manually to prevent backflow.	This valve is required to function only if leaks are present between this valve and V4-001.

TABLE 10.4.9-4 (SHEET 10 OF 31)

Item No.	Description of Component[a]	Safety Function	Plant Operating Mode[b]	Failure Mode(s)	Method of Failure Detection[c]	Failure Effect on System Safety Function Capability	General Remarks
				18B. Fails closed with HV-5113 closed (HV-5113 normally closed - see item 9).	18B. Decreased TDP P4-001 suction pressure indication PI-5110A on QMCB and PI-5110 on PAFP	18B. None - Potentially damage to TDP P4-001 but MDP's P4-002 and P4-003 available. Also, HV-5113 can be opened to restore flow to MDP P4-001.	
19	Check valve 058 in supply line from condensate storage tank V4-001 to MDP P4-002 suction	Same as item 18	Modes 1 through 6	19A. Fails open with HV-5118 open	19A. Same as item 18A	19A. None - Same as item 18A	See item 18.
				19B. Fails closed with HV-5118 closed (HV-5118 normally closed - see item 8)	19B. Decreased MDP P4-002 suction pressure indication PI-5128A on QMCB and PI-5128 on PSDB	19B. None - Potential damage to MDP P4-002 but, TDP P4-001 and MDP P4-003 are available. Also, HV-5118 can be opened to restore flow to MDP P4-002.	
20	Check valve 033 in supply line from condensate storage tank V4-001 to MDP P4-003 suction	Same as item 18	Modes 1 through 6	20A. Fails open with HV-5119 open	20A. Same as item 18A.	20A. None - same as item 18A except manual valves are HV-5092 and HV-5095	See item 18.
				20B. Fails closed with HV-5119 closed (HV-5119 normally closed - see item 7).	20B. Decreased MDP P4-003 suction pressure indication PI-5129A on QMCB and PI-5129B on PSDA	20B. None - Potential damage to MDP P4-003, but TDP P4-001 and MDP P4-003 available. Also, HV-5119 can be opened to restore flow to MDP P4-003.	See item 18.
21	Check valve 051 in supply line from condensate storage tank V4-002 to TDP P4-001 suction	Prevents backflow from tank V4-001 to tank V4-002 when latter depleted.	Modes 1 through 6	21A. Fails open with HV-5113 closed	21A. None	21A. None - HV-5113 prevents backflow	During normal auxiliary feedwater operation, tank V4-001 is depleted before pump aligned

TABLE 10.4.9-4 (SHEET 11 OF 31)

Item No.	Description of Component[(a)]	Safety Function	Plant Operating Mode[(b)]	Failure Mode(s)	Method of Failure Detection[(c)]	Failure Effect on System Safety Function Capability	General Remarks
							to tank V4-002. (See also item 18A except tank is V4-002).
				21B.Fails open with HV-5113 open	21B.Same as item 18A	21B. None - no loss of inventory MDV HV-5113 and manual valve HV-5097 can be closed to prevent backflow.	
				21C. Fails closed	21C. Same as item 18B	21C. None - same as item 9A1	
22	Check valve 061 in supply line from condensate storage tank V4-002 to MDP P4-002 suction	Same as item 21.	Modes 1 through 6	22A. Fails open with HV-5118 closed	22A. None	22A. None - HV-5118 prevents backflow.	See items 18 and 21.
				22B. Fails open with HV-5118 open	22B. Same as item 18A	22B. None - Same as item 21B except valves are HV-5118 and HV-5098	
				22C. Fails closed	22C. Same as item 19B	22C. None - Same as item 8A1	
23	Check valve 052 in supply line from condensate storage tank V4-002 to MDP P4-003 suction.	Same as item 21	Modes 1 through 6	23A. Fails open with HV-5119 closed	23A. None	23A. None - HV-5119 prevents backflow	See item 10.
				23B. Fails open with HV-5119 open	23B. Same as item 18A	23B. None - Same as 21B except valves are HV-5119 and HV-5099	
				23C. Fails closed	23C. Same as item 20B	23C. None - Same as item 10A	
24	Check valve 001 in MDP P4-003 discharge	Prevents reduction in AFW flow to SG via backflow through P4-003 when pump is not running	24A. Modes 1, 2, and 3	24A. Fails closed with MDP P4-003 running	24A. Position indicators for miniflow valve FV-5155 on QMCB	24A. None - Same as item 10A	See item 10.

TABLE 10.4.9-4 (SHEET 12 OF 31)

Item No.	Description of Component[(a)]	Safety Function	Plant Operating Mode[(b)]	Failure Mode(s)	Method of Failure Detection[(c)]	Failure Effect on System Safety Function Capability	General Remarks
					or PSDA show FV-5155 open		
			24B. Mode 4, SG 2	24B. Same as item 24A	24B. Same as item 24A	24B. None - Same as item 10C	
			24C. Mode 5	24C. Same as item 24A	24C. Same as item 24A	24C. None - Same as item 10D	
			24D. Mode 6	24D. Same as item 24A	24D. Same as item 24A	24D. None - Same as item 10E	
			24E. All	24E. Fails open with MDP P4-003 not running	24E. None	24E. None - Stop checks 043 and 046 prevent backflow, thus ensuring adequate AFW flow.	
25	Check valve 002 in MDP P4-002 discharge	Same as item 24, except pump is MDP P4-002	25A. Modes 1, 2, and 3	25A. Fails closed with MDP P4-002 running	25A. Position lights for miniflow valve FV-5154 on QMCB or PSDB show FV-5154 open	25A. None - Same as item 11A	See item 10.
			25B. Mode 4, SG 1 or 4	25B. Same as item 25A	25B. Same as item 25A	25B. None - Same as item 11C	
			25C. Mode 5	25C. Same as item 25A	25C. Same as item 25A	25C. None - Same as item 11D	
			25D. Mode 6	25D. Same as item 25A	25D. Same as item 25A	25D. Same as item 11E	
			25E. All	25E. Fails open with MDP P4-002 not running	25E. None	25E. None - Stop checks 037 and 040 prevent backflow, thus ensuring adequate AFW flow.	
26	Check valve 014 in TDP P4-001 discharge	Same as item 24, except pump is TDP P4-001	26A. Mode 3	26A. Fails closed with TDP P4-001 running	26A. None	26A. None - Same as item 12A	See item 12.
			26B. Mode 4, any SG	26B. Same as item 26A	26B. None	26B. None - Same as item 12B	

TABLE 10.4.9-4 (SHEET 13 OF 31)

Item No.	Description of Component[a]	Safety Function	Plant Operating Mode[b]	Failure Mode(s)	Method of Failure Detection[c]	Failure Effect on System Safety Function Capability	General Remarks
			26C. Mode 6	26C. Same as item 26A	26C. None	26C. None - Same as item 12D	
			26D. All	26D. Fails open with TDP P4-001 not running	26D. None	26D. None - Stop checks 017, 020, 023, and 026 prevent backflow, thus ensuring adequate AFW flow.	
27	AFW pump P4-003 miniflow valve FV-5155, normally open MOV, train A	Opens automatically or manually from PSDA to ensure MDP P4-003 miniflow requirements are satisfied at all times	27A. Modes 1, 2, 3, 4, and 6	27A1. Fails open with MDP P4-003 running	27A1. Position lights on QMCB or PSDA	27A1. None - Reduced flow available from MDP P4-003, but MDP P4-002 and TDP P4-001 are available.	See item 10 for effect of loss of one MDP for HELB at TDP discharge (mode 5).
				27A2. Fails closed with MDP P4-003 running	27A2. Same as item 27A1	27A2. None - Under some conditions MDP P4-003 may cavitate, but MDP P4-002 and TDP P4-001 are available.	
			27B. Mode 5	27B1. Same as item 27A1	27B1. Same as item 27A1	27B1. None - Reduced flow available from MDP P4-003, but MDP P4-002 satisfies 100% AFW requirements.	
				27B2. Same as item 27A2	27B2. Same as item 27A1	27B2. None - Same as item 27A2, except only MDP P4-002 is available; this is sufficient for this case.	
28	AFW P4-002 miniflow valve FV-5154, normally open MOV, train B	Same as item 27, except MDP is P4-002 and manual control from PSDB	28A. Modes 1, 2, 3, 4, and 6	28A1. Fails open with MDP P4-002 running	28A1. Position indicator on QMCB or PSDB	28A1. None - Reduced flow available from MDP P4-002, but MDP P4-003 and TDP P4-001 are available.	See items 27 and 10.

TABLE 10.4.9-4 (SHEET 14 OF 31)

Item No.	Description of Component[(a)]	Safety Function	Plant Operating Mode[(b)]	Failure Mode(s)	Method of Failure Detection[(c)]	Failure Effect on System Safety Function Capability	General Remarks
				28A2. Fails closed with MDP P4-002 running	28A2. Same as item 28A1	28A2. None - Under same conditions, MDP P4-002 may cavitate, but MDP P4-003 and TDP P4-001 are available.	
			28B. Mode 5	28B1. Same as item 28A1	28B1. Same as item 28A1	28B1. None - Reduced flow available from MDP P4-002, but MDP P4-003 satisfies 100% AFW requirements.	
				28B2. Same as item 28A2	28B2. Same as item 28A1	28B2. None - Same as item 28B1, except only MDP-003 available; this is sufficient for this case.	
29	AFW control valve HV-5139 from MDP P4-003 to SG 1, normally open MOV, train A	Regulates AFW flow from MDP P4-003 to SG 1 via remote manual control from QMCB and PSDA; opens automatically upon MDP automatic start signal	29A. Modes 1, 2, and 3	29A. Fails closed or fails to open upon command	29A. Position light on QMCB or position light on PSDA plus reduced flow indicators to SG 1; FI-5152A on QMCB, FI-5152B on PSDA, and FI-5152C on PAFP	29A. None - No flow to SG 1 from MDP P4-003, but MDP P4-002 and TDP P4-001 are available. Also, MDP P4-003 still supplies AFW SG 4.	For FMEA of MDP P4-003, see item 10. For system capability to withstand 30-min spillage. See item 10. HV-5122 can be closed to isolate TDP P4-001 from SG 1 for mode 4 (item 29B).
			29B. Mode 4, SG 1	29B. Fails open or fails to close upon command	29B. Same as item 29A	29B. None - MDP P4-003 can be turned off to terminate spillage. MDP P4-002 and TDP P4-001 supply 100% of flow.	
			29C. Mode 4, SG 4	29C. Fails closed or fails to open upon command	29C. Same as item 29A	29C. None - MDP P4-003 not effective, but MDP P4-002 available to feed SGs 2 and 3. Also,	

TABLE 10.4.9-4 (SHEET 15 OF 31)

Item No.	Description of Component[a]	Safety Function	Plant Operating Mode[b]	Failure Mode(s)	Method of Failure Detection[c]	Failure Effect on System Safety Function Capability	General Remarks
						TDP P4-001 available. MDP P4-003 can be turned off and HV-5120 can be closed to terminate AFW spillage from No. 4 feedline.	
			29D. Mode 5	29D. Fails closed or fails to open upon command	29D. Same as item 29A, plus zero flow indication to SG 1; FI-5152A on QMCB, FI-5152B on PSDA, and FI-5152C on PAFP	29D. None - MDP P4-003 still supplies SG 4 while MDP P4-002 supplies SGs 2 and 3.	
			29E. Mode 6 (LOCA)	29E. Fails closed or fails to open upon command	29E. Same as item 29A	29E. None - Same as item 29A	
			29F. Mode 6, MSLB in SG 1	29F. Fails open or fails to close upon command	29F. Same as item 29A	29F. None - Same as item 29B	
30	AFW control valve HV-5137 from MDP P4-003 to SG 4, normally open MOV, train A	Same as item 29, except SG 4	30A. Modes 1, 2, and 3	30A. Same as item 29A	30A. Same as item 29A, except flow indicators to SG 4 are FI-5150A on QMCB, FI-5150B on PSDA, and FI-5150C on PAFP	30A. None - Same as item 29A, except no flow to SG 4, but with flow to SG 1 from P4-003.	See items 29 and 10. HV-5120 can be closed to isolate TDP P4-001 from SG 4 for mode 4 (item 30B).
			30B. Mode 4, SG 4	30B. Same as item 29B	30B. Same as item 30A	30B. None - Same as item 29B	
			30C. Mode 4, SG 1	30C. Same as item 29C	30C. Same as item 30A	30C. None - Same as item 29C, except HV-5122 can be closed to terminate AFW spillage from No. 1 feedline.	
			30D. Mode 5	30D. Same as item 29D	30D. Same as item 30A, plus zero	30D. None - Same as item 29D, except	

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Item No.	Description of Component[(a)]	Safety Function	Plant Operating Mode[(b)]	Failure Mode(s)	Method of Failure Detection[(c)]	Failure Effect on System Safety Function Capability	General Remarks
					flow indication to SG 4; FI-5150A on QMCB, FI-5150B on PSDA, and 5150C on PAFP	MDP P4-003 still supplies SG 1.	
			30E. Mode 6 (LOCA)	30E. Same as item 29E	30E. Same as item 30A	30E. None - Same as item 30A	
			30F. Mode 6, MSLB in SG 4	30F. Same as item 29F	30F. Same as item 30A	30F. None - Same as item 29B	
31	AFW control valve HV-5134 from MDP P4-002 to SG 3, normally open MOV, train B	Same as item 29, except from MDP P4-002 to SG 3 and control from QMCB and PSDB	31A. Modes 1, 2, and 3	31A. Same as item 29A	31A. Same as item 29A, except flow indicators to SG 3 are FI-5153A on QMCB, FI-5153V on PSDB, and FI-5153C on PAFP	31A. None - No flow to SG 3 from MDP P4-002, but MDP P4-003 and TDP P4-001 are available. Also, MDP P4-002 still supplies AFW to SG 2.	See items 10 and 29. HV-5127 can be closed to isolate TDP P4-001 from SG 3 for mode 4 (item 31B).
			31B. Mode 4, SG 3	31B. Same as item 29B	31B. Same as item 31A	31B. None - MDP P4-002 can be turned off to terminate spillage. MDP P4-002 and TDP P4-001 supply 100% AFW flow.	
			31C. Mode 4, SG 2	31C. Same as item 29C	31C. Same as item 31A	31C. None - MDP P4-002 not effective, but MDP P4-003 is available to feed SGs 1 and 4. Also, TDP P4-001 is available. MDP P4-002 can be turned off and HV-5125 closed to terminate spillage.	
			31D. Mode 5	31D. Same as item 29D	31D. Same as item 31A, plus zero flow indication to SG 3; FI-5153A on QMCB, FI-5153B	31D. None - MDP P4-002 still supplies SG 2 while MDP P4-003 supplies SGs 1 and 4.	

TABLE 10.4.9-4 (SHEET 17 OF 31)

Item No.	Description of Component[(a)]	Safety Function	Plant Operating Mode[(b)]	Failure Mode(s)	Method of Failure Detection[(c)]	Failure Effect on System Safety Function Capability	General Remarks
					on PSDB, and FI-5153C on PAFP		
			31E. Mode 6 (LOCA)	31E. Same as item 29E	31E. Same as item 31A	31E. None - Same as item 31A	
			31F. Mode 6 (MSLB in SG 3)	31F. Same as item 29F	31F. Same as item 31A	31F. None - Same as item 31B	
32	AFW control valve HV-5132 from MDP P4-002 to SG 2, normally open MOV, train B	Same as item 31, except from MDP P4-002 to SG 2	32A. Modes 1, 2, and 3	32A. Same as item 29A	32A. Same as item 29A, except flow indicators to SG 2 are FI-5151A on QMCB, FI-5151B on PSDB, and FI-5151C on PAFP	32A. None - Same as item 31A, except no flow to SG 2, but with flow from MDP P4-002 to SG 3.	See items 10 and 29. HV-5125 can be closed to isolate TDP P4-001 from SG 2 for mode 4 (item 32B).
			32B. Mode 4, SG 2	32B. Same as item 29B	32B. Same as item 32A	32B. None - Same as item 31B	
			32C. Mode 4, SG 3	32C. Same as item 29C	32C. Same as item 32A	32C. None - Same as item 31C, except HV-5127 can be closed to terminate AFW spillage from No. 3 feedline.	
			32D. Mode 5	32D. Same as item 29D	32D. Same as item 32A, plus zero flow indication to SG 2; FI-5151A on QMCB, FI-5151B on PSDB, and FI-5151C on PAFP	32D. None - Same as item 31E, except MDP P4-002 still supplies SG 3.	
			32E. Mode 6 (LOCA)	32E. Same as item 29E	32E. Same as item 32A	32E. None - Same as item 32A	
			32F. Mode 6 (MSLB in SG 2)	32F. Same as item 29F	32F. Same as item 32A	32F. None - Same as item 31B	
33	AFW control valve HV-5127 from TDP P4-001 to SG 3, normally open dc-	Regulates AFW flow from TDP P4-001 to SG 3 via remote manual control from QMCB and PAFP; opens	33A. Mode 3	33A. Same as item 29A	33A. Same as item 29A, except flow indicators to SG 3 are FI-5153A	33A. None - No flow to SG 3 from TDP P4-001, but MDPs P4-002 and 003 are	See items 12 and 19. The faulted SG (mode 4, item 33C) can be

TABLE 10.4.9-4 (SHEET 18 OF 31)

Item No.	Description of Component[a]	Safety Function	Plant Operating Mode[b]	Failure Mode(s)	Method of Failure Detection[c]	Failure Effect on System Safety Function Capability	General Remarks
	powered MOV, train C	automatically upon TDP auto start signal			on QMCB, FI-5153B on PSDB, and FI-5153C on PAFP	available. Also, TDP P4-001 still supplies AFW to SGs 1, 2, and 4.	isolated from its respective MDP by closing the appropriate AFW control valve. Mode 5 is not applicable since break is assumed to be upstream of flow control valves.
			33B. Mode 4, SG 3	33B. Same as item 29B	33B. Same as item 33A	33B. None - TDP P4001 can be turned off and HV-5134 closed to terminate spillage. MPDs P4-002 and P4-003 provide 100% AFW flow.	
			33C. Mode 4, SG 1, 2, or 4	33C. Same as item 29C	33C. Same as item 33A	33C. None - TDP P4-001 still supplies two effective steam generators. Also, MDPs P4-002 and P4-003 available. (See remarks.)	
			33D. Mode 6 (LOCA)	33D. Same as item 29E	33D. Same as item 33A	33D. None - Same as item 33A	
			33E. Mode 6 (MSLB in SG 3)	33E. Same as item 29F	33E. Same as item 33A	33E. None - Same as item 33B	
34	AFW control valve HV-5125 from TDP P4-001 to SG 2, normally open, dc-powered MOV, train C	Same as item 33, except to SG 2	34A. Mode 3	34A. Same as item 29A	34A. Same as item 29A, except flow indicators to SG 2 are FI-5151A on QMCB, FI-5151B on PSDB, and FI-5151C on PAFP	34A. None - Same as item 33A, except no flow from TDP P4-001 to SG 2, but with flow to SGs 1, 3, and 4.	See items 12, 29, and 33.
			34B. Mode 4, SG 2	34B. Same as item 29B	34B. Same as item 34A	34B. None - Same as item 33B, except HV-5132 can be closed to isolate SG 2.	

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Item No.	Description of Component[a]	Safety Function	Plant Operating Mode[b]	Failure Mode(s)	Method of Failure Detection[c]	Failure Effect on System Safety Function Capability	General Remarks
			34C. Mode 4, SG 1, 3, or 4	34C. Same as item 29C	34C. Same as item 34A	34C. None - Same as item 33C (see remarks item 33)	
			34D. Mode 6 (LOCA)	34D. Same as item 29E	34D. Same as item 34A	34D. None - Same as item 34A	
			34E. Mode 6 (MSLB in SG 2)	34E. Same as item 29F	34E. Same as item 34A	34E. None - Same as item 34B	
35	AFW control valve HV-5122 from TDP P4-001 to SG 1, normally open dc-powered MOV, train C	Same as item 33, except to SG 1	35A. Mode 3	35A. Same as item 29A	35A. Same as item 29A, except flow indicators to SG 1 are FI-5152A on QMCB, FI-5152B on PSDA, and FI-5152C on PAFP	35A. None - Same as item 33A, except no flow from TDP P4-001 to SG 1, but flow to SGs 2, 3, and 4.	See items 12, 29, and 33.
			35B. Mode 4, SG 1	35B. Same as item 29B	35B. Same as item 35A	35B. None - Same as item 33B, except HV-5139 can be closed to isolate SG 1.	
			35C. Mode 4, SG 2, 3, or 4	35C. Same as item 29C	35C. Same as item 35A	35C. None - Same as item 33C (see remarks item 33)	
			35D. Mode 6 (LOCA)	35D. Same as item 29E	35D. Same as item 35A	35D. None - Same as item 35A	
			35E. Mode 6 (MSLB in SG 2)	35E. Same as item 29E	35E. Same as item 35A	35E. None - Same as item 35B	
36	AFW control valve HV-5120 from TDP P4-001 to SG 4, normally open dc-powered MOV, train C	Same as item 33, except to SG 4	36A. Mode 3	36A. Same as item 29F	36A. Same as item 29A, except flow indicators to SG 4 are FI-5150A on QMCB, FI-5150B on PSDA, and FI-5150C on PAFP	36A. None - Same as item 33A, except no flow from TDP P4-001 to SG 4, but with flow to SGs 1, 2, and 3	See items 12, 29, and 33.
			36B. Mode 4, SG 4	36B. Same as item 29B	36B. Same as item 36A	36B. None - Same as item 33B, except	

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Item No.	Description of Component[(a)]	Safety Function	Plant Operating Mode[(b)]	Failure Mode(s)	Method of Failure Detection[(c)]	Failure Effect on System Safety Function Capability	General Remarks
						HV-5137 can be closed to isolate SG 4.	
			36C. Mode 4, SG 2, 3, or 4	36C. Same as item 29C	36C. Same as item 36A	36C. None - Same as item 33C (see remarks item 33)	
			36D. Mode 6 (LOCA)	36D. Same as item 29E	36D Same as item 36A	36D. None - Same as item 36A	
			36E. Mode 6 (MSLB in SG 4)	36E. Same as item 29F	36E. Same as item 36A	36E. None - Same as item 36B	
37	Stop-check valve 046 in AFW supply from MDP P4-003 to SG 1	Prevents AFW backflow when MDP P4-003 not running and HV-5139 open	37A. Modes 1, 2, and 3	37A. Fails closed with MDP P4-003 running and HV-5139 open	37A. None	37A. None - Same as item 29A	
			37B. Mode 4, SG 4	37B. Same as item 37A	37B. None	37B. None - Same as item 29C	
			37C. Mode 5	37C. Same as item 37A	37C. Zero flow indication to SG 1 (see item 29D)	37C. None - Same as item 29D	
			37D. Mode 6 (LOCA)	37D. Same as item 37A	37D. None	37D. None - Same as item 29E	
			37E. All	37E. Fails open with MDP P4-003 not running and HV-5139 open	37E. None	37E. None - Other check valves in flow path prevent backflow. Also, HV-1539 can be closed.	
38	Stop-check valve 043 in AFW supply from MDP P4-003 to SG 4	Same as item 37, except HV is HV-5137	38A. Modes 1, 2, and 3	38A. Same as item 37A, except HV-5137 open	38A. None	38A. None - Same as item 30A	
			38B. Mode 4, SG 1	38B. Same as item 38A	38B. None	38B. None - Same as item 30C	
			38C. Mode 5	38C. Same as item 38A	38C. Zero flow indication to SG 4 (see item 30D)	38C. None - Same as item 30D	

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Item No.	Description of Component(a)	Safety Function	Plant Operating Mode(b)	Failure Mode(s)	Method of Failure Detection(c)	Failure Effect on System Safety Function Capability	General Remarks
			38D. Mode 6 (LOCA)	38D. Same as item 38A	38D. None	38D. None - Same as item 30E	
			38E. All	38E. Same as item 37E, except HV-5137 open	38E. None	38E. None - Same as item 37E, except HV-5137 can be closed.	
39	Stop-check valve 040 in AFW supply from MDP P4-002 to SG 3	Prevents AFW backflow when MDP P4-002 not running and HV-5134 open	39A. Modes 1, 2, and 3	39A. Fails closed with MDP P4-002 running and HV-5134 open	39A. None	39A. None - Same as item 31A	
			39B. Mode 4, SG 2	39B. Same as item 39A	39B. None	39B. None - Same as item 31C	
			39C. Mode 5	39C. Same as item 39A	39C. Zero flow indication to SG 3 (see item 31D)	39C. None - Same as item 31D	
			39D. Mode 6 (LOCA)	39D. Same as item 39A	39D. None	39D. None - Same as item 31E	
			39E. All	39E. Fails open with MDP P4-002 not running and HV-5134 open	39E. None	39E. None - Other check valves in flow path prevent backflow. Also, HV-5134 can be closed.	
40	Stop-check valve 037 in AFW supply from MDP P4-002 to SG 2	Same as item 39, except HV is HV-5132	40A. Modes 1, 2, and 3	40A. Same as item 39A, except HV-5132 open	40A. None	40A. None - Same as item 32A	
			40B. Mode 4, SG 3	40B. Same as item 40A	40B. None	40B. None - Same as item 32C	
			40C. Mode 5	40C. Same as item 40A	40C. Zero flow indication to SG 3 (see item 32D)	40C. None - Same as item 32D	
			40D. Mode 6 (LOCA)	40D. Same as item 40A	40D. None	40D. None - Same as item 32E	
			40E. All	40E. Same as item 39E, except HV-1532 open	40E. None	40E. None - Same as item 39E, except HV-5132 can be closed.	

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Item No.	Description of Component[(a)]	Safety Function	Plant Operating Mode[(b)]	Failure Mode(s)	Method of Failure Detection[(c)]	Failure Effect on System Safety Function Capability	General Remarks
41	Stop-check valve 026 in AFW supply from TDP P4-001 to SG 3	Prevents AFW backflow when TDP P4-001 not running and HV-5127 open	41A. Mode 3	41A. Fails closed with TDP P4-001 running and HV-5127 open	41A. None	41A. None - Same as item 33A	See items 12 and 33.
			41B. Mode 4, SG 1, 2, or 4	41B. Same as item 41A	41B. None	41B. None - Same as item 33C	
			41C. Mode 5	41C. Fails open with HV-5127 open	41C. None	41C. None - Same as item 12C. Also, HV-5127 can be closed.	
			41D. Mode 6 (LOCA)	41D. Same as item 41C	41D. None	41D. None - Same as item 33D	
			41E. All except 5	41E. Fails open with TDP P4-001 not running and HV-5127 open	41E. None	41E. None - Other check valves in flow path prevent backflow. Also, HV-5127 can be closed.	
42	Stop-check valve 023 in AFW supply from TDP P4-001 to SG 2	Same as item 41, except HV is HV-5125	42A. Mode 3	42A. Same as item 41A, except HV-5125 open	42A. None	42A. None - Same as item 34A	See items 12, 33, and 34.
			42B. Mode 4, SG 1, 3, or 4	42B. Same as item 42A	42B. None	42B. None - Same as item 34C	
			42C. Mode 5	42C. Fails open with HV-5125 open	42C. None	42C. None - Same as item 12C. Also, HV-5127 can be closed.	
			42D. Mode 6 (LOCA)	42D. Same as item 42A	42D. None	42D. None - Same as item 34D	
			42E. All except 5	42E. Same as item 41E, except HV-5125 open	42E. None	42E. None - Same as item 41E, except HV-5125 can be closed.	
43	Stop-check valve 020 in AFW supply from TDP P4-001 to SG 1	Same as item 41, except HV is HV-5122	43A. Mode 3	43A. Same as item 41A, except HV-5122 open	43A. None	43A. None - Same as item 35A	See items 12, 33, and 35

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Item No.	Description of Component[a]	Safety Function	Plant Operating Mode[b]	Failure Mode(s)	Method of Failure Detection[c]	Failure Effect on System Safety Function Capability	General Remarks
			43B. Mode 4, SG 2, 3, or 4	43B. Same as item 43A	43B. None	43B. None - Same as item 35C	
			43C. Mode 5	43C. Fails open with HV-5122 open	43C. None	43C. None - Same as item 12C. Also, HV-5122 can be closed.	
			43D. Mode 6 (LOCA)	43D. Same as item 43A	43D. None	43D. None - Same as item 35D	
			43E. All except 5	43E. Same as item 41E, except HV-5122 open	43E. None	43E. None - Same as item 41E, except HV-5122 can be closed.	
44	Stop-check valve 017 in AFW supply from TDP to SG 4	Same as item 41, except HV is HV-5120	44A. Mode 3	44A. Same as item 41A, except HV-5120 open	44A. None	44A. None - Same as item 36A	See items 12, 33, and 36.
			44B. Mode 4, SG 2, 3, or 4	44B. Same as item 41A	44B. None	44B. None - Same as item 36C	
			44C. Mode 5	44C. Fails open with HV-5120 open	44C. None	44C. None - Same as item 12C. Also, HV-5120 can be closed.	
			44D. Mode 6 (LOCA)	44D. Same as item 44A	44D. None	44D. None - Same as item 36D	
			44E. All except 5	44E. Same as item 41E except HV-5120 open	44E. None	44E. None - Same as item 41E, except HV-5120 can be closed.	
45	Stop-check valve 113 in AFW supply lines to SG 1	Prevents backflow from No. 1 feedwater bypass line whenever the AFW system not in operation	45A. Mode 7	45A. Fails open	45A. None	45A. None - Other check valves in flow path prevent backflow.	
			45B. Modes 1 through 6	45B. Fails closed	45B. Zero flow indication to SG 1; FI-5152A on QMCB, FI-5152B on PSDA, and FI-5152C on PAFP	45B. None - Same as item 29D, plus TDP P4-001 available to supply AFW to SGs 2, 3, and 4.	

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Item No.	Description of Component[(a)]	Safety Function	Plant Operating Mode[(b)]	Failure Mode(s)	Method of Failure Detection[(c)]	Failure Effect on System Safety Function Capability	General Remarks
46	Stop-check valve 114 in AFW supply line to SG 2	Same as item 45, except for No. 2 feedwater bypass line	46A. Mode 7	46A. Fails open	46A. None	46A. None - Same as item 45A	
			46B. Modes 1 through 6	46B. Fails closed	46B. Zero flow indication to SG 2; FI-5151A on QMCB, FI-5151B on PSDB, and FI-5151C on PAFP	46B. None - Same as item 32D, plus TDP P4-001 is available to supply AFW to SGs 1, 3, and 4.	
47	Stop-check valve 115 in AFW supply line to SG 3	Same as item 45, except for No. 3 feedwater bypass line	47A. Mode 7	47A. Fails open	47A. None	47A. None - Same as item 45A	
			47B. Modes 1 through 6	47B. Fails closed	47B. Zero flow indication to SG 3; FI-5153A on QMCB, FI-5153B on PSDB, and FI-5153C on PAFP	47B. None - Same as item 31D, plus TDP P4-001 is available to supply AFW to SGs 1, 2, and 4.	
48	Stop-check valve 116 in AFW supply line to SG 4	Same as item 45, except for No. 4 feedwater bypass line	48A. Mode 7	48A. Fails open	48A. None	48A. None - Same as item 45A	
			48B. Modes 1 through 6	48B. Fails closed	48B. Zero flow indication to SG 4; FI-5150A on QMCB, FI-5150B on PSDA, and FI-5150C on PAFP	48B. None - Same as item 30D, plus TDP P4-001 is available to supply AFW to SGs 1, 2, and 3.	
49	Feedwater bypass isolation valve HV-15196, normally open, fail closed air-operated valve	Isolates feedwater bypass line to SG 1 upon feedwater isolation signal; admits main feedwater to AFW nozzle during power generation	49A. Modes 1 through 6	49A. Fails open or fails to close upon command	49A. Position lights on QMCB	49A. None - Check valve 053 prevents backflow or injection of AFW into main feedwater nozzle.	This valve is normally open only during power generation.
			49B. Mode 7	49B. Fails closed or fails to open upon command	49B. Position lights on QMCB	49B. None - AFW path not affected. All main feedwater enters via main feedwater nozzle.	

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Item No.	Description of Component[a]	Safety Function	Plant Operating Mode[b]	Failure Mode(s)	Method of Failure Detection[c]	Failure Effect on System Safety Function Capability	General Remarks
50	Feedwater bypass isolation valve HV-15197, normally open, fail closed air-operated valve	Same as item 49, except for SG 2	50A. Modes 1 through 6	50A. Same as item 49A	50A. Same as item 49A	50A. None - Same as item 49A, except check valve is 118.	See item 49.
			50B. Mode 7	50B. Same as item 49B	50B. Same as item 49B	50B. None - Same as item 49B	
51	Feedwater bypass isolation valve HV-15198, normally open, fail closed air-operated valve	Same as item 49, except for SG 3	51A. Modes 1 through 6	51A. Same as item 49A	51A. Same as item 49A	51A. None - Same as item 49A, except check valve is 120.	See item 49.
			51B. Mode 7	51B. Same as item 49B	51B. Same as item 49B	51B. None - Same as item 49B	
52	Feedwater bypass isolation valve HV-15199, normally open, fail closed air-operated valve	Same as item 49, except for SG 4	52A. Modes 1 through 6	52A. Same as item 49A	52A. Same as item 49A	52A. None - Same as item 49A, except check valve is 119.	See item 49.
			52B. Mode 7	52B. Same as item 49B	52B. Same as item 49B	52B. None - Same as item 49B	
53	Feedwater bypass check valve 117	Prevents backflow of SG 1 AFW flow into main feedwater system whenever latter not operating	53A. Modes 1 through 6	53A. Fails open	53A. None	53A. None - Condensate storage facility sized to accommodate loss due to 30-min backflow. Also, HV-15196 can be closed.	See item 10.
			53B. Mode 7	53B. Fails closed	53B. None	53B. None - Same as item 49B	
54	Feedwater bypass check valve 118	Same as item 53, except for SG 2	54A. Modes 1 through 6	54A. Fails open	54A. None	54A. None - Same as item 53A, except HV is HV-15197.	See item 10.
			54B. Mode 7	54B. Fails closed	54B. None	54B. None - Same as item 49B	

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Item No.	Description of Component[(a)]	Safety Function	Plant Operating Mode[(b)]	Failure Mode(s)	Method of Failure Detection[(c)]	Failure Effect on System Safety Function Capability	General Remarks
55	Feedwater bypass check valve 120	Same as item 53, except for SG 3	55A. Modes 1 through 6	55A. Fails open	55A. None	55A. None - Same as item 53A, except HV is HV-15198.	See item 10.
			55B. Mode 7	55B. Fails closed	55B. None	55B. None - Same as item 49B	
56	Feedwater bypass check valve 119	Same as item 53, except for SG 4	56A. Modes 1 through 6	56A. Fails open	56A. None	56A. None - Same as item 53A, except HV is HV-15199.	See item 10.
			56B. Mode 7	56B. Fails closed	56B. None	56B. None - Same as item 49B	
57	Not used						
58	AFW line check valve 125	Prevents blowdown at No. 1 steam generator or auxiliary building overpressurization in the event of a HELBA in the No. 1 steam generator auxiliary feedwater line outside containment.	58A. All but AFW HELB	58A. Fails closed	58A. Same as for item 45B except "None" for Mode 7	58A. None - Same as item 45B	
			58B. AFW HELB	58B. Fails open	58B. None	58B. None - Check valve 117, 113, and 133 provides redundancy.	
59	Not Used						
60	AFW line check valve 126	Same as item 58 except for steam generator No. 2 and for control building overpressure protection.	60A. All but AFW HELB	60A. Fails closed	60A. Same as for item 46B except "None" for Mode 7	60A. None - Same as item 46B	
			60B. AFW HELBA	60B. Fails open	60B. None	60B. None - Check valve 118, 114, 134 provides redundancy.	
61	Not used						
62	AFW line check valve 128	Same as item 58 except for steam generator No. 3 and for control building overpressure protection.	62A. All but AFW HELB	62A. Fails closed	62A. Same as for item 47B except "None" for Mode 7	62A. None - Same as item 47B	

TABLE 10.4.9-4 (SHEET 27 OF 31)

Item No.	Description of Component[(a)]	Safety Function	Plant Operating Mode[(b)]	Failure Mode(s)	Method of Failure Detection[(c)]	Failure Effect on System Safety Function Capability	General Remarks
			62B. AFW HELB	62B. Fails open	62B. None	62B. None - Check valve 120, 116, 136 provides redundancy.	
63	Not used						
64	AFW line check valve 127	Same as item 58 except for steam generator No. 4.	64A. All but AFW HELB	64A. Fails closed	64A. Same as for item 48B except "None" for Mode 7	64A. None - Same as item 48B	
			64B. AFW HELB	64B. Fails open	64B. None	64B. None - Check valve 119, 116, 135 provides redundancy.	
65	TDP steam line drain valve HV-5178, normally closed MOV, not train aligned	Opens manually from QMCB or automatically to drain excessive amounts of condensate from TDP steam supply line	Modes 3, 4, 5, and 6	65A. Fails open	65A. Position light on QMCB	65A. None - Possible degraded TDP P4-001 performance, but MDPs are available.	
				65B. Fails closed	65B. Position light on QMCB	65B. None - Condensate will drain via FO-15133. Also, turbine accepts water slugs.	
66	Not used						
67	System components powered from essential ac bus: HV-5119 (item 7), HV-5118 (item 8), MDP P4-003 (item 10), and MDP P4-002 (item 11)	Refer to items 7, 8, 9, and 10 as indicated	Mode 8	Both MDPs fail to operate and the HVs remain in position (normally closed)	Status lights indicate loss of all ac power, plus all pump valve indicators remain dark	None - TDP P4-001 (item 12) provides 100% of AFW requirements. All valves, controls, etc., associated with the TDPs are dc powered from emergency battery supply.	See also items 9, 12, 13, 14, and 15 for discussion of valves associated with operation of TDP P4-100.
68	Chemical injection system isolation valve HV-5194 (normally closed, fail closed air-operated valve, non-1E)	Provides isolation of SG 1 from chemical injection system by remote-manual control from QPCP	All	68A. Fails closed or fails to open upon command	68A. Position light on QPCP	68A. None – SG chemistry control not required for safe shutdown	

TABLE 10.4.9-4 (SHEET 28 OF 31)

Item No.	Description of Component[a]	Safety Function	Plant Operating Mode[b]	Failure Mode(s)	Method of Failure Detection[c]	Failure Effect on System Safety Function Capability	General Remarks
				68B. Fails open or fails to close upon command	68B. Same as item 68A	68B. None - Check valve 133 provides isolation.	
69	Chemical injection system isolation valve HV-5195 (normally closed, fail closed air-operated valve, non-1E)	Same as item 68 except for SG 2	All	69A. Fails closed or fails to open upon command	69A. Position light on QPCP	69A. None - Same as item 68A	
				69B. Fails open or fails to close upon command	69B. Same as item 69A	69B. None - Check valve 134 provides isolation	
70	Chemical injection system isolation valve HV-5196 (normally closed, fail closed air-operated valve, non-1E)	Same as item 68 except for SG 3	All	70A. Fails closed or fails to open upon command	70A. Position light on QPCP	70A. None - Same as item 68A	
				70B. Fails open or fails to close upon command	70B. Same as item 70A	70B. None - Check valve 136 provides isolation.	
71	Chemical injection system isolation valve HV-5197 (normally closed, fail closed air-operated valve, non-1E)	Same as item 68 except for SG 4	All	71A. Fails closed or fails to open upon command	71A. Position light on QPCP	71A. None - Same as item 68A	
				71B. Fails open or fails to close upon command	71B. Same as item 71A	71B. None - Check valve 135 provides isolation.	
72	Check valve 133 in chemical injection line to SG 1	Prevents loss of AFW through failure of the chemical injection system	All	72A. Fails closed	72A. None	72A. None - Same as item 68A	
				72B. Fails open	72B. None	72B. None - HV-5122 and HV-5139 can be closed to prevent loss of AFW inventory.	

TABLE 10.4.9-4 (SHEET 29 OF 31)

Item No.	Description of Component(a)	Safety Function	Plant Operating Mode(b)	Failure Mode(s)	Method of Failure Detection(c)	Failure Effect on System Safety Function Capability	General Remarks
73	Check valve 134 in chemical injection line to SG 2	Same as item 72	All	73A. Fails closed	73A. None	73A. None - Same as item 68A	
				73B. Fails open	73B. None	73B. None - HV-5125 and HV-5132 can be closed to prevent loss of AFW inventory.	
74	Check valve 136 in chemical injection line to SG 3	Same as item 72	All	74A. Fails closed	74A. None	74A. None - Same as item 68A	
				74B. Fails open	74B. None	74B. None - HV-5127 and HV-5134 can be closed to prevent loss of AFW inventory.	
75	Check valve 135 in chemical injection line to SG 4	Same as item 72	All	75A. Fails closed	75A. None	75A. None - Same as item 68A	
				75B. Fails open	75B. None	75B. None - HV-5120 and HV-5137 can be closed to prevent loss of AFW inventory.	
76	TDP steam line drain valve HV-5179 (normally closed MOV not train aligned)	Opens manually from QMCB or automatically drain excessive amounts of condensate from top steam supply line	Modes 3, 4, 5, and 6	76A. Fails open	76A. Position light on QMCB	76A. None - Possible degraded TDP P4-001 performance, but MDP available.	
				76B. Fails closed	76B. Position light on QMCB	76B. None - Condensate will drain via FO-15141. Also, turbine accepts water slugs.	
77	Relief valve PSV-5110 in TDP/P-4-001	Prevents over pressure and damage to pump suction line and line instrumentation if	77A. All but mode 5	77A. PSV fails to open with pump out of service	77A. None	77A. None - Potential suction line/instrumentation	

TABLE 10.4.9-4 (SHEET 30 OF 31)

Item No.	Description of Component[a]	Safety Function	Plant Operating Mode[b]	Failure Mode(s)	Method of Failure Detection[c]	Failure Effect on System Safety Function Capability	General Remarks
	train "C" suction line 010-10	pump suction check valves 013 & 051 hold tight & pump discharge check valve 014 leaks even TDP is not in service				damage: remaining two MDP's, train "B" & "A" each available, either satisfies 100% flow requirement.	
			77B. All but mode 5	77B. Spurious PSV opening with TDP out of service	77B. None	77B. None	
			77C. All but mode 7	77C. Spurious PSV opening with TDP in service	77C. None	77C. None - TDP still delivers flow requirement, since flow thru PSV is insignificant.	
78	Relief valve PSV-5128 in MDP/P-4-002 train "B" suction line 017-8	Prevents over pressure and damage to pump suction line & instrumentation if pump suction check valves 050 & 061 hold tight & pump discharge check valve 002 leaks when MDP is not in service	78A. All modes	78A. Same as item 77A above with pump-002	78A. Same as item 77A above	78A. None - Potential suction line/ instrumentation damage; remaining MDP & TDP available; MDP provides 100%, TDP provides 200% flow requirement	
			78B. All modes	78B. Same as item 77B above with pump-002	78B. Same as item 77B above	78B. Same as item 77B above	
			78C. All but mode 7	78C. Same as item 77C above with pump-002	78C. Same as item 77C above	78C. Same as item 77C above	
79.	Relief valve PSV-5129 in MDP/P-4-002 train "A" suction line 016-8	Prevents over pressure and damage to pump suction line & line instrumentation if pump suction check valve 033 & 052 hold tight & pump discharge check valve 001 leaks when MDP is not in service	79A. All modes	79A. Same as item 77A above with pump-003	79A. Same as item 77A above	79A. None - potential suction line/instrumentation damage remaining MDP & TDP available. MDP provides 100%, TDP pump provides 200% flow requirement	
			79B. All modes	79B. Same as item 77B above with pump-003	79B. Same as item 77B above	79B. Same as item 77B above	

TABLE 10.4.9-4 (SHEET 31 OF 31)

Item No.	Description of Component[(a)]	Safety Function	Plant Operating Mode[(b)]	Failure Mode(s)	Method of Failure Detection[(c)]	Failure Effect on System Safety Function Capability	General Remarks
			79C. All but mode 7	79C. Same as item 77C above with pump-003	79C. Same as item 77C above	79C. Same as item 77C above	

a. P&IDs IX4DB159-2, rev 15; 1X4DB161-1, rev 15; 1X4DB161-2, rev 15; 1X4DB161-3, rev 13; 1X4DB168-3, rev 15. Logic Diagrams 1X5DN108-1, rev 4, 1X5DN109-1, rev 2; 1X5DN109-2, rev 2; 1X5DN117-1, rev 4; 1X5DN120-1, rev 2; 1X5DN120-2, rev 3; 1X5DN120-3, rev 1; 1X5DN120-4, rev 3; 1X5DN120-5, rev 3; 1X5DN120-6, rev 2; 1X5DN121-1, rev 3; 1X5DN121-2, rev 3; 1X5DN122-1, rev 3; 1X5DN122-2, rev 3.

b. Mode 1, plant startup; mode 2, plant cooldown; mode 3, loss of normal feedwater coincident with loss of offsite power (LOP); mode 4, auxiliary feedline rupture coincident with LOP; mode 5, HELB in TDP discharge coincident with LOP; mode 6, safety injection coincident with LOP; mode 7, power generation; and mode 8, loss of all ac power.

c. In general, valve and other indicators will illuminate either on the main control board or on the shutdown panel, but not on both simultaneously, depending on the position of the transfer switch.

TABLE 10.4.9-5 (SHEET 1 OF 2)

AUXILIARY FEEDWATER SYSTEM SUMMARY OF INDICATION AND CONTROLS

INDICATION

Parameter	Main Control Board	Shutdown Panel	Turbine-Driven Pump AFW Panel
Steam generator water level	X	X	X
Steam generator pressure	X	X	
AFW motor-driven pump suction pressure	X	X	
AFW motor-driven pump discharge pressure	X	X	
Turbine-driven pump steam inlet pressure	X		X
Turbine-driven pump differential steam pressure	X		X
Auxiliary feedwater flowrate	X	X	X
AFW isolation/control valve position	X	X	X
Turbine-driven pump steam inlet valve position	X		X
CST level	X	X	X
Motor-driven pump miniflow valve position	X	X	
CST motor operated outlet valve position	X	X	X
AFW pump motor running	X	X	
CST level alarms	X		X
AFW pump discharge pressure low alarm	X		
Control transferred to local panel alarm	X		

TABLE 10.4.9-5 (SHEET 2 OF 2)

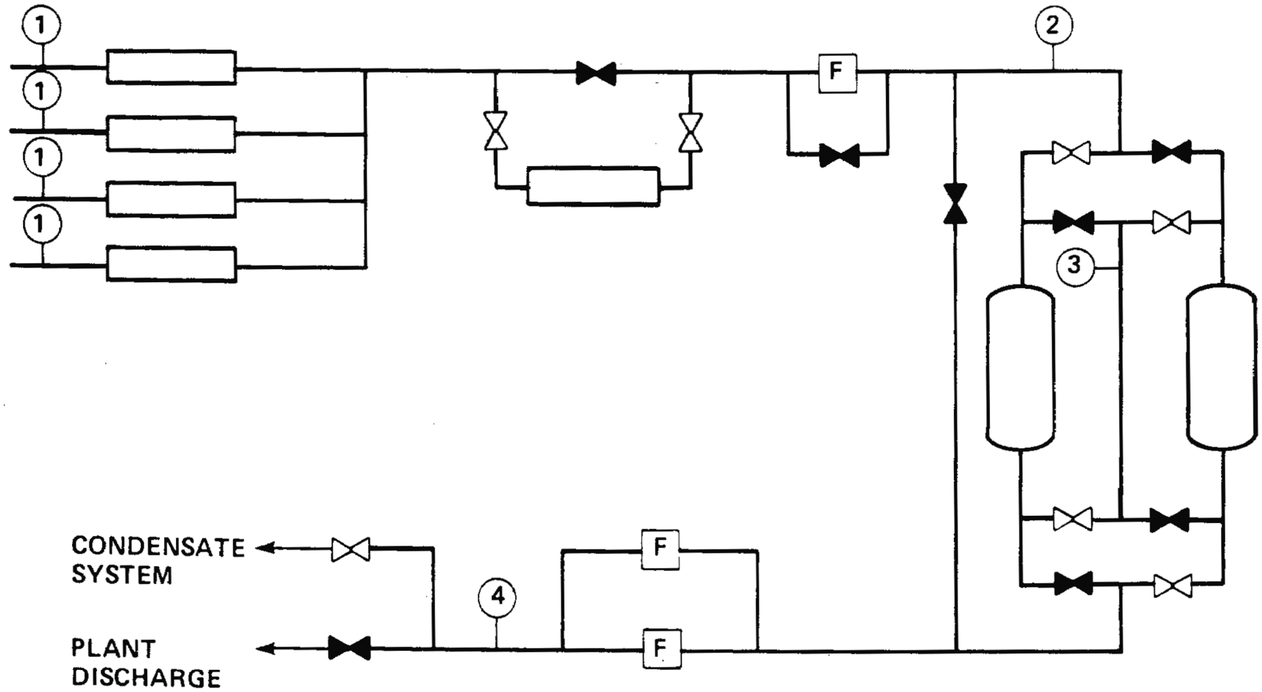
CONTROLS

Parameter	Main Control Board	Shutdown Panel	Turbine-Driven Pump AFW Panel
Motor-driven pumps	X	X	
Turbine-driven pump	X		X
AFW isolation/control valves	X	X	X
Turbine-driven pump speed control	X		X

TABLE 10.4.10-1

CONDENSATE AND FEEDWATER CHEMICAL
INJECTION SYSTEM

Major Component	Design Parameters
Methoxypropylamine (MPA) storage tank	12,789 gal
Hydrazine storage tank	6644 gal
Day tanks	250 gal
Metering cylinders	10 liter
MPA storage transfer pump	43 gal/min at 178 ft head
Hydrazine storage transfer pump	43 gal/min at 110 ft head
Condensate chemical pump	Hydrazine - 11.7 gal/h at 650 psig MPA - 0.5 gal/h at 650 psig
Steam generator layup chemical pump	100 gal/h at 1200 psig
Auxiliary feedwater layup chemical pump	100 gal/h at 1200 psig
Hydrazine dispensing pump/high volume	3 gal/min at 24 ft head
Day tank mixing pump	10 gal/min at 20 ft head
Deleted	



LOCATION		TEMP. (°F)	PRESSURE (PSIG)	FLOW. (GPM)(1)	ISOTOPIC CONC.
1	INPUT FROM STEAM GENERATOR	545	927-1075(5)	90	(2)
2	INPUT TO DEMINERALIZERS	130	<200	360	(2)
3	OUTLET FROM 1st DEMINERALIZER	130	<200	360	(3)
4	DISCHARGE FROM SGBPS	130	<200	360	(4)

NOTES:
1. DESIGN MAXIMUM FLOW AT 60°F
2. ISOTOPIC CONCENTRATIONS AS LISTED ON TABLE 11.1-7
3. CONCENTRATIONS OF CESIUM AND RUBIDIUM ISOTOPES ARE 0.1 OF THE VALUES LISTED ON TABLE 11.1-7. CONCENTRATIONS OF OTHER ISOTOPES ARE 0.01 OF THE VALUES LISTED ON TABLE 11.1-7.
4. CONCENTRATIONS OF CESIUM AND RUBIDIUM ISOTOPES ARE 0.01 OF THE VALUES LISTED ON TABLE 11.1-7. CONCENTRATIONS OF OTHER ISOTOPES ARE 0.001 OF THE VALUES LISTED ON TABLE 11.1-7.
5. ZERO-TO 100-PERCENT STEAM GENERATOR LOAD.

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STEAM GENERATOR BLOWDOWN PROCESSING SYSTEM PROCESS FLOW DIAGRAM

FIGURE 10.4.8–1

APPENDIX 10A

VEGP AUXILIARY FEEDWATER SYSTEM RELIABILITY ANALYSIS

10A.1 INTRODUCTION

This appendix documents the results of the reliability analysis discussed below. This analysis was performed in accordance with Item II.E.1.1 of NUREG-0737 at the time of initial licensing. Since that time, the analysis has not been updated to reflect current plant configuration. Therefore, this appendix is retained for historical information only.

10A.1.1 STATEMENT OF PURPOSE

The purpose of this appendix is to address the conformance of the VEGP auxiliary feedwater system (AFWS) with the Standard Review Plan's recommendation that a reliability analysis of the AFWS be performed.

10A.1.2 BACKGROUND AND OBJECTIVES

The Three Mile Island Unit 2 (TMI-2) accident occurred on March 28, 1979. The TMI-2 accident and subsequent investigations and studies highlighted the importance of the AFWS in the mitigation of transients and accidents. Prompted by the TMI-2 accident, the staff of the Nuclear Regulatory Commission (NRC) evaluated the AFWS for all operating plants having nuclear steam supply systems designed by Westinghouse (NUREG-0611, January 1980[1]). The NRC subsequently mandated in its March 10, 1980, letter to all pending operating license applicants of Westinghouse and Combustion Engineering nuclear steam supply systems that a comparable reliability evaluation of the AFWS be performed for each licensee. The requirements outlined in that letter have since been integrated into the Standard Review Plan.[2] Consistent with the recommendations of the Standard Review Plan, this analysis has the following objectives:

A. Outline the details of the AFWS and present the information required to perform a reliability evaluation.

B. Perform a reliability evaluation of the AFWS which corresponds with the outline recommended by the NRC in Appendix III and Annex 1 of Appendix X in NUREG-0611 and Item II.E.1.1-1 in NUREG-0737[3] (section 10A.3).

C. Identify and address the important contributors to the AFWS unreliability (subsection 10A.4.2).

D. Develop a comparative basis for the AFWS reliability study versus the generic AFWS reliability study which has been performed by the NRC staff (subsection 10A.4.3).

10A.2 AFWS DESCRIPTION

10A.2.1 SAFETY-RELATED FUNCTION

The safety-related function of the AFWS is to maintain water inventory in the steam generators for reactor residual heat removal when the main feedwater system is unavailable. The AFWS in conjunction with the steam generators and the main steam line atmospheric relief and/or safety valves, is used to cool the reactor coolant system to 350°F, at which point the residual heat removal system is used to further cool the reactor coolant system. The AFWS may also be used to temporarily hold the plant in a hot standby condition while main feedwater flow is restored, with the option of cooling the reactor coolant system to the residual heat removal initiation temperature.

10A.2.2 SYSTEM CONFIGURATION

The AFWS for a given unit, shown schematically in figures 10A-1 and 10A-2, provides auxiliary feedwater from the condensate storage tanks (CSTs) (two per unit) to the four steam generators by means of mechanical trains A, B, and C.

Each train initially takes suction from a CST which is maintained above the minimum level of 330,000 gal, with the capability of manual switchover to the other CST. The minimum water level of a CST is designed to maintain the reactor in a hot standby condition for 4 h followed by a 5-h cooldown period, at which time the residual heat removal system can be used to further cool the reactor coolant system. The combined minimum operating capacity of the CSTs (660,000 gal) is designed to allow a hot standby condition for 31 h followed by a 5-h cooldown period until operation of the residual heat removal system is initiated.

From the CSTs, the auxiliary feedwater passes through the respective pump of each mechanical train and exits through a discharge header into the steam generators. The motor-driven pumps of mechanical trains A and B each provide more than 100 percent of the required auxiliary feedwater flow, with mechanical train A capable of providing feedwater to steam generators 1 and 4 and mechanical train B capable of providing feedwater to steam generators 2 and 3. The steam-driven turbine pump of mechanical train C is capable of providing 200 percent of the minimum required auxiliary feedwater flow for cooldown of the reactor. Therefore, any one of the AFWS mechanical trains operating will provide sufficient feedwater to two effective steam generators to allow cooldown of the reactor.

With a few exceptions, all valves in the AFWS flowpath from the CSTs to the steam generators are normally open. The exceptions to this are the motor-operated valves located between the normal standby CST and the suction of each auxiliary feedwater pump. These valves are opened from the control room when the water from the normal standby CST is required.

Though normally open, the motor-operated valves in the discharge of the AFWS receive an actuation signal to open automatically (5120, 5122, 5125, 5127, 5132, 5134, 5137, and 5139). Also, these valves are used by the operator to control auxiliary feedwater flow to the steam generators.

The turbine-driven pump of mechanical train C is powered by steam tapped from the main steam lines of steam generators 1 and 2. The motor-operated valves in each of these lines (3009 and 3019, respectively) are normally open. The steam from either steam generator is sufficient to drive the turbine-driven pump.

All three mechanical trains of the AFWS have miniflow lines on the discharge side of the pumps which allow a minimum flow through the pumps during their start. Additionally, these miniflow lines are used for inservice testing of the pumps. The flow, once passing through the pumps, is returned to the CSTs. The miniflow lines of mechanical trains A and B each have a motor-operated valve (5155 and 5154, respectively) which closes automatically after flow has been established through each respective pump. The miniflow lines of mechanical trains A and B also have flow-limiting orifices in each line which limit the flow to 175 gal/min/line. The miniflow line of the turbine-driven pump has flow through it whenever the turbine-driven pump is operating. A flow-limiting orifice in the turbine-driven pump miniflow line limits the flow to 160 gal/min.

On the discharge side of the motor-driven pumps there is a crossover tieline which has been added to allow versatility in the operation of the AFWS. The two valves in this line are normally locked closed (055 and 056); therefore, this crossover tieline is exempted from normal use. However, for emergency use, the operator may use the crossover tieline to maintain the water level in all four steam generators with only one motor-driven pump if an extended use of the AFWS is required.

For flushing of the AFWS, an auxiliary feedwater recirculation line draws water from the AFWS header of each steam generator and returns it to the condensate system.

Each recirculation line is fitted with an orifice that limits the amount of flow diverted away from the steam generators.

When the AFWS is not being flushed, the recirculation valves (81, 82, 83, and 84) must remain closed.

10A.2.3 INSTRUMENTATION AND CONTROLS

10A.2.3.1 <u>Automatic Controls</u>

The AFWS is aligned to be placed in service automatically in the event of a demand. Following the receipt of a safety injection signal, a two-out-of-four low-low steam generator water level signal from any one steam generator, a trip signal from both main feedwater pumps, or a loss of offsite power signal, the auxiliary feedwater discharge valves go to the full-open position if not already open and the two motor-driven auxiliary feedwater pumps are actuated and begin to deliver flow from the online CST to the steam generators. Once flow has been established, the motor-operated valves in the miniflow lines close automatically. The turbine-driven pump is actuated automatically on two-out-of-four low-low water level in any two steam generators or on a loss of offsite power signal. To actuate the turbine-driven pump, a normally closed dc motor-operated valve (5106) in the steam supply line to the turbine is opened automatically. The speed governing valve and the trip/throttle valve, which are in the same line as the steam inlet valve, are automatically controlled by the speed governor on the turbine-driven pump. Following a transient or accident, the minimum flow is delivered to at least two effective steam generators within 1 min of an automatic auxiliary feedwater actuation signal. Once actuated, the operator can remotely manipulate the position of the auxiliary feedwater control valves in order to control steam generator water level.

10A.2.3.2 Manual Controls

For normal operation, the AFWS is used to fill and/or maintain the water level in the steam generators during startup, shutdown, and hot standby conditions. The AFWS may be actuated and controlled manually during normal operation or abnormal conditions. The components designated in table 10A-1 can be remotely controlled from the control room.

10A.2.3.3 Information Available to Operator

The parameters which the control room operators can use to judge the status of the AFWS are designated in table 10A-1.

10A.2.4 SUPPORT SYSTEMS

10A.2.4.1 Electric Power

The components dependent on electric power in mechanical trains A and B receive their electric power from the Class 1E switchgear which have diesel generators in reserve, should offsite power be lost. The diesel generators are designed to actuate on a safety injection signal or a loss of ac power signal. The loss of ac power signal is generated by an undervoltage condition on the 4.16-kV bus of train A or B. The safety injection signal is generated by low steam pressure, low pressurizer pressure, high containment pressure, or manual actuation. The components of mechanical trains A and B are powered from the corresponding emergency ac electric power trains A and B, respectively. Motor-operated valves 3019 and 3009 of mechanical train C are powered from electrical trains A and B, respectively. The electrically powered components of mechanical train C, with the exception of motor-operated valves 3009 and 3019, are all powered from dc electrical train C, which is backed up by the station batteries. The batteries for each electrical train are sized for a LOCA/LOSP duration of 2 3/4 h and a station blackout (SBO) duration of 4 h to provide continuous dc power without recharging. The electrical power sources for the components of the AFWS are listed in table 10A-2. As indicated in this table, the power to the controls of each component is the same as the component itself.

10A.2.4.2 Heating and Cooling

The heating and cooling systems are designed to maintain the temperature in the pump rooms between 40°F and 120°F. The fans that regulate the temperature in each of the motor-driven pump rooms are powered from the Class 1E onsite electrical system. In the pump rooms there are heaters powered from the non-Class 1E onsite electrical system. The turbine-driven pump room is provided with a nonsafety grade fan that is powered by a nonemergency electrical bus. The heaters and TDAFW pump room fan are intended for nonemergency use of the AFWS. For emergency use of the turbine-driven pump, the turbine-driven pump room is designed to be cooled by natural ventilation.

Independent cooling is provided for each of the AFWS pumps. Each motor-driven pump has air-cooled motors and water-cooled pumps. The turbine-driven pump uses water to cool the pump and the lube oil. The three pumps are located in separate rooms in the AFWS pumphouse, with each room cooled by its own ventilation system.

The stuffing boxes of each of the pumps are cooled by water tapped from the respective pump so that cooling water is provided if a pump is running. The lube oil of the turbine-driven pump is cooled by water drawn from the discharge of the turbine-driven pump.

10A.2.4.3 Water Sources

There are two water sources to the AFWS. They are the two CSTs. Each tank is a Seismic Category 1 structure and has a capacity of 480,000 gal. A total of 330,000 gal from one tank is sufficient to maintain the reactor in the hot standby mode for 4 h followed by a 5-h cooldown period. The minimum safety capacity is ensured by all nozzles of nonsafety systems being located on the storage tanks above the corresponding elevation. The condensate level in each tank is automatically maintained by a level control valve in the line (to the tank) from the demineralized water system, which actuates when the volume in the tank drops to 472,250 gal.

As the water in the online CST is depleted, the operator may manually realign the system so that the standby CST serves all three pumps. A separate line connects each pump to each CST.

10A.2.4.4 Main Steam System

The main steam system provides the steam supply to the AFWS turbine-driven pump. The steam to the turbine is provided from steam generators 1 and 2, either of which are sufficient for operation of the AFWS turbine-driven pump. Each of the steam supply lines to the turbine is equipped with a check valve, a locked open gate valve, and a normally open dc motor-operated gate valve. These steam supply lines join to form a header which leads to the turbine through a normally closed supply valve and normally open trip/throttle valve, both of which are dc motor-operated, and a normally open electro-hydraulically operated speed governing valve.

The main steam system is not the only supply of steam to the AFWS turbine-driven pump. A backup source is provided by a connection on the turbine steam supply header from an auxiliary steam system boiler inter-tie between units. This connection is normally isolated from the turbine by a manually operated normally closed gate valve.

10A.2.5 TESTING, MAINTENANCE, AND PLANT PROCEDURES

As of the date of this evaluation, the Technical Specifications, operating procedures, maintenance procedures, and testing procedures applicable to the VEGP AFWS have not been written.

Thus, in order to model and analyze the contribution of human error, testing and maintenance to the reliability of the VEGP AFWS relevant generic documents were used.

The Technical Specifications used were extracted from the Westinghouse Standard Technical Specifications.(4) The most notable factors of these preliminary Technical Specifications are:

A. With one AFWS pump inoperable, the limiting condition of operation action time to hot standby is 78 h.

B. With two AFWS pumps inoperable, the limiting condition of operation action time to hot standby is 6 h.

C. The testing frequency for AFWS pumps is once per 31 days.

D. The verification frequency of valves in the flowpath is once per 31 days.

E. The testing frequency of pumps and valves with automatic actuation is performed once per 18 months.

F. With one or more steam generators inoperable, the limiting condition of operation action time is 1 h.

G. With less than 330,000 gal in the CSTs, the limiting condition for operation action time to hot shutdown is 16 h.

H. The verification frequency of the CSTs water volume is once per 12 h.

I. With one 125-V dc train inoperable, the limiting condition for operation action time to hot standby is 2 h.

J. The testing frequency of each dc train is once per 7 days.

The generic plant operating procedures were assumed in the VEGP AFWS reliability evaluation. The most notable factors of these generic operating procedures are:

A. As part of the reactor trip procedures, the operator is to verify the proper automatic actuation of the AFWS (with at least one AFWS pump feeding to two steam generators).

B. As part of the station blackout procedures, the operator is to verify the proper automatic actuation of the AFWS (with at least one AFWS pump feeding to two steam generators). Station blackout coping is discussed in section 8.4.

C. As part of the station blackout procedures, the operator is to immediately align AFWS flowpaths to all steam generators to minimize the possibility of emptying any steam generator. Station blackout coping is discussed in section 8.4.

D. As part of the station blackout procedures, the operator is to maintain the steam generator level by manual control of the AFWS. Station blackout coping is discussed in section 8.4.

E. As part of the loss of secondary coolant procedures, the operator is to verify the proper automatic actuation of the AFWS.

F. As part of the loss of secondary coolant procedures, the operator is to isolate the AFWS flow to the affected steam generator to protect it from undesirable thermal transient.

G. As part of any procedure, manual actuation of the AFWS by the operator will be a situation the operator is readily familiar with.

H. As part of any procedure, manual actuation of the AFWS by the operator can be accomplished from the control room.

The generic plant testing and maintenance procedures used in the AFWS reliability evaluation were a synthesis of generic procedures. These generic procedures are based on current industry practice, lessons learned from previous human reliability analysis, and the VEGP AFWS design capabilities. The design capabilities of the AFWS allow testing while the plant is operating without affecting main feedwater flow. The alignment of any train of the AFWS for testing is such that suction is taken from a CST and the flow passes through the pump through the miniflow line, and back to the CST. Closure of the motor-operated valves in the discharge lines prevents flow to the steam generators during AFWS testing.

A. Human events for redundant trains as modeled are noncoupled events.

B. The motor-operated valves from CST 002 (5113, 5118, and 5119) are manually controlled with no automatic signals to close (if CST 002 is being used for testing or flushing of an AFWS train).

C. The motor-operated valves in the miniflow lines of trains A and B (5154 and 5155) are subject to maintenance for calibration of the flow element actuation device in these valves.

D. The motor-operated valves in the discharge lines (5120, 5122, 5125, 5127, 5132, 5134, and 5137) are used to manually throttle AFWS flow and pressure during testing to keep AFWS flow from entering a steam generator.

E. The performance of maintenance on a component requires the component be manually isolated on both the upstream and downstream sides.

F. The motor-operated valves in the discharge lines receive an automatic actuation signal to go to their full-open position even if they are being used for testing.

10A.2.6 PHYSICAL SEPARATION

Physical separation between the trains of the AFWS is maintained in regard to the prevention of common cause failures created by fire, flooding, and missiles. A simplified piping layout schematic of the AFWS is provided in figure 10A-3. Excluding the containment building, there are only two locations where a portion of all three trains lie in a common area. The first is in the building that houses the CSTs and the second is in a pipe chase in the auxiliary feedwater pumphouse. Both of these locations:

A. Are protected from external missiles and have no internal source for missiles.

B. Have no components subject to disabling damage due to flooding.

C. Have minimal sources of fire.

Physical separation between electrical components of the AFWS is provided in accordance with Regulatory Guide 1.75 and Institute of Electrical and Electronics Engineers (IEEE) Standard 384.

10A.3 <u>RELIABILITY EVALUATION</u>

10A.3.1 METHODOLOGY

As recommended in NUREG-0800, the reliability evaluation of the AFWS was performed utilizing the methodology in Appendix III and Annex I of Appendix X in NUREG-0611. This subsection provides the specific VEGP AFWS methodology utilized in the development and analysis of the reliability model. The outline of the methodology is presented as a flow chart in figure 10A-4. A discussion of the flow chart follows.

10A.3.1.1 <u>Development of System Bounds</u>

To perform the reliability analysis, it was necessary to define the system boundaries. The AFWS was defined to be bounded by:

A. The CSTs (i.e., begins at).

B. The inlet to the steam generators (i.e., ends at).

C. The Class 1E electric power system immediate to the components of the AFWS (i.e., begins at).

D. The steam lines that lead from the main steam system to the turbine-driven pump (i.e., begins at).

E. The cooling and ventilation system immediate to the AFWS (i.e., begins at).

10A.3.1.2 Development of the Reliability Block Diagram

In accordance with NUREG-0611, fault tree analysis was utilized to model and quantify the reliability of the AFWS. (The methodology of fault tree analysis is not explained herein; references 5, 6, and 7 are recommended for the unfamiliar reader.) To develop the appropriate fault tree logic, a reliability block diagram (block diagram for short) was developed. The block diagram developed is shown in figure 10A-5. A block diagram was developed, as an intermediate step because it essentially depicts the relationship between groups of components (for a fault tree analyst, the blocks may be thought of as representing supercomponents) and because it represents the logical relationship of components in a format similar to the system diagram. The components within each block of figure 10A-5 are listed in table 10A-3. From the original block diagram, an expanded block diagram was developed. This second block diagram, shown in figure 10A-6, was developed to emphasize the component groupings which can be unavailable due to maintenance or which can fail randomly. As is evident, the steam generator blocks are allowed to only fail randomly because maintenance upon the components represented by these blocks is not allowed while the plant is operating.

10A.3.1.3 Review of the Generic AFWS Analysis Performed by the NRC

A review of the generic AFWS analysis (NUREG-0611) was performed to ensure the assumptions and boundary conditions used in the AFWS reliability analysis are consistent with similar studies so as not to influence the results disproportionably.

The review of the generic AFWS reliability analysis revealed the generic analysis consisted of:

A. Quantifying a point estimate of the probability of failure on demand of the AFWS.

B. Internal plant events (not including fire).

C. A qualitative evaluation of common cause.

D. A time frame equivalent to the time it takes a steam generator to boil dry with no makeup available.

E. An evaluation of the AFWS as it pertains to three accident sequences. The accident sequences are:

   1. Case 1 - Loss of main feedwater with a reactor trip occurring and offsite ac power available (referred to as LMFW).

   2. Case 2 - Loss of offsite ac power causing main feedwater to be lost and the reactor to trip (referred to as LMFW/LOOP).

   3. Case 3 - Loss of main feedwater and only dc electrical power available (referred to as LMFW/LOAC).

The aforementioned potential accident sequences and time frame designations are actually boundary conditions of the reliability evaluation. But because they have a major impact on the quantification of the AFWS reliability, they are discussed herein. The major impact these two designations have is that they change the evaluation from the misnomered system reliability to a system conditional unavailability. The reasons that the two designations change the evaluation from a reliability evaluation to a conditional unavailability are:

A. The three accident sequences place the plant in an unconventional state, and as such there is a probability associated with the plant being placed in any of the three states. Also, there is an associated probability of the plant recovering from any of the three states; thus, the result is conditional to these probabilities.

B. The designated time frame does not encompass the amount of time the AFWS would be expected to operate, i.e., the mission time. Because of this, the calculations (quantification of the fault tree) are toward calculating the system availability or unavailability. (System unavailability is defined as the probability of the system being unavailable due to testing or maintenance or the probability of the system failing on demand.)

The effect of three accident sequences places the unavailabilities into a conditional state. This means that the unavailabilities are important only so far as the likelihood (or probability) of the plant being placed in any of these states. The effect of the quantification of a system unavailability (versus reliability) is that the result is an indication of system performance and is not complete. To emphasize this point, the quantification of the fault tree will be referred to as a conditional unavailability.

10A.3.1.4 Development of the AFWS Fault Tree Model to Component Failure Causes

One of the objectives of the AFWS reliability analysis was to develop a logic model which could be quantified. Toward this end, the basic tool utilized was a fault tree model. The fault tree model was developed as a logical progression from the reliability block diagram. Also, the fault tree was synthesized from the defined boundary conditions and necessary assumptions including those from the NRC's generic analysis.

The initial fault tree was developed to the component failure mode level and then expanded to the component failure cause level. The component failure causes considered were:

A. Random failure on demand.

B. Unavailability due to maintenance.

The fault tree developed for the analysis is shown in figure 10A-7.

10A.3.1.5 Minimal Cut Set Analysis

Once developed, the fault tree was analyzed for determining the minimal cut sets using the WAM-CUT computer code.[8] The list of minimal cut sets was then analyzed for their viability in reference to plant Technical Specifications. Also, the possibility of operator action (within the constricting time frame) was evaluated.

10A.3.1.6 Quantification

The final version of the fault tree was quantified for each of the three cases. The quantification process included applying a statistically independent probability to each basic event in the fault tree for each of the three cases. The three cases were evaluated to determine a point estimate of the conditional unavailability of the VEGP AFWS given:

A. Case 1 - Loss of main feedwater with a reactor trip and electric power available (LMFW).

B. Case 2 - Loss of offsite ac power causing a loss of main feedwater with a reactor trip and onsite electric power available (LMFW/LOOP).

C. Case 3 - Loss of main feedwater with a reactor trip and station blackout so that only dc electric power is available (LMFW/LOAC).

10A.3.1.7 Deterministic Common Cause Analysis

To assess the reliability of the VEGP AFWS against common cause failures, a common cause analysis was performed. This analysis was performed deterministically and in two parts. The first part was performed explicitly for common cause hardware failure by location. The analysis is discussed in subsection 10A.2.6. The second part of the common cause analysis was performed implicitly throughout the evaluation. That is, any potential failure of a support system (such as electrical power) resulting in the failure of supposedly redundant components, was accounted for in the logic fault tree. The results of the entire common cause analysis revealed no significant common cause potential within the VEGP AFWS.

10A.3.1.8 Derivation of Analysis Conclusions

Following the completion of the evaluation, conclusions were derived drawn on:.

A. The quantified conditional unavailability.

B. The dominant contributors to the conditional unavailability.

C. A comparison of the VEGP AFWS reliability with those AFWS evaluations already performed as reported in NUREG-0611.

10A.3.2 SYSTEM SUCCESS CRITERIA

The AFWS is composed of three mechanical trains which serve the four steam generators at a given unit. The steam generators have been analyzed to require 510 gal/min of flow under the most severe accident conditions. Each motor-driven pump of mechanical trains A and B has a capacity of 630 gal/min and provides more than 100 percent of the required auxiliary feedwater flow. Mechanical train A provides feedwater to steam generators 1 and 4, and mechanical train B provides feedwater to steam generators 2 and 3. The (steam) turbine-driven pump of mechanical train C has a capacity of 1300 gal/min and provides more than 200 percent of the required auxiliary feedwater flow. The turbine-driven pump provides feedwater to all four steam generators. The success criterion for the AFWS is flow to any two steam generators. Furthermore, as outlined by the NRC evaluation of generic AFWSs (NUREG-0611), the AFWS must actuate within the time it takes for the steam generators to boil dry when no flow is provided to the steam generators. At VEGP, the boiloff time (and therefore the limit on the AFWS actuation time) is approximately 30 min.

10A.3.3 COMPONENT FAILURE DATA

The component failure data (data for short) used in quantifying the fault trees were taken, when possible, from NUREG-0611, which was derived in part from WASH-1400. When the appropriate data were not available from NUREG-0611, WASH-1400 data were used. Table 10A-4 is a listing of the data used in the VEGP AFWS reliability evaluation. All data were used to quantify point estimates of unavailability on demand, and uncertainty is not accounted for in the analysis. It should be noted that the data utilized in the reliability analysis is generic, and as such the results are an evaluation of the AFWS design. The implication of the data is that they do not account for the actual characteristics of how the plant is to be operated and maintained.

It should be noted that the formula used to calculate maintenance unavailability in NUREG-0611 was believed to be far too conservative for estimating valve maintenance unavailability. Instead a formula from NUREG-0492 was adapted. The NUREG-0611 formula appears to assume maintenance is performed every 4.5 months. Possibly this is routine maintenance. However, it was believed that maintenance on valves would be performed only when the valve fails. Therefore the formula used to estimate valve unavailability due to maintenance was:

$$\lambda_s T_R$$

where:

λ_s = valve standby failure rate.

T_R = average repair or maintenance time for valve.

10A.3.4 ANALYSIS BOUNDARY CONDITIONS AND ASSUMPTIONS

In order to properly model the AFWS, certain assumptions had to be made; in order to be consistent with the NRC's generic evaluation of AFWSs, the scope of the analysis had to be altered accordingly. The following is a list of the boundary conditions and assumptions which were used to model the conditional unavailability of the AFWS.

A. The AFWS is modeled for its safety function and not for its use during normal operation.

B. The AFWS can only be configured in modes consistent with the preliminary Technical Specifications and the preliminary plant procedures.

C. It is assumed that steam is available to the main steam lines that lead to the turbine of the turbine-driven pump.

D. The crossover tieline between trains A and B is not modeled because the use of this line falls outside of the 30-min limit on operator action.

E. The operator is allowed 30 min to actuate the AFWS from within the control room. This is based on the conservative assumption that it will take 30 min for the steam generators to boil dry after loss of main feedwater.

F. The loss of actuation of the AFWS is modeled as a basic event.

G. Sample lines are not modeled.

H. The loss of dc electric power is modeled as a basic event.

I. The loss of electric power to components of different voltages but powered from the same electrical train is modeled as the same basic event.

J. Common cause failures such as missiles, fires, floods, etc., are not modeled.

K. Pipe ruptures are not modeled (because of their negligible probability).

L. A single unit of VEGP is modeled and quantified.

10A.4 QUANTITATIVE FINDINGS

10A.4.1 CONDITIONAL UNAVAILABILITIES

The quantitative results of the conditional unavailabilities for the three cases designated by the NRC for the AFWS are:

A. Case 1 - LMFW

For the case where there is an assumed loss of main feedwater with a reactor trip occurring and offsite ac power available, the conditional unavailability of the AFWS was calculated to be 6.3 x 10^{-6}.

B. Case 2 - LMFW/LOOP

For the case where there is an assumed loss of main feedwater with a reactor trip occurring and offsite ac power not available, the conditional unavailability of the AFWS was calculated to be 2.6 x 10-5.

C. Case 3 - LMFW/LOAC

For the case where there is an assumed loss of main feedwater with a reactor trip occurring and no ac power available, the conditional unavailability of the AFWS was calculated to be 1.0 x 10-2.

10A.4.2 DOMINANT CONTRIBUTORS

The quantitative measure of importance was used as an indication of the dominant contributors to the AFWS conditional unavailability. The value of importance was taken as the sum of all cut set probabilities containing a component failure divided by the top event probability. The failure categories analyzed for each case are: random failure of valves on demand; unavailability of valves due to maintenance; operator error; and pump unavailabilities (random or maintenance).

A. Case 1 - LMFW

The most significant contributor to system failure was pump unavailabilities. The importance value to pump unavailabilities was calculated to be 86 percent. An examination of the category of pump unavailabilities revealed that pump failures were occurring in combination with electric power system failure.

Furthermore, it was determined that the unavailability of the turbine-driven pump was the most significant single component of the AFWS, but this pump did not dominate system unavailability. This finding is then an indication that due to redundancy of the AFWS, no single component of the AFWS can be thought of as dominating (or controlling) system unavailability.

B. Case 2 - LMFW/LOOP

The findings for case 2 revealed pump unavailabilities contribute 80 percent to system unavailability. An examination of this category revealed, as did case 1, no single component of the AFWS can be thought of as dominating (or controlling) system unavailability. The reduction of the system conditional availability for this case was found to be directly attributable to the assumed loss of redundancy in ac power sources.

C. Case 3 - LMFW/LOAC

The findings for case 3 revealed (under assumed conditions) that the AFWS is reduced to only the turbine-driven pump. Thus, any single failure along this pump train would be sufficient to fail the AFWS. The dominant contributors to system unavailability were as follows:

1. The turbine-driven pump package (pump, trip throttle valve, and speed governing valve).
2. The steam inlet valve (motor-operated valve 5106).

10A.4.3 COMPARATIVE STUDIES RESULTS

The availability of the AFWS for various plants, including VEGP, is compared in table 10A-5. This table is derived from NUREG-0611. For the case of LMFW, the AFWS compares very favorably, ranking in the very high category. For the case of LMFW/LOOP, the AFWS compares favorably, ranking in the high category. For the case of LMFW/LOAC, the AFWS ranks in the medium category.

10A.4.4 REFERENCES

1. U.S. Nuclear Regulatory Commission, "Generic Evaluation of Feedwater Transients and Small-Break Loss-of-Coolant Accidents in Westinghouse-Designed Operating Plants," NUREG-0611, Bulletins and Orders Task Force, Office of Nuclear Reactor Regulation, January 1980.
2. U.S. Nuclear Regulatory Commission, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants - LWR Edition," NUREG-0800, Revision 2, Office of Nuclear Reactor Regulation, July 1981.
3. U.S. Nuclear Regulatory Commission, "Clarification of TMI Action Plan Requirements," NUREG-0737, Office of Nuclear Reactor Regulation, Division of Licensing, November 1980.
4. U.S. Nuclear Regulatory Commission, "Standard Technical Specifications for Westinghouse Pressurized Water Reactors," NUREG-0452, Revision 4, Office of Nuclear Reactor Regulation, Fall 1981.
5. Hickman, J. W., et al., "PRA Procedures Guide: A Guide to the Performance of Probabilistic Risk Assessments for Nuclear Power Plants," NUREG/CR-2300, Vol. I and II, January 1983.
6. Haasl, D. F., "Advanced Concepts in Fault Tree Analysis," Presented at the Boeing Company Safety Symposium, Seattle, Washington, June 8-9, 1965.
7. Vesely, W. E., et al., "Fault Tree Handbook," NUREG-0492, January 1981.

8. Erdmann, R. C., Leverenz, F. L., and Kirch, H., "WAM-CUT: A Computer Code for Fault Tree Evaluation," EPRI-NP-803, Science Applications, Inc., June 1978.

10A.4.5 BIBLIOGRAPHY

Kirch, H., and Leverenz, F. L., "User's Guide for the WAM-BAM Computer Code - Key Phase Report," EPRI Research Project 217-2-5, Science Applications, Inc., January 1976.

Ross, D. F., Jr., Acting Director of U.S. Nuclear Regulatory Commission, "Actions Required from Operating License Applicants of Nuclear Steam Supply Systems Designed by Westinghouse and Combustion Engineering Resulting from the NRC Bulletins and Orders Task Force Review Regarding the Three Mile Island Unit 2 Accident," Letter of March 10, 1980.

Westinghouse Electric Corporation, Westinghouse Owners Group- Emergency Response Guideline Seminar, Vol I, II, and III, 1981.

TABLE 10A-1 (SHEET 1 OF 2)

AFWS CONTROL CAPABILITIES[a]

Component	Automatic Actuation	Indication CR	Indication Local	Remote/Manual Controls CR	Remote/Manual Controls Local	Alarms/Light CR	Alarms/Light Local
Motor-operated valves from CST 002 (5113, 5118, and 5119)	N	N	N	Y	Y	L(P)	L(P)
Motor-operated valves on miniflow line (5154 and 5155)	Y(F)[b]	N	Y(F)	N	Y	L(P)	L(P)
Motor-operated valves on discharge line from motor-driven pumps (5132, 5134, 5137, and 5139)	Y(AFS)	Y(P)	Y(P)	Y	Y	L(P)	L(P)
Motor-driven pumps (002 and 003)	Y(AFS)	Y(DP) Y(SP)	Y(DP) Y(SP)	Y	Y	L(P) A(DP)	L(P) A(DP)
Pump room drain sump	N	N	N	N	N	L(P) A(WL)	L(P) N
CSTs (001 and 002)	Y(WL)	Y(WL)	Y(WL)	Y	Y	L(P)[d] A(WL)	L(P)[d] A(WL)
Steam generators (1, 2, 3, and 4)	NA	Y(WL) Y(T)	Y(WL) Y(PR) Y(T)	NA Y(PR)	NA	A(T)	A(T) A(WL) A(PR)
Steam generators intake from AFWS (each of four lines)	NA	Y(F)	Y(F)	NA	NA	N	N
Motor-operated valves on turbine-driven pump discharge lines (5120, 5122, 5125, and 5127)	Y(AFS)	Y(P)	Y(P)	Y	Y	L(P)	L(P)
Turbine-driven pump (001)	Y(AFS) Y(SV)[f]	Y(DP) Y(SP) Y(P) Y(S)	Y(DP) Y(SP) Y(P) Y(MF) Y(ST)	Y	N	A(DP)	A(DP)
Turbine-steam intake motor-operated valve (5106)	Y(AFS)	N	N	Y	Y	L(P)	L(P)
Turbine-steam generator intake motor-operated valves (3009 and 3019)	N	N	N	Y	Y	L(P)	L(P)
Turbine steam intake	NA	Y(PR)	Y(PR) Y(T)	NA	NA	N	N
Turbine speed governing valve	Y(SG)	Y(S)	Y(S)	Y[c]	Y[c]	A(OS)	N
Turbine trip and throttle valve	Y(SG)	Y(S)	Y(S)	Y[c]	Y[c]	L(P)	L(P)

TABLE 10A-1 (SHEET 2 OF 2)

Component	Automatic Actuation	Indication		Remote/Manual Controls		Alarms/Light	
		CR	Local	CR	Local	CR	Local
Turbine lube oil cooler							
Lube oil inlet	NA	N	Y(T)	NA	NA	N	N
Lube oil return	NA	N	Y(PR)	NA	NA	N	N

a. Table explanation:

CR = Location inside the main control room.
Local = Location anywhere in the plant outside of the main control room.
Alarm/Light = A bipositional annunciator.
Indication = A continuous variable display.

N = No.
Y( ) = Yes (variable).
L( ) = Annunciating light (variable).
A( ) = Annunciating alarm (variable).
NA = Not applicable.

Variables:

F = Flow.
P = Position.
MF = Miniflow.
ST = Shaft temperature.
DP = Discharge pressure.
SP = Suction pressure.
ΔP = Discharge pressure between pump discharge pressure and turbine inlet pressure.
AFS = AFWS actuation signal.
PR = Pressure.
T = Temperature.
S = Speed.
WL = Water level.
OS = Overspeed.
SG = Speed governor.
SV = Turbine speed governing valve.
TV = Turbine trip and throttle valve.

b. Flow on pump discharge line.

c. Controls are not on valve per se, but are on the speed governor.

d. Position of motor-operated valve on tank fill line.

e. Low-low discharge pressure

f. See component 14.

TABLE 10A-2

ELECTRIC POWER SOURCE LIST FOR THE AFWS[(a)]

Component Description	Component Number	125-V dc Voltage Level/ Electrical Train	Normal Position[(b)]	AFWS Actuation Signal Train
Motor-driven pump controls	002	4.16 kV/B	S	-
		125-V dc/B	S	B
Motor-driven pump controls	003	4.16 kV/A	S	-
		125-V dc/A	S	A
Motor-driven pump miniflow motor-operated valves	5154	480 V/B	NO	-
	5155	480 V/A	NO	-
Motor-driven pump intake motor-operated valves	5118	480 V/B	NC	-
	5119	480 V/A	NC	-
Motor-driven pump discharge motor-operated valves	5132	480 V/B	NO	B
	5134	480 V/B	NO	B
	5137	480 V/A	NO	A
	5139	480 V/A	NO	A
Turbine-driven pump intake motor-operated valve	5113	125-V dc/C	NC	-
Turbine-driven pump discharge motor-operated valves	5120	125-V dc/C	NO	A, B[(c)]
	5122	125-V dc/C	NO	A, B[(c)]
	5125	125-V dc/C	NO	A, B[(c)]
	5127	125-V dc/C	NO	A, B[(c)]
Turbine-steam inlet motor-operated valve	5106	125-V dc/C	NC	A, B[(c)]
Turbine-steam generators inlet motor-operated valves	3009	125-V dc/B	NO	-
	3019	125-V dc/A	NO	-

a. Abbreviations:
S = Standby.
NO = Normally open.
NC = Normally closed.

NOTE: Component control power is from the same voltage level as component unless designated otherwise; 125-V dc/C derives power from ac electric power train A with backup power provided by batteries.

b. Normal position is during plant operating mode of power generation.
c. Actuation signal from train A or B.

TABLE 10A-3

COMPONENT GROUPINGS IN AFWS BLOCK DIAGRAM
(Corresponds to Figure 10A-5)

Block	Components in Block
SG1	Check valve 121, check valve 125, check valve 113
SG2	Check valve 122, check valve 126, check valve 114
SG3	Check valve 124, check valve 128, check valve 115
SG4	Check valve 123, check valve 127, check valve 116
PMPA	Gate valve 035, check valve 001, motor-driven pump 003, butterfly valve 5095, butterfly valve 5092, motor-operated valve 5119, butterfly valve 5099
PMPB	Gate valve 060, check valve 002, motor-driven pump 002, butterfly valve 5094, butterfly valve 5091, motor-operated valve 5118, butterfly valve 5098
PMPC	Check valve 014, turbine-driven pump 001, butterfly valve 5093, butterfly valve 5090, motor-operated valve 5113, butterfly valve 5097, motor-operated valve 5106, trip and throttle valve, speed governing valve, motor-operated valve 3009, motor-operated valve 3019, check valve 008, check valve 006, gate valve 007, gate valve 005
A1	Check valve 046, motor-operated valve 5139, gate valve 045
A4	Check valve 043, motor-operated valve 5137, gate valve 042
B2	Check valve 037, motor-operated valve 5132, gate valve 036
B3	Check valve 040, motor-operated valve 5134, gate valve 039
C1	Check valve 020, motor-operated valve 5122, gate valve 019
C2	Check valve 023, motor-operated valve 5125, gate valve 022
C3	Check valve 026, motor-operated valve 5127, gate valve 025
C4	Check valve 017, motor-operated valve 5120, gate valve 016

TABLE 10A-4 (SHEET 1 OF 3)

AFWS COMPONENT FAILURE DATA

Fault Event/Tree Description	Component	Failure on Demand	Reference[a]	Repair Time (h)	Unavailability Due to Maintenance[b]	Reference[a]
Check valve (at steam generator intake) fails to open on demand	121, 122, 123, 124, 125, 126, 127, 128	1×10^{-4}	1	NA	NA	NA
Stop check valve (at steam generator intake) fails to open on demand	113, 114, 115, 116	1×10^{-4}	1	NA	NA	NA
Stop check valve (on AFWS discharge) fails to open on demand	017, 020, 023, 026, 037, 040, 043, 046	1×10^{-4}	1	7	2.17×10^{-6}	1, 3
Motor-operated valve (on discharge line) transfers closed	5120, 5122, 5125, 5127, 5132, 5134 5137, 5139	1×10^{-4}	1	7	2.17×10^{-6}	1, 3
Gate valve (on discharge line) transfers closed	015, 016, 019, 022, 025, 035, 036, 039, 042, 045, 060	1×10^{-4}	1	7	7×10^{-8}	1, 3
Check valve (on discharge line) fails to open on demand	001, 002, 014	1×10^{-4}	1	7	2.17×10^{-8}	1, 3
Motor-driven pump fails (includes controls)	003, 002	5×10^{-3}	1	19	5.81×10^{-3}	1
Turbine-driven pump fails (includes controls)	001	5×10^{-3}	1	19	5.81×10^{-3}	1

TABLE 10A-4 (SHEET 2 OF 3)

Fault Event/Tree Description	Component	Failure on Demand	Reference[(a)]	Repair Time (h)	Unavailability Due to Maintenance[(b)]	Reference[(a)]
Motor-operated valve (on turbine intake) fails on demand	5106	3.1×10^{-3}	1	7	2.17×10^{-6}	1
Check valves (on turbine steam intake) fail to open on demand	006, 008	1×10^{-4}	1	7	2.17×10^{-6}	1, 3
Motor-operated valves (on turbine steam intake) transfer closed on demand	3009, 3019	1×10^{-4}	1	7	2.17×10^{-6}	1, 3
Gate valve (on turbine steam intake) transfers closed on demand	005, 007	1×10^{-4}	1	NA	NA	NA
Butterfly valve (on suction line) transfers closed	093, 094, 095	1×10^{-4}	1	7	7.0×10^{-8}	1, 3
Motor-operated valve (pump suction line) fails on demand	5113, 5118, 5119	3.1×10^{-3}	1	7	2.17×10^{-6}	1, 3
Butterfly valve (on CST discharge line) transfers closed	090, 091 092, 097, 098, 099	1×10^{-4}	1	40	4×10^{-7}	2, 3
CST fails	001, 002	1×10^{-8}	3	NA	NA	NA
Failure of actuation signal	Train A, train B, speed governor	7×10^{-3}	1	NA	NA	NA
Loss of offsite power	Case 1	0.2	3	NA	NA	NA
Failure of 125-V dc electric power	Train A, train B, train C	2.4×10^{-6}	NA	2	NA	3

TABLE 10A-4 (SHEET 3 OF 3)

Fault Event/Tree Description	Component	Failure on Demand	Reference[a]	Repair Time (h)	Unavailability Due to Maintenance[b]	Reference[a]
Failure of ac electric power (onsite - case 1 and 2)	Train A, train B	3×10^{-2}	3	NA	NA	NA
Motor-operated valve closed by error	3009, 3019, 5120, 5122, 5125, 5127, 5132, 5134, 5137, 5139	5×10^{-4}	1	NA	NA	NA
No manual open signal to motor-operated valve	3009, 3019, 5106, 5113, 5118, 5119, 5120, 5122, 5125, 5132, 5134, 5137, 5139	5×10^{-3}	1	NA	NA	NA
No manual start signal to pump	001, 002, 003, speed governor	5×10^{-3}	1	NA	NA	NA
Trip and throttle valve or speed governing valve fails to open on demand	Trip and throttle valve, speed governing valve	1.1×10^{-3}	1	7	2.17×10^{-6}	3

a. References

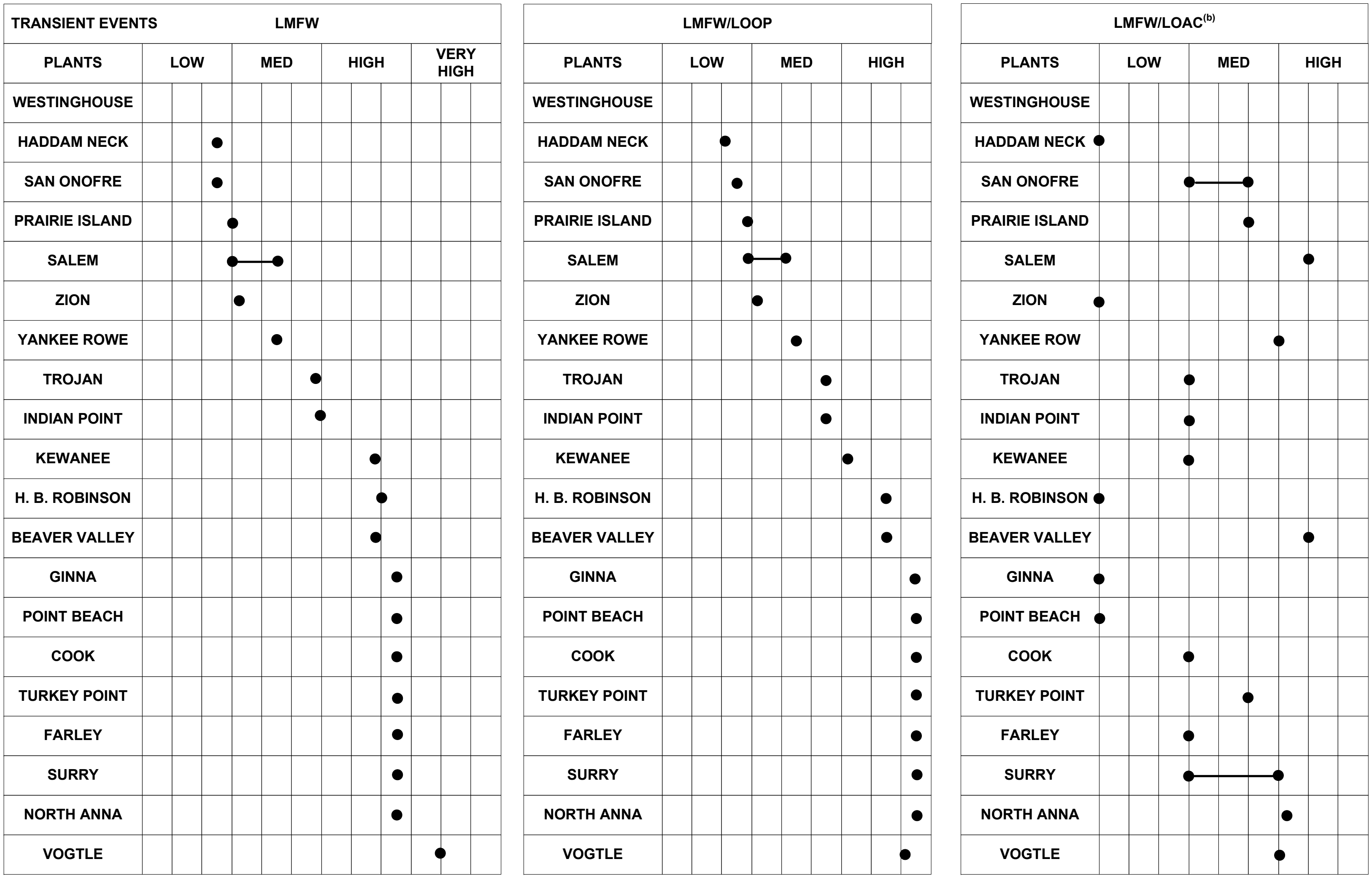
1. U.S. Nuclear Regulatory Commission, "Generic Evaluation of Feedwater Transients and Small-Break Loss-of-Coolant Accidents in Westinghouse-Designed Operating Plants," NUREG-0611, Bulletins and Orders Task Force, Office of Nuclear Reactor Regulation, January 1980.

2. Engineering judgment.

3. Rasmussen, N. C., et al., "Reactor Safety Study - An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants," U.S. Nuclear Regulatory Commission, WASH-1400 (NUREG-75/014), October 1975.

b. Maintenance is defined to be maintenance whereby the component is unable to perform its function. Also, unavailability due to maintenance is calculated as the frequency of failure times the repair time.

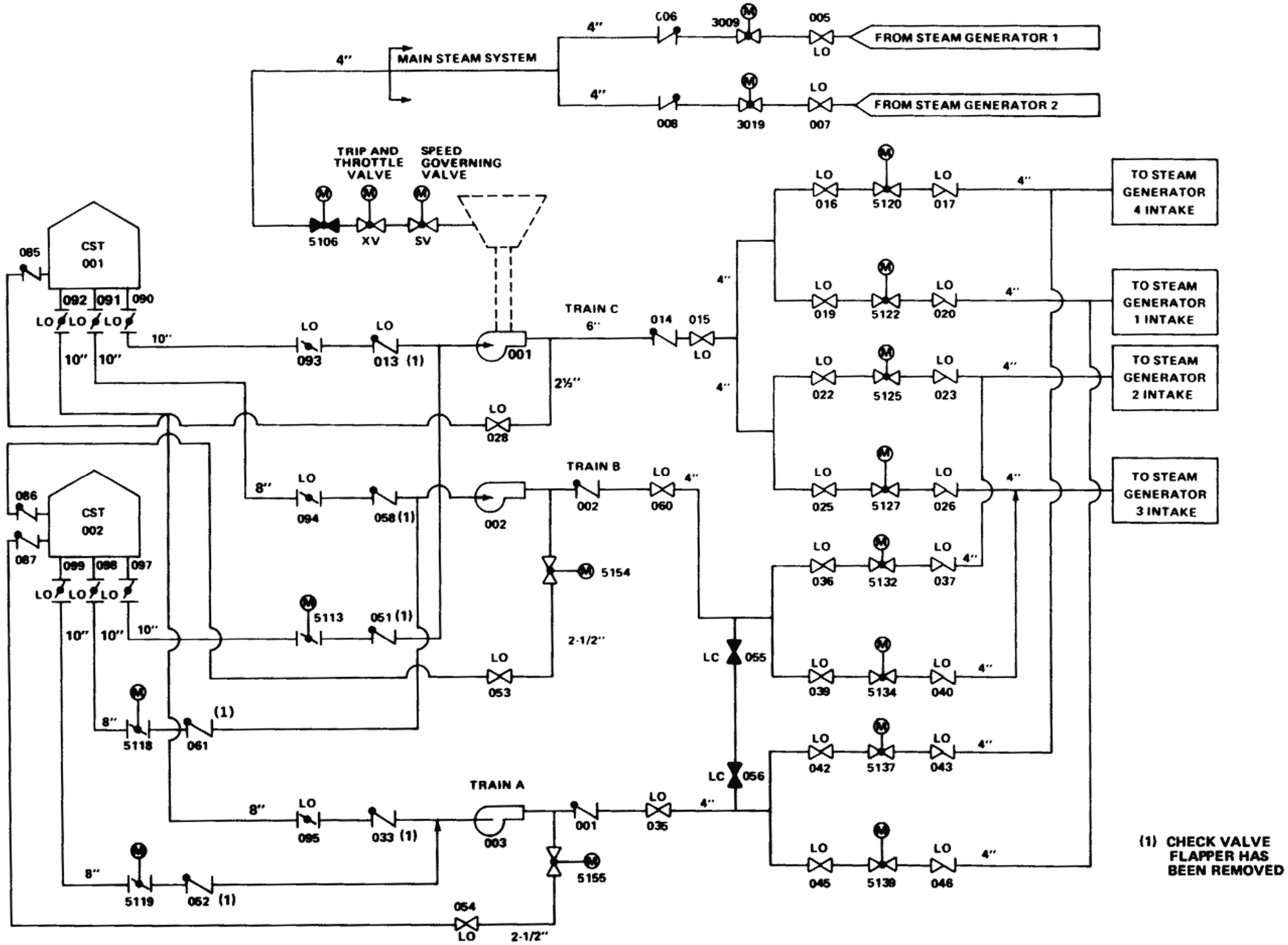
TABLE 10A-5
AFWS CONDITIONAL AVAILABILITY COMPARISON



A. ⟶| |⟵ ORDER OF MAGNITUDE IN UNAVAILABILITY REPRESENTED.

⟶ INCREASING AVAILABILITY.

B. THE SCALE FOR THIS EVENT IS NOT THE SAME AS THAT FOR THE LMFW AND THE LMFW/LOOP.



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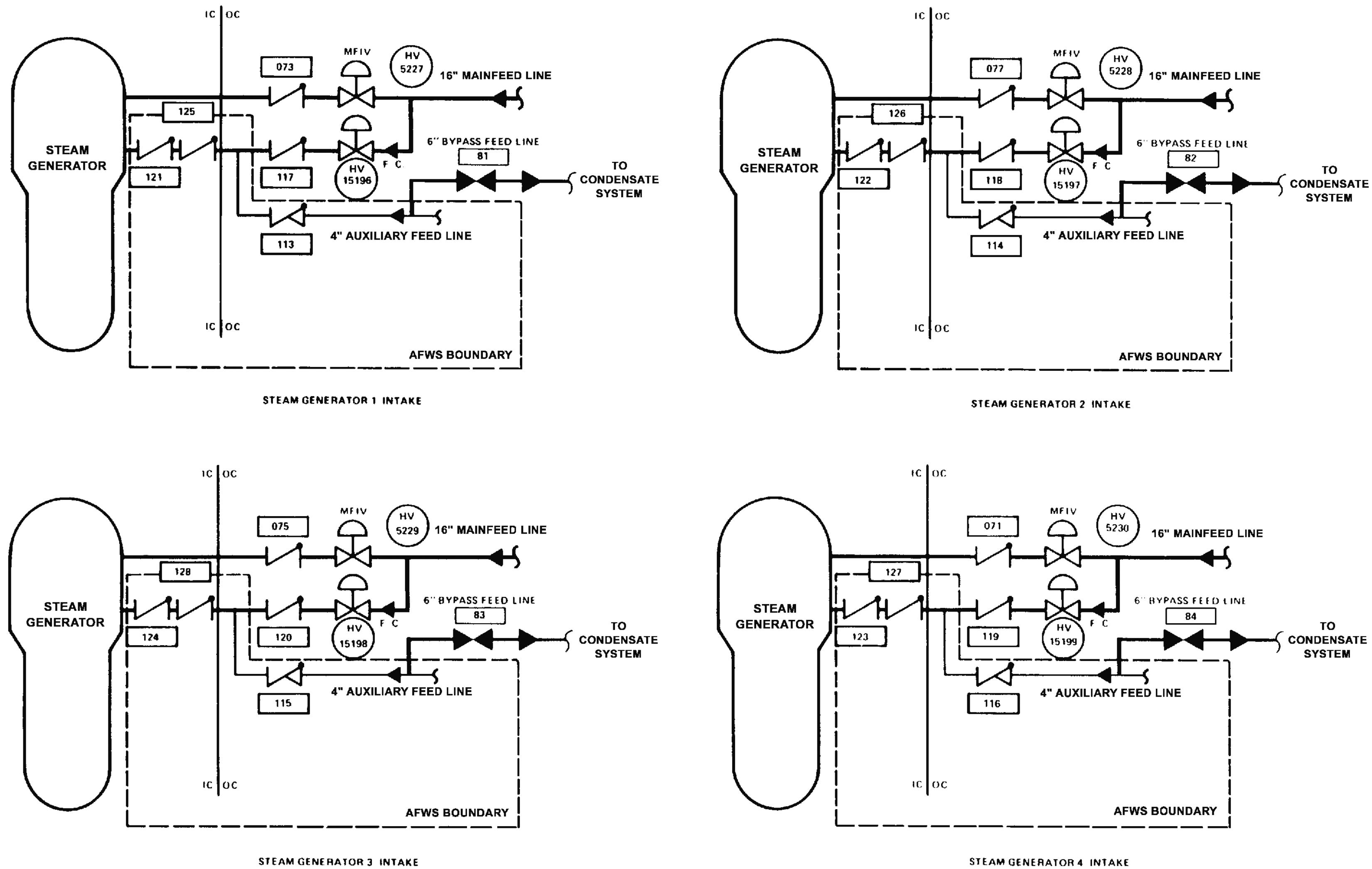
UNIT 1 AND UNIT 2

AFWS

FIGURE 10A-1



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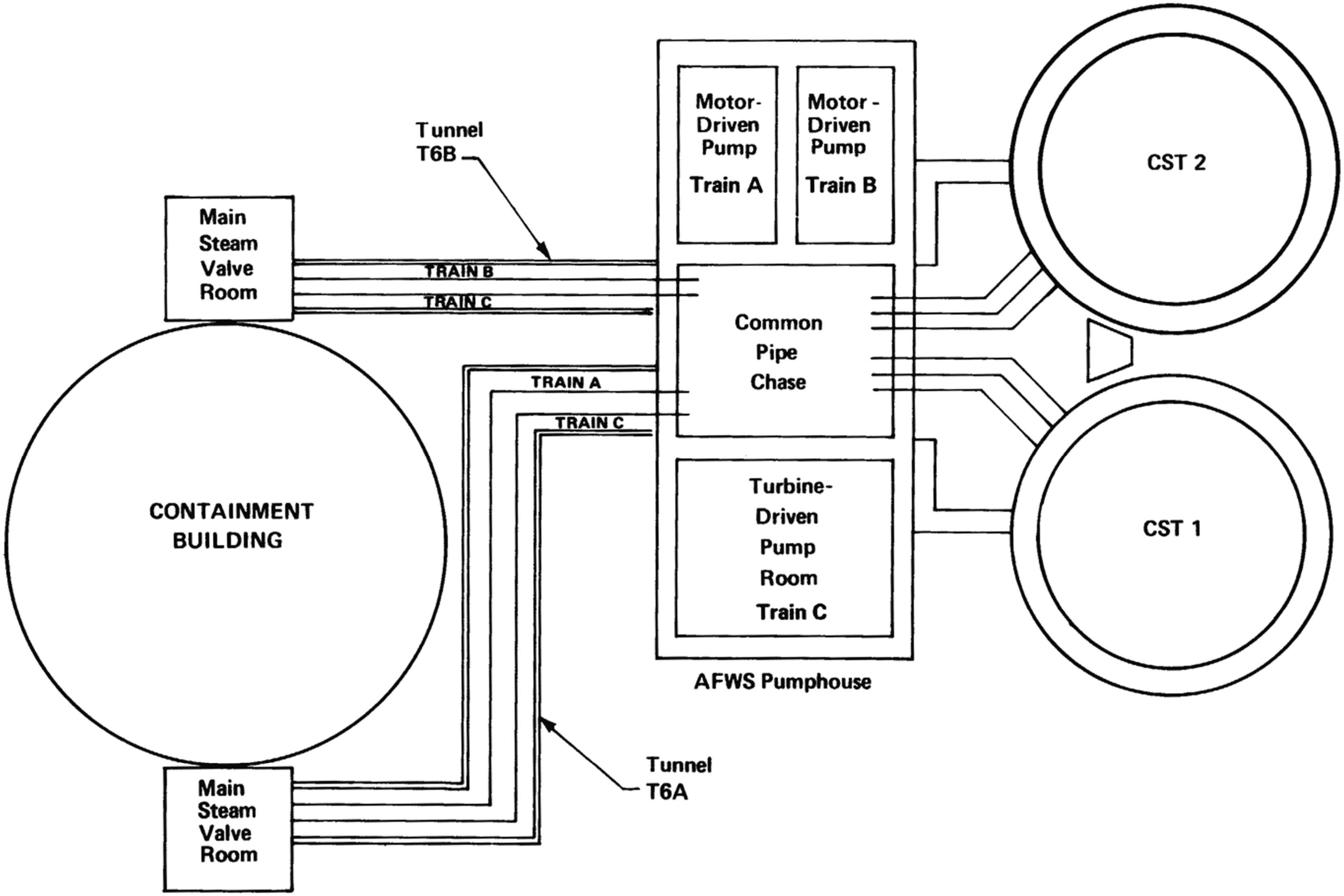
SOUTHERN COMPANY *Energy to Serve Your World®* VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2	UNIT 1 AUXILIARY FEEDWATER/ STEAM GENERATORS INTAKE
	FIGURE 10A-2



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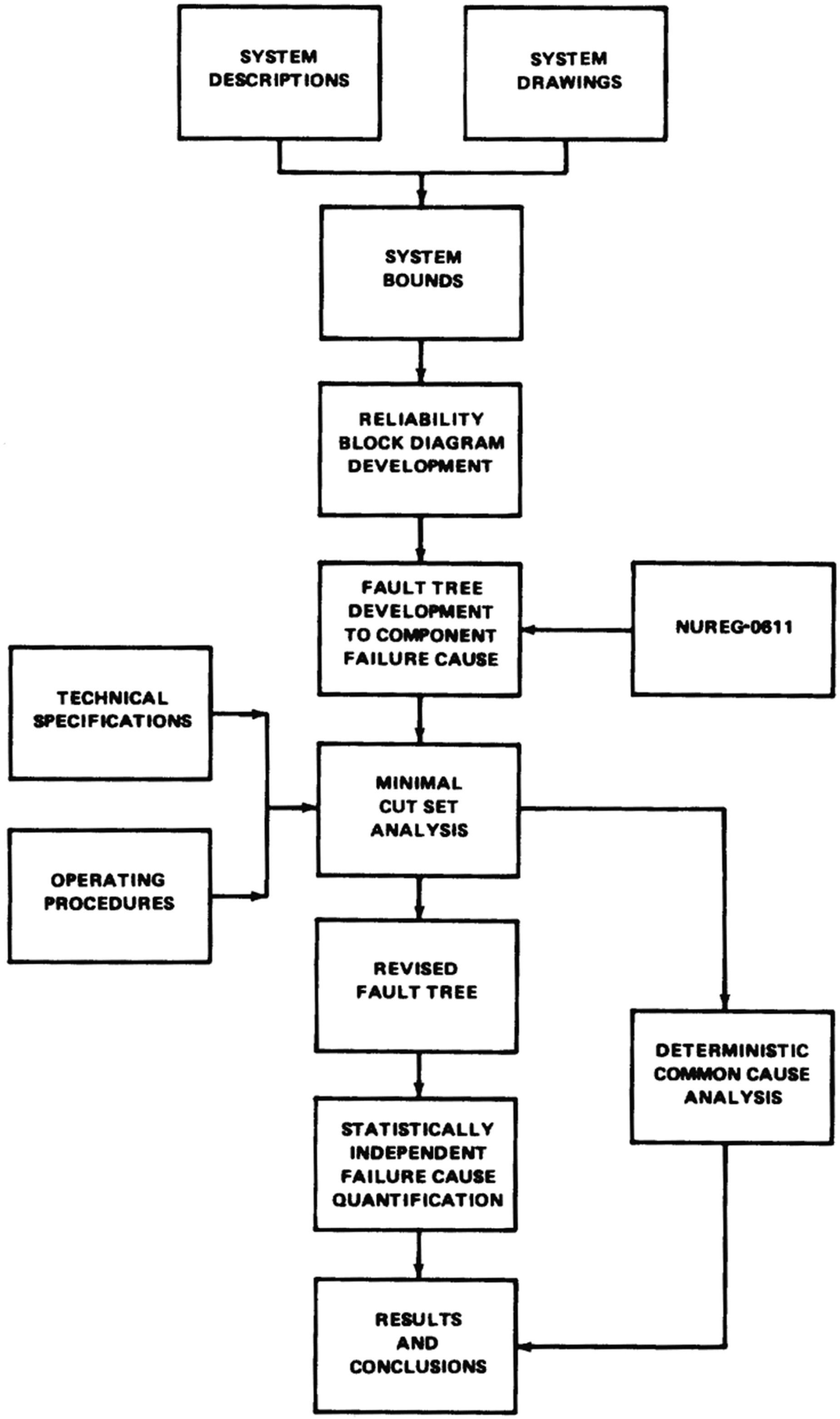
AFWS SIMPLIFIED PIPING LAYOUT

FIGURE 10A-3



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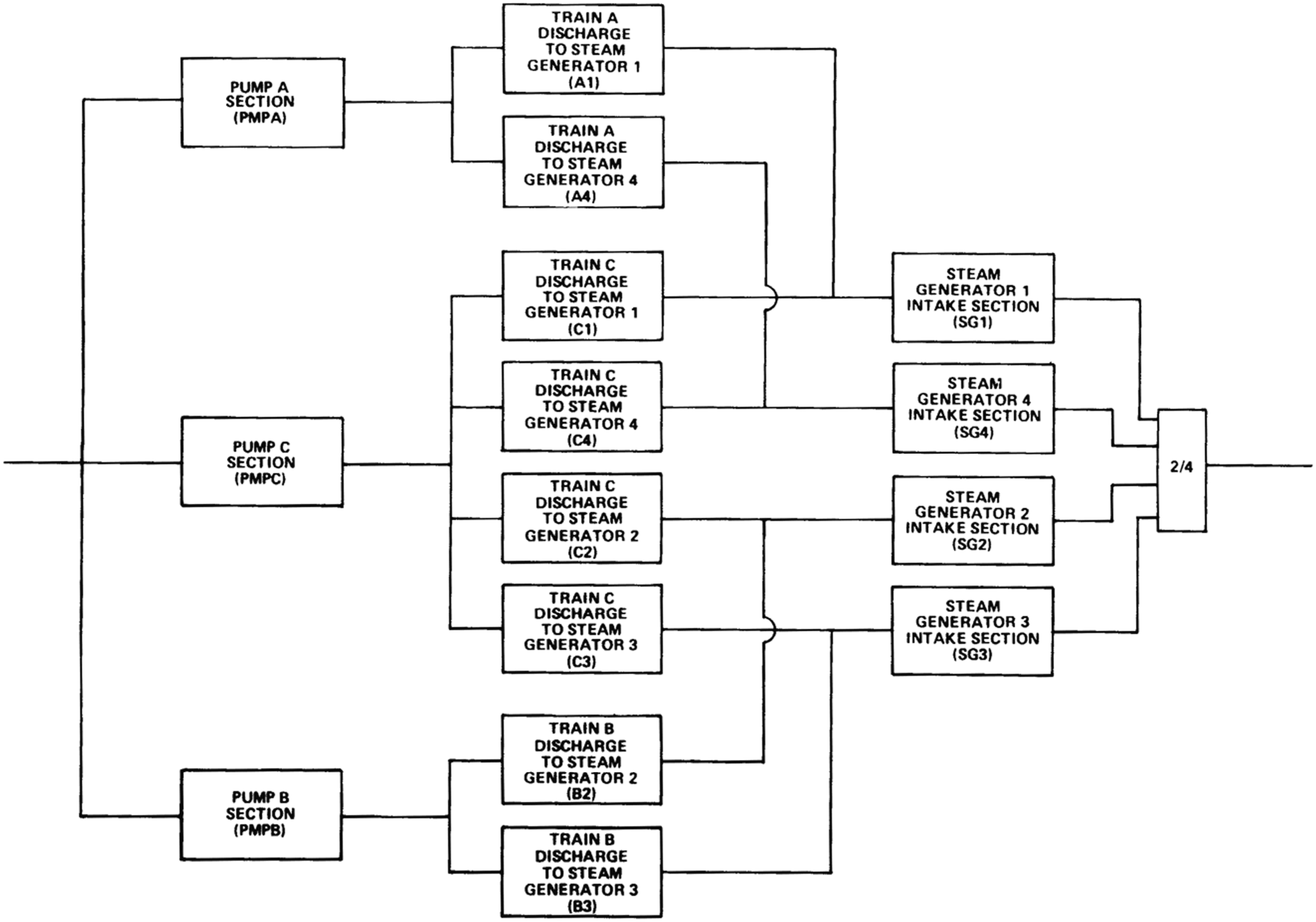
UNIT 1 AND UNIT 2

AFWS RELIABILITY EVALUATION METHODOLOGY FLOW CHART

FIGURE 10A-4



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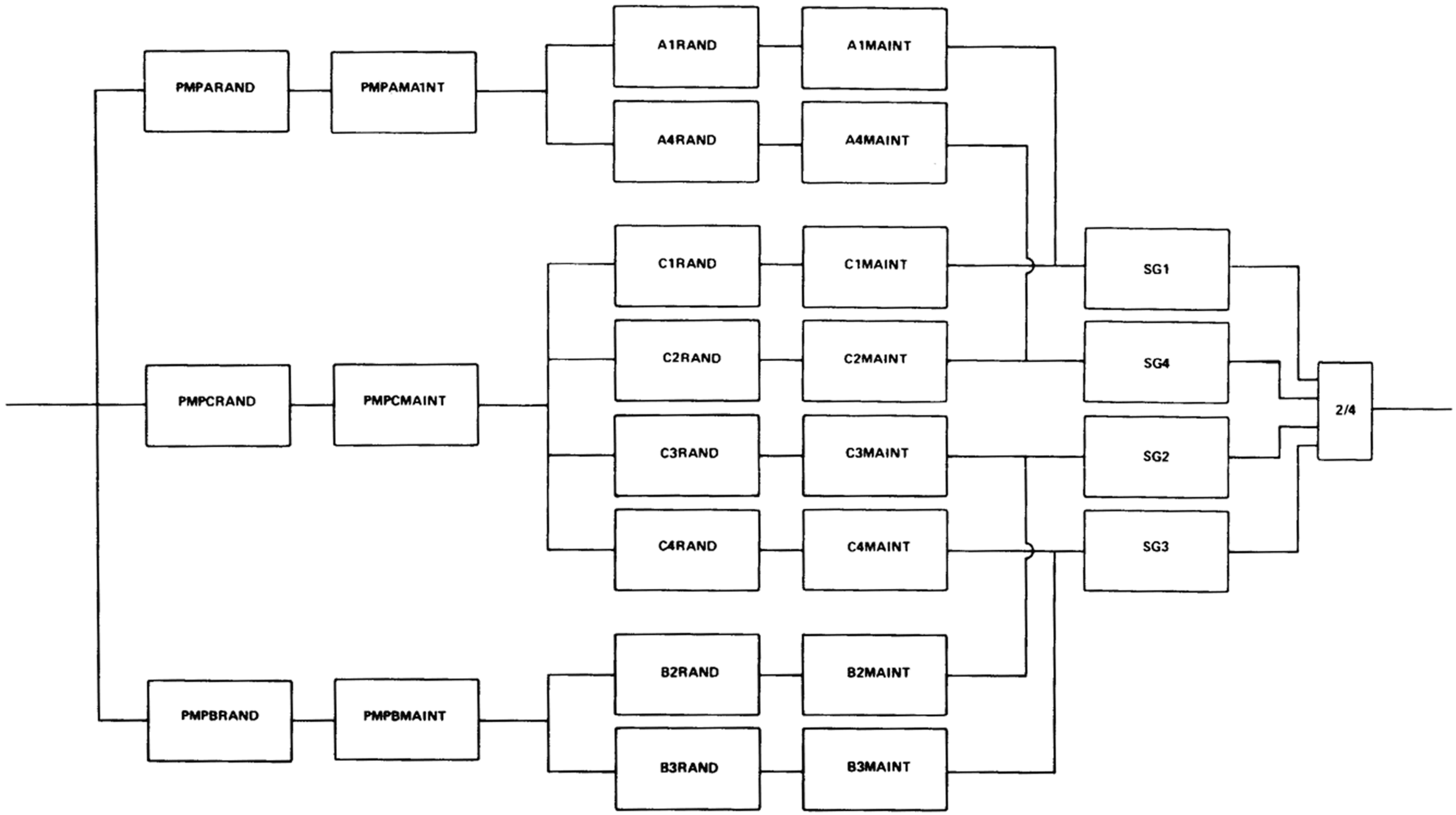
UNIT 1 AND UNIT 2

UNIT 1 AFWS BLOCK DIAGRAM

FIGURE 10A-5



AFWS BLOCK DIAGRAM

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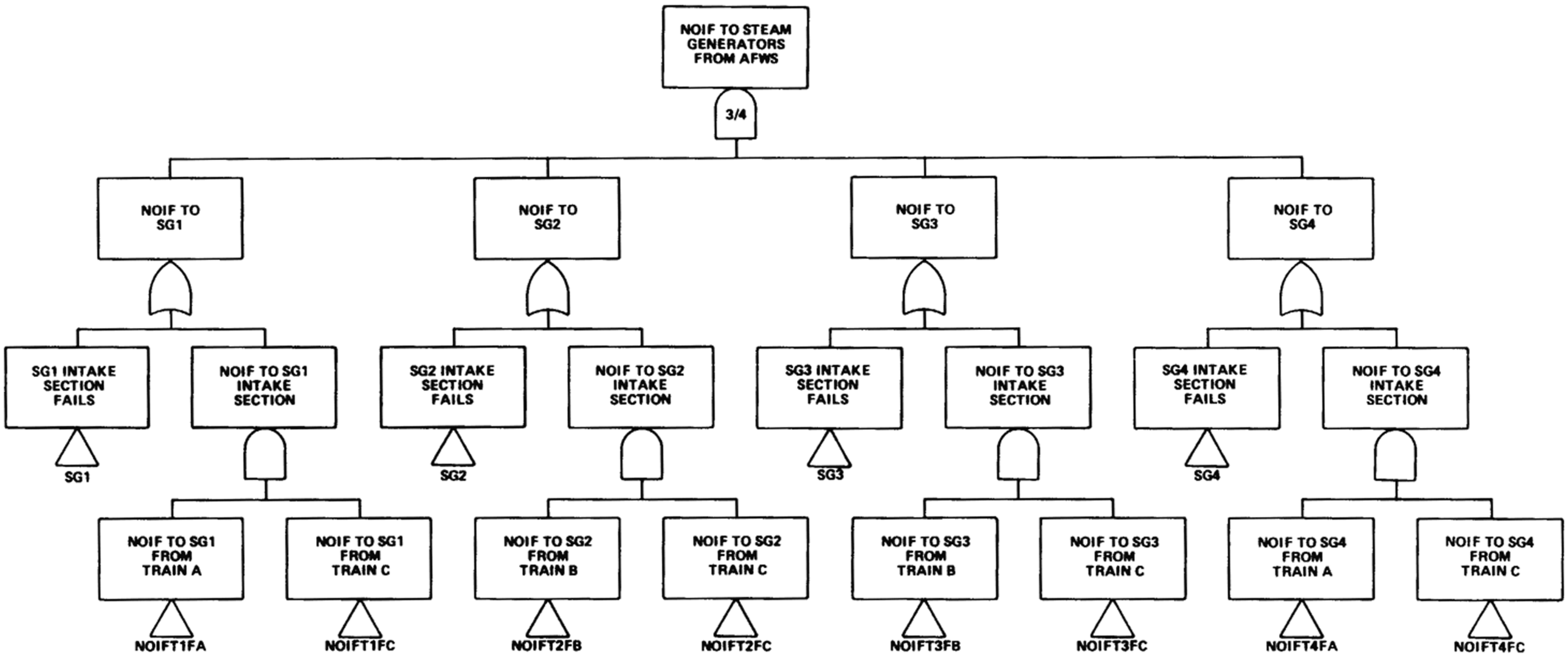
UNIT 1 AND UNIT 2

AFWS EXPANDED BLOCK DIAGRAM

FIGURE 10A-6



NOIF = NO OR INSUFFICIENT FLOW

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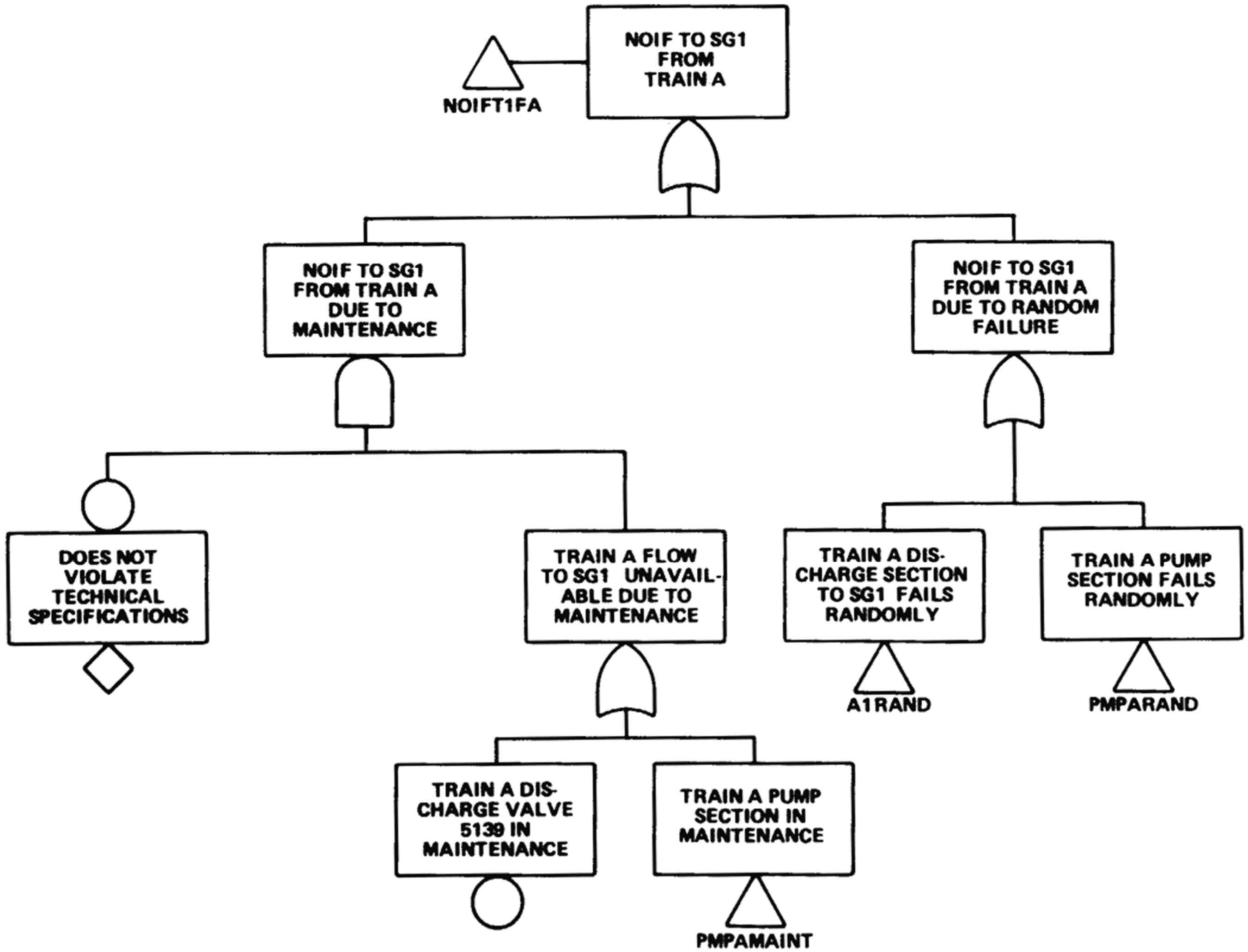
UNIT 1 AND UNIT 2

UNIT 1 AFWS FAULT TREE MODEL

FIGURE 10A-7 (SHEET 1 OF 30)



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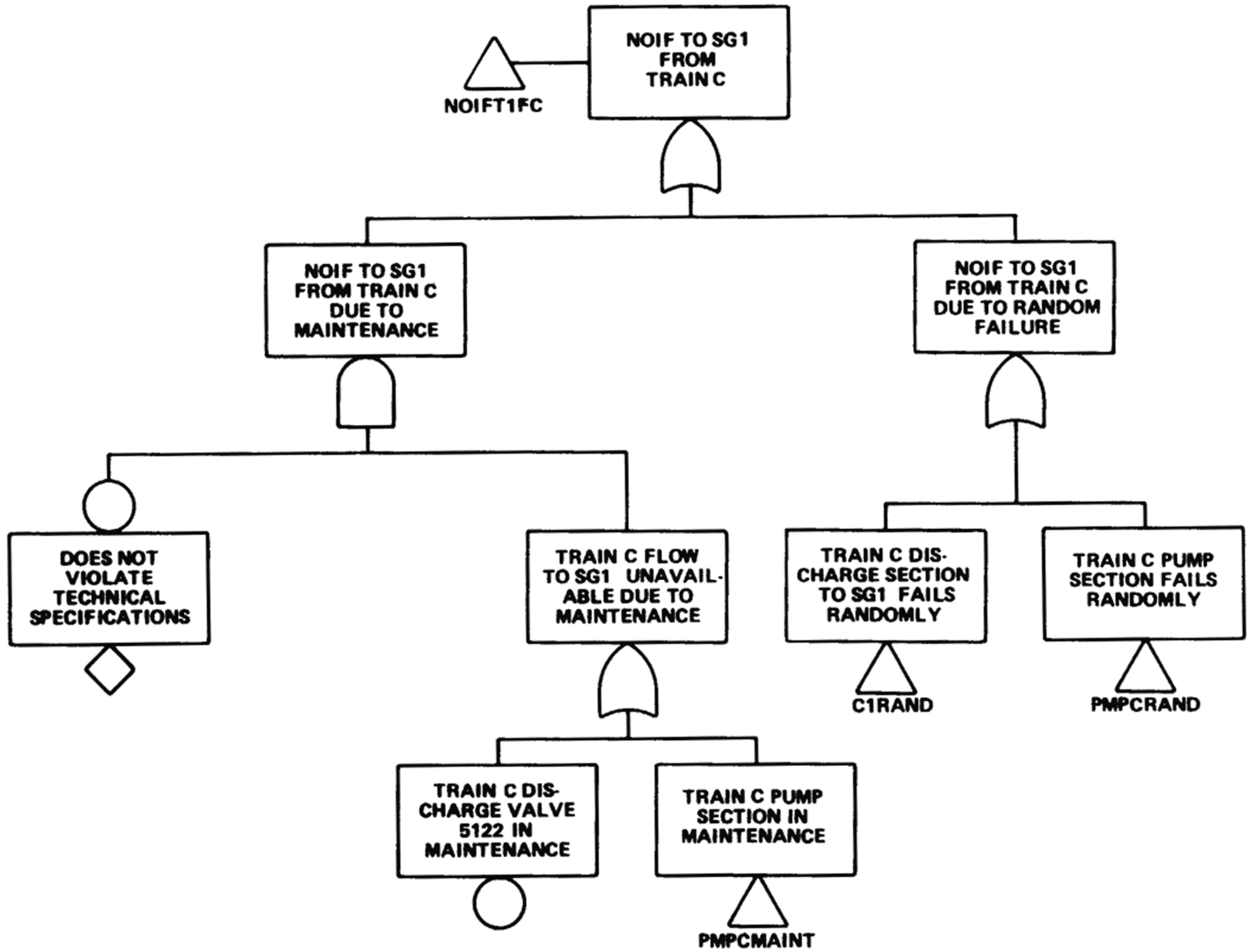
UNIT 1 AND UNIT 2

UNIT 1 AFWS FAULT TREE MODEL

FIGURE 10A-7 (SHEET 2 OF 30)



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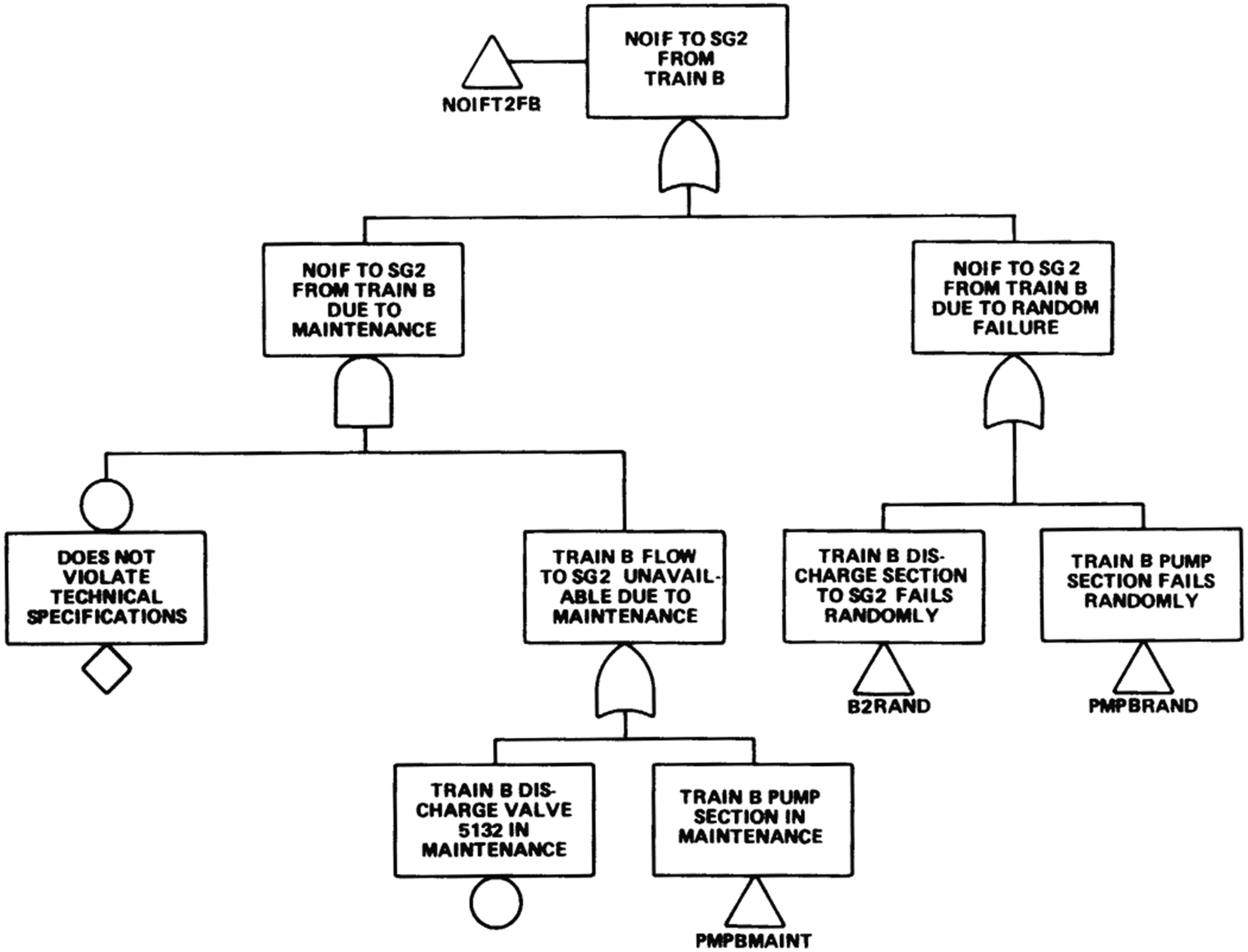
SOUTHERN COMPANY *Energy to Serve Your World®*	VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2	UNIT 1 AFWS FAULT TREE MODEL FIGURE 10A-7 (SHEET 3 OF 30)



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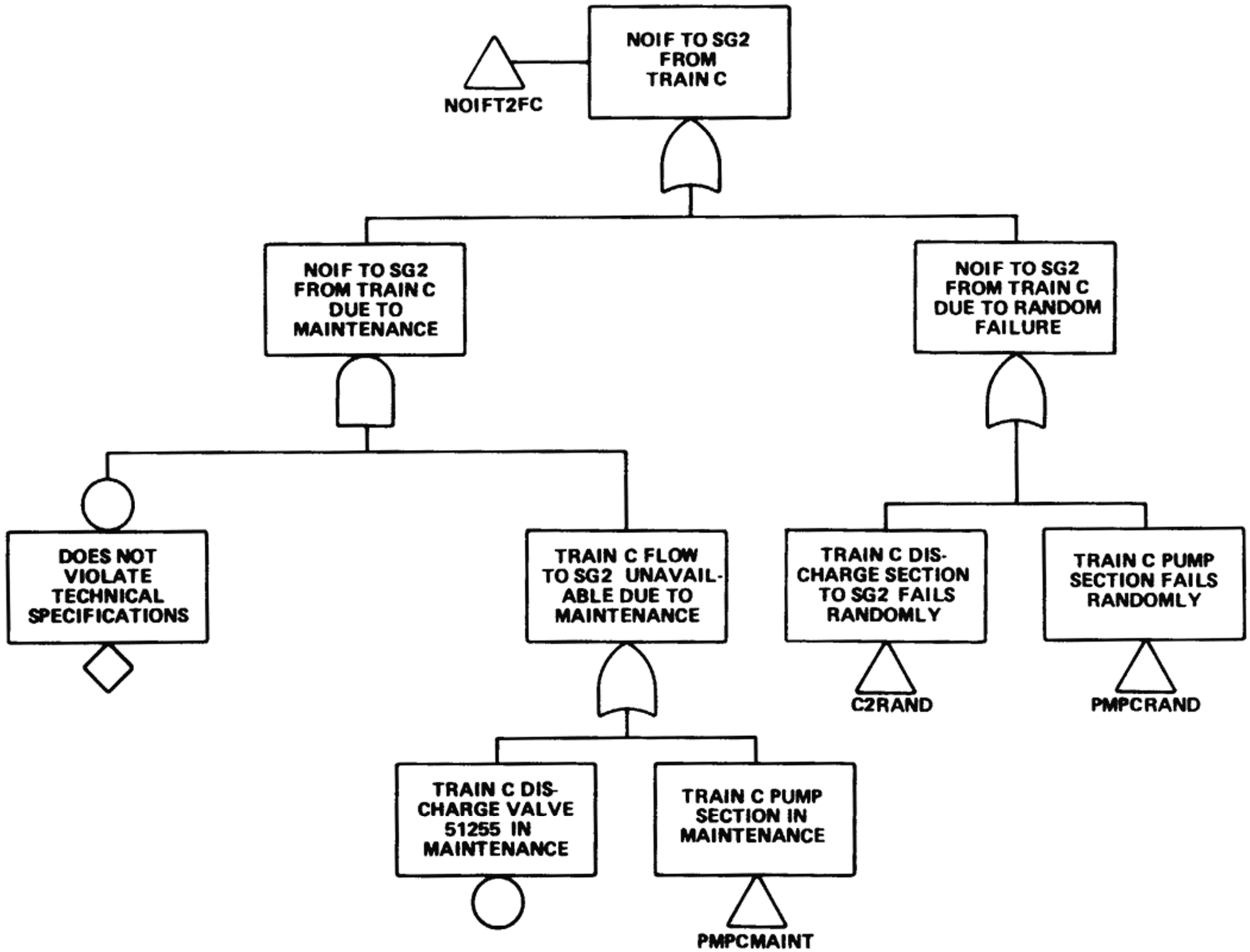
UNIT 1 AND UNIT 2

UNIT 1 AFWS FAULT TREE MODEL

FIGURE 10A-7 (SHEET 4 OF 30)



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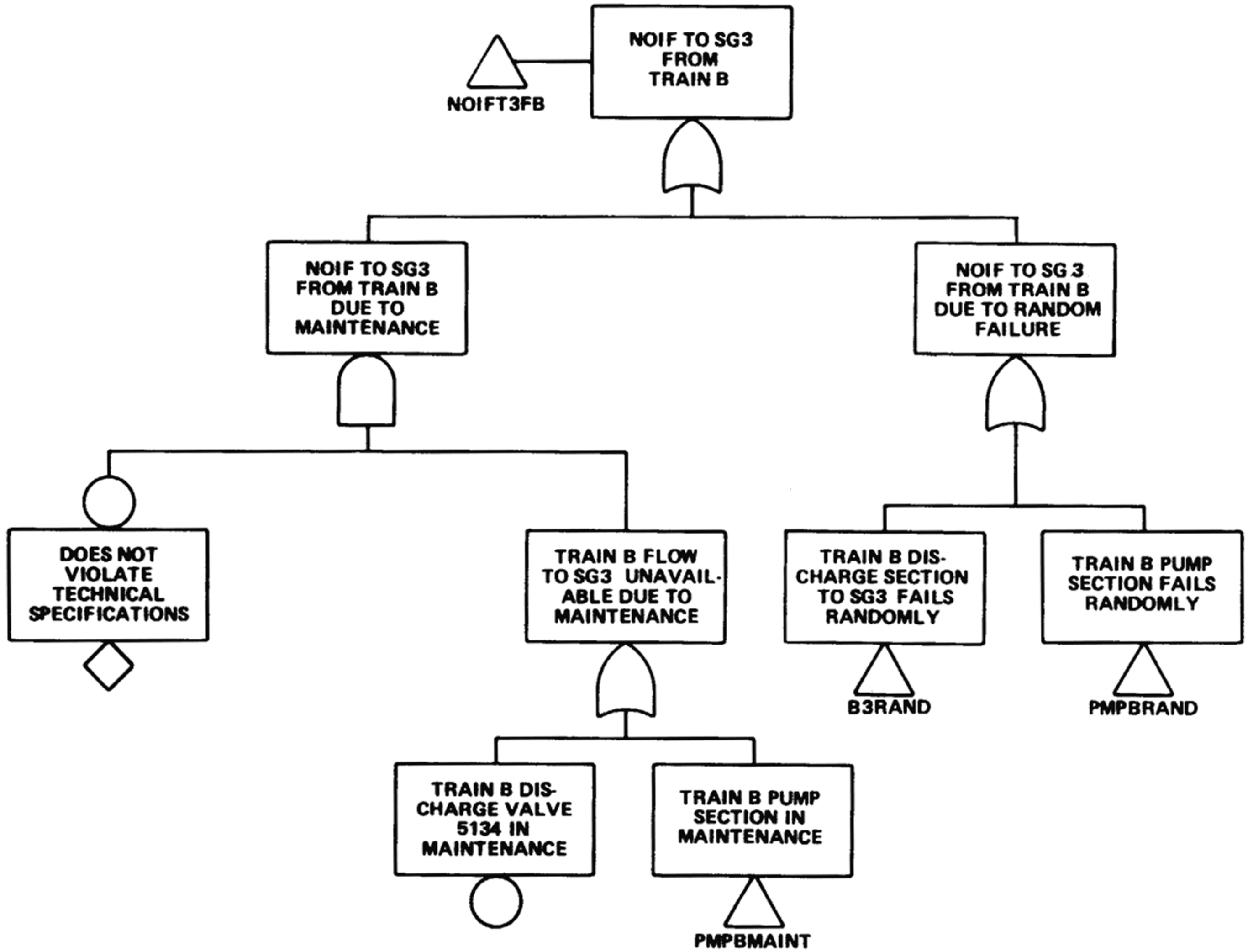
UNIT 1 AND UNIT 2

UNIT 1 AFWS FAULT TREE MODEL

FIGURE 10A-7 (SHEET 5 OF 30)



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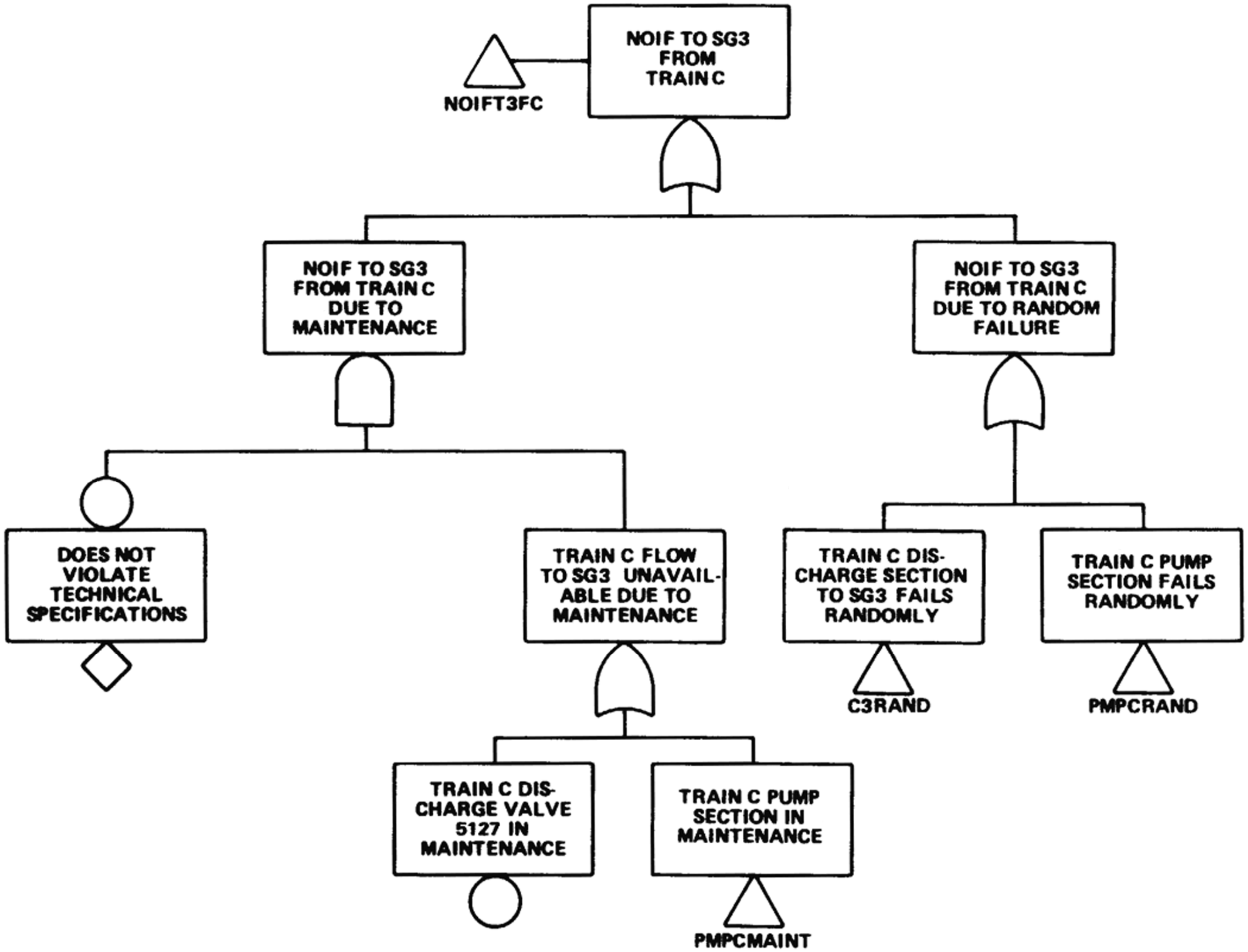
SOUTHERN COMPANY Energy to Serve Your World®	VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2	UNIT 1 AFWS FAULT TREE MODEL FIGURE 10A-7 (SHEET 6 OF 30)



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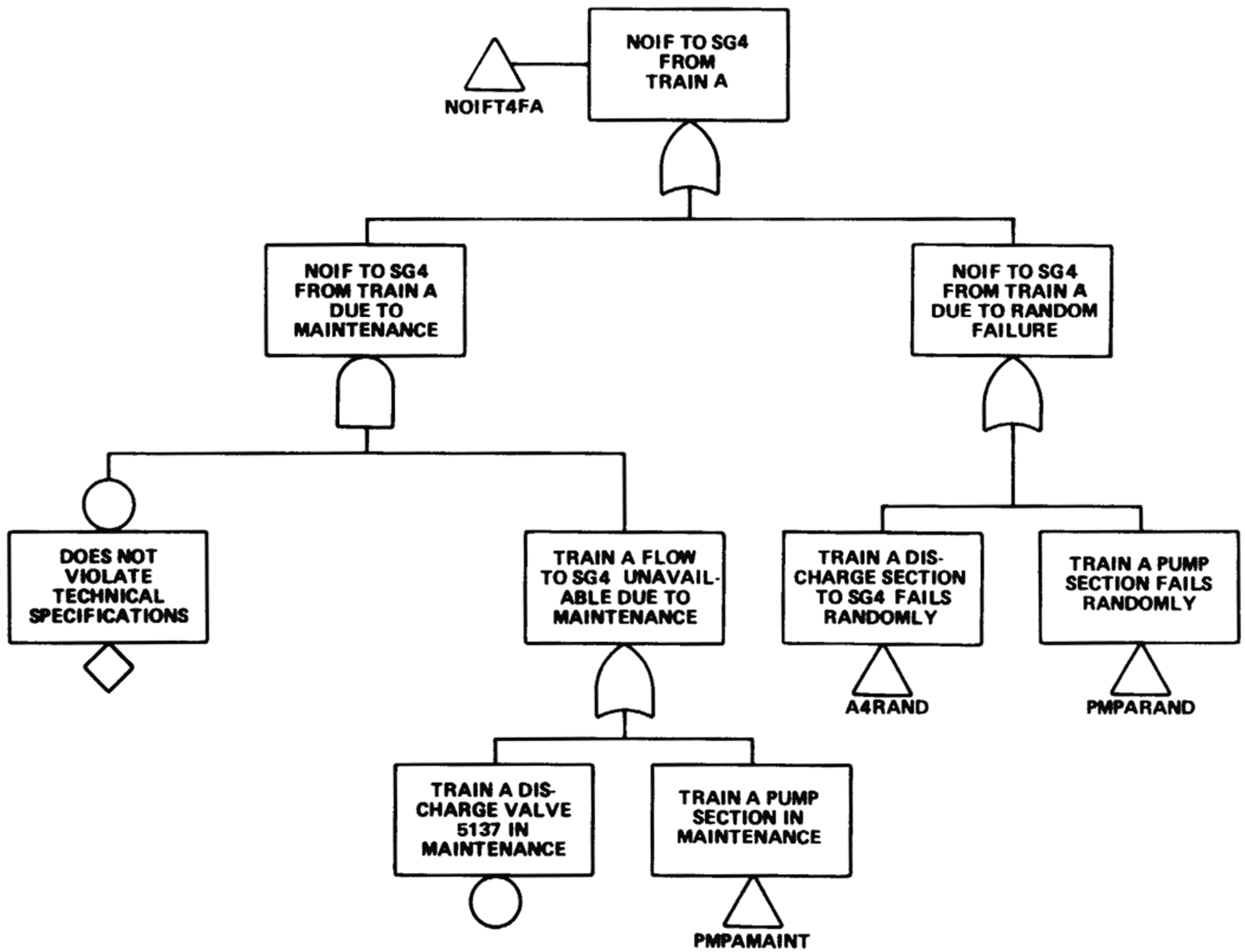
UNIT 1 AND UNIT 2

UNIT 1 AFWS FAULT TREE MODEL

FIGURE 10A-7 (SHEET 7 OF 30)



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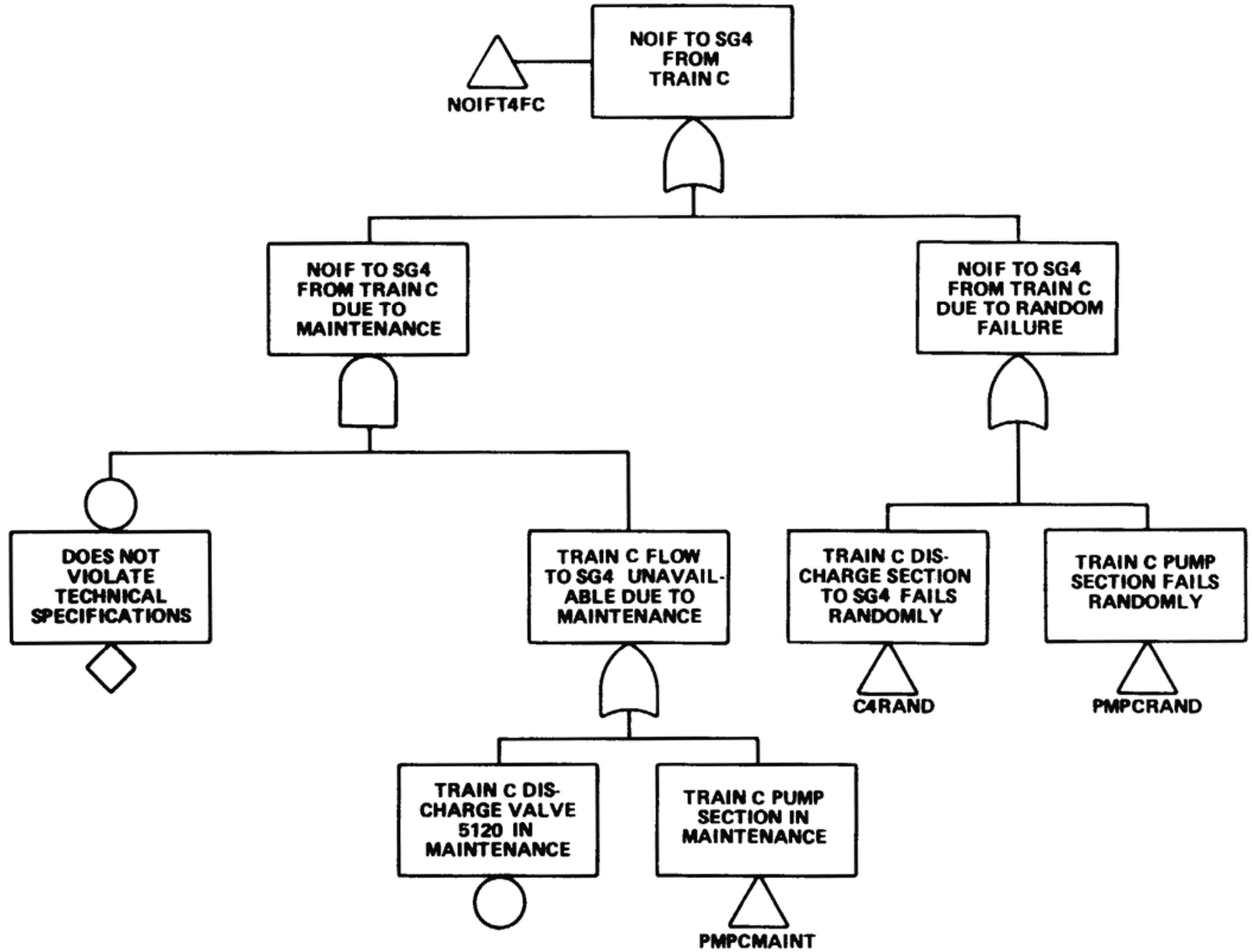
SOUTHERN COMPANY *Energy to Serve Your World®*	VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2	UNIT 1 AFWS FAULT TREE MODEL FIGURE 10A-7 (SHEET 8 OF 30)



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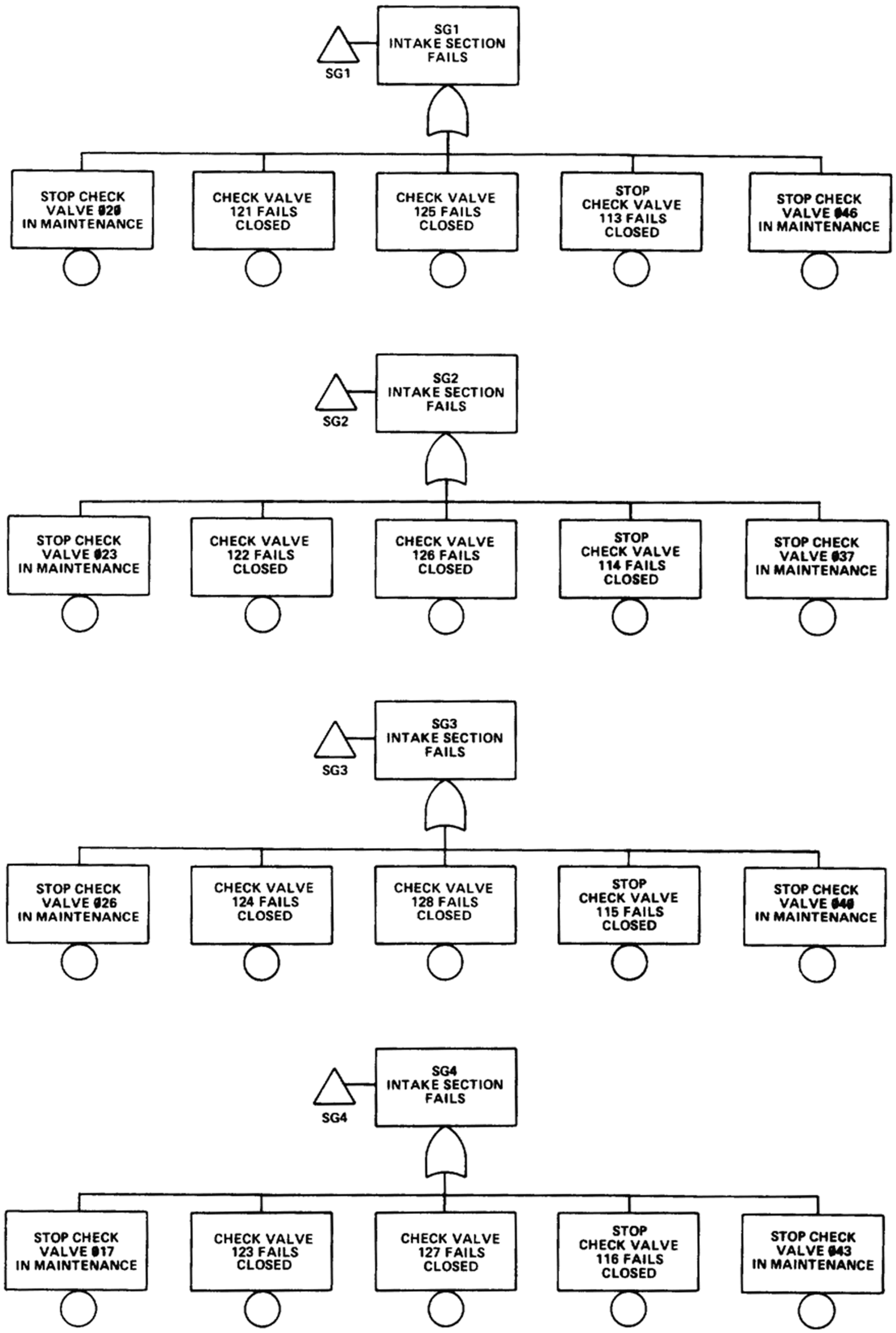
UNIT 1 AND UNIT 2

UNIT 1 AFWS FAULT TREE MODEL

FIGURE 10A-7 (SHEET 9 OF 30)



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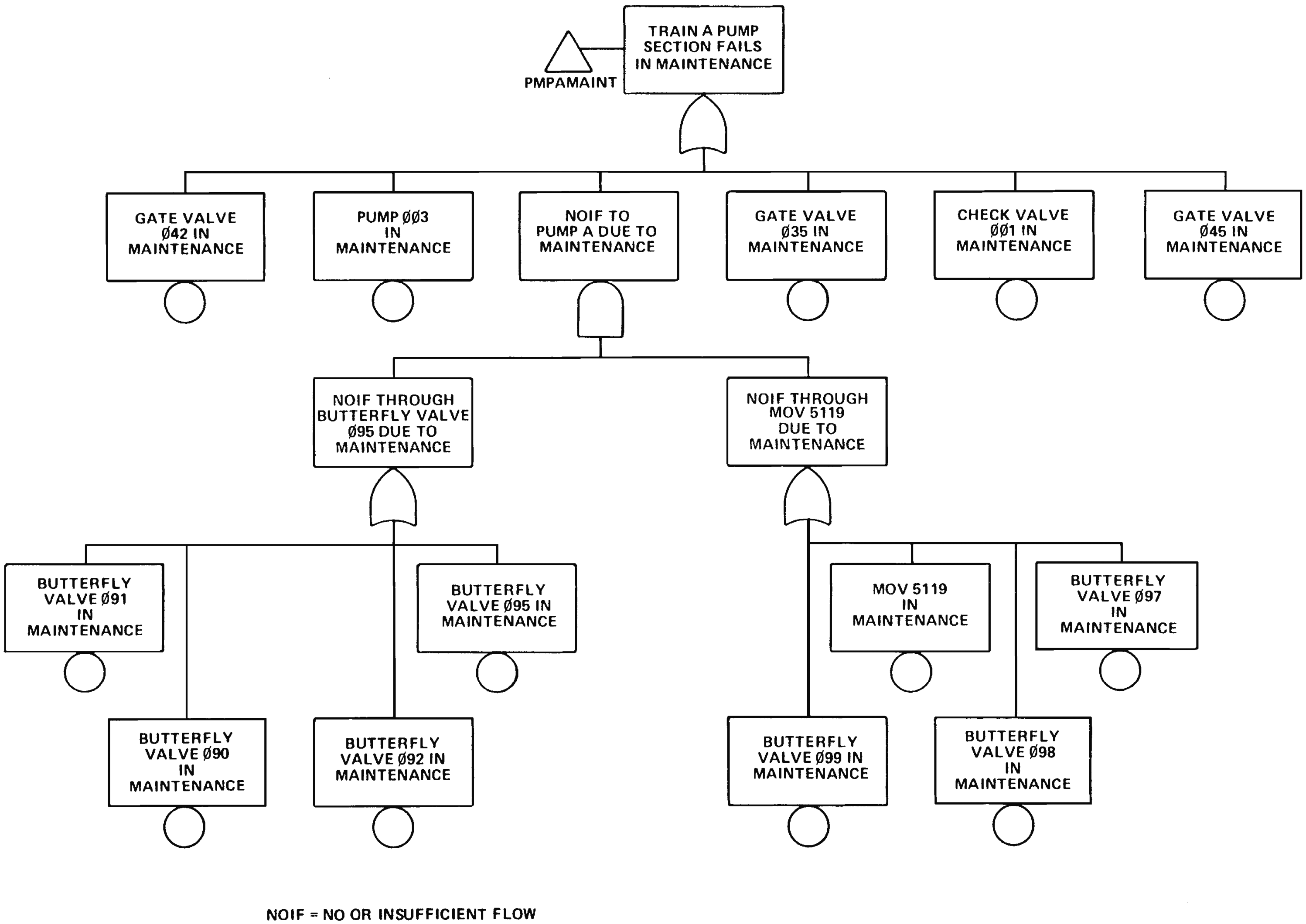
SOUTHERN COMPANY *Energy to Serve Your World®*	VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2	UNIT 1 AFWS FAULT TREE MODEL FIGURE 10A-7 (SHEET 10 OF 30)



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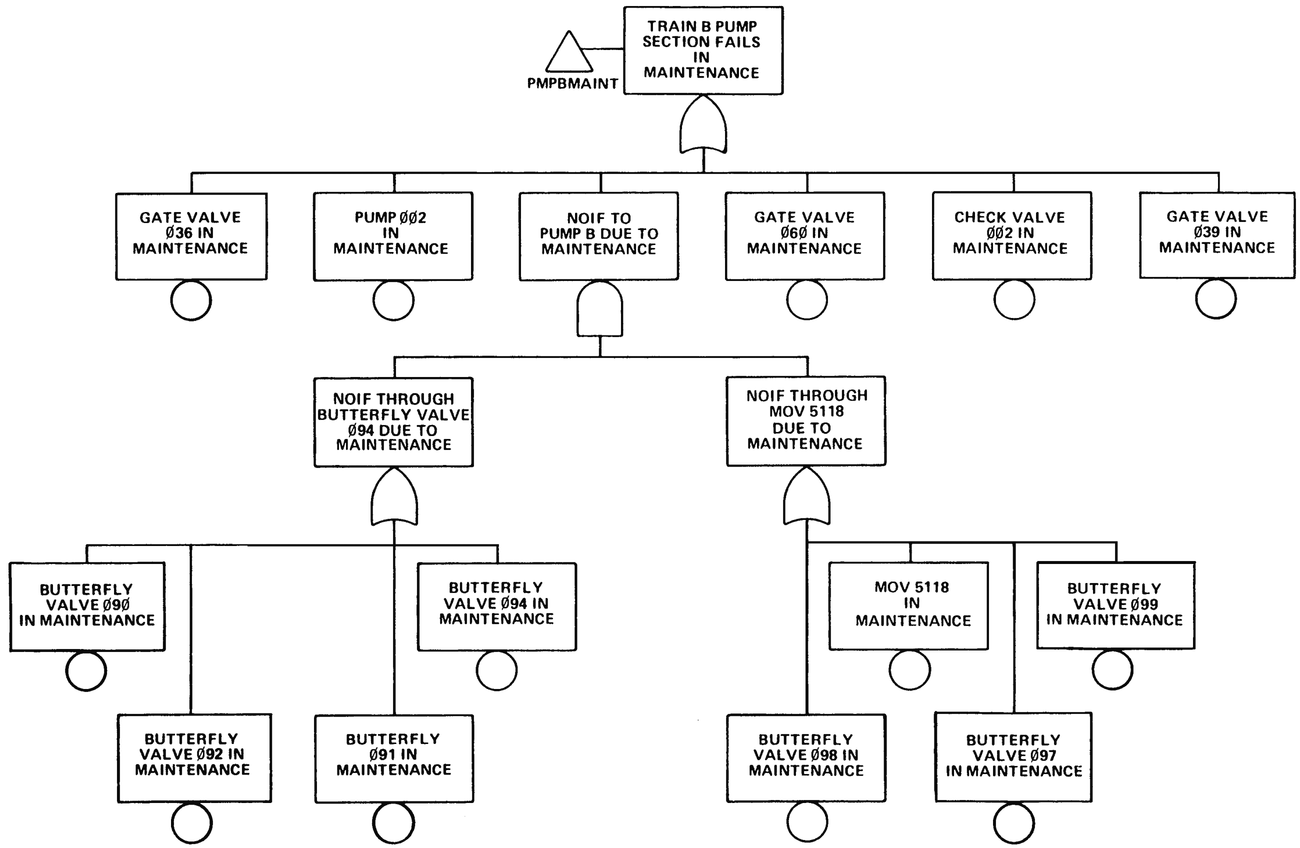
UNIT 1 AND UNIT 2

UNIT 1 AFWS FAULT TREE MODEL

FIGURE 10A-7 (SHEET 11 OF 30)



NOIF = NO OR INSUFFICIENT FLOW

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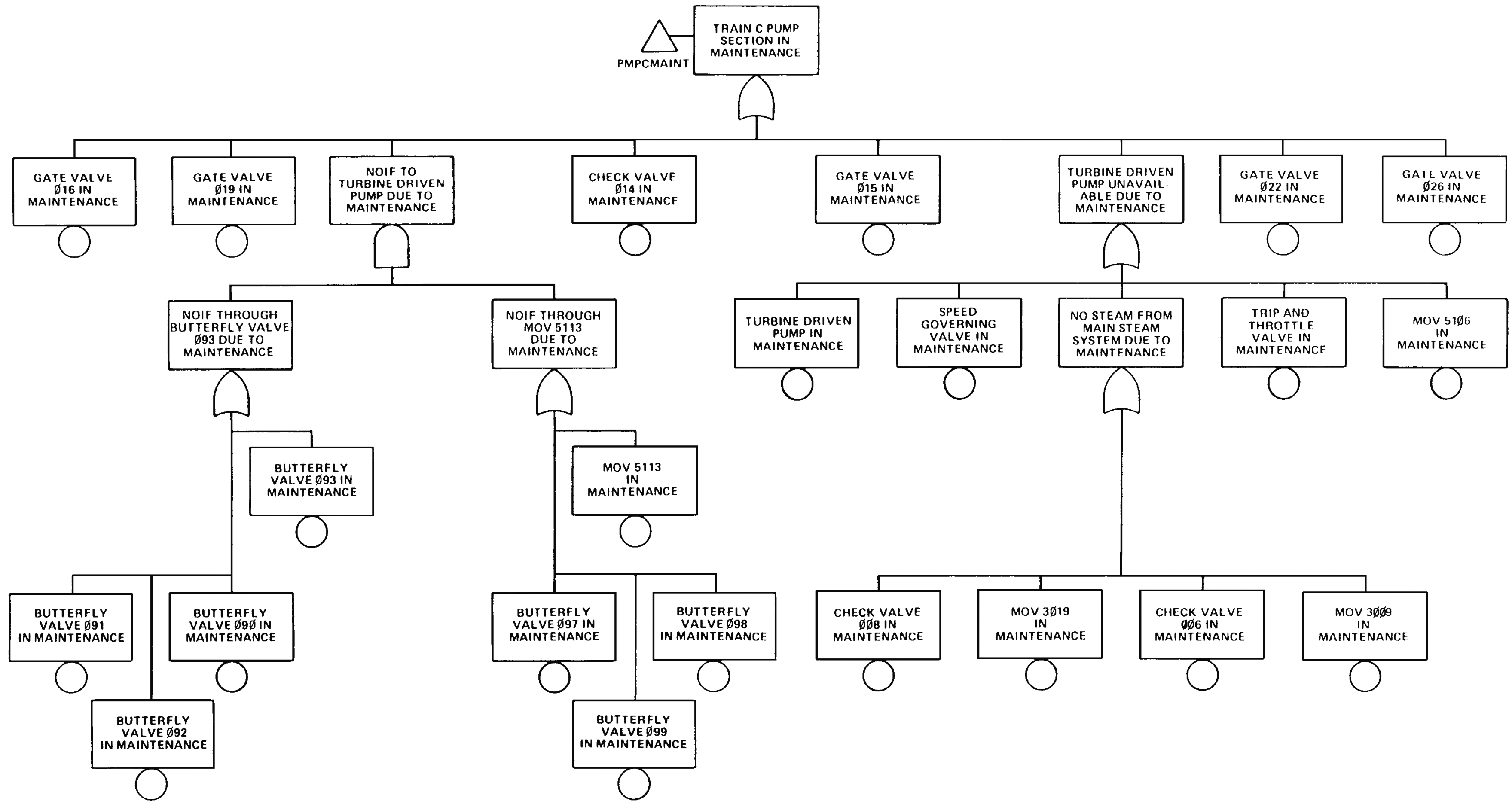
UNIT 1 AND UNIT 2

UNIT 1 AFWS FAULT TREE MODEL

FIGURE 10A-7 (SHEET 12 OF 30)



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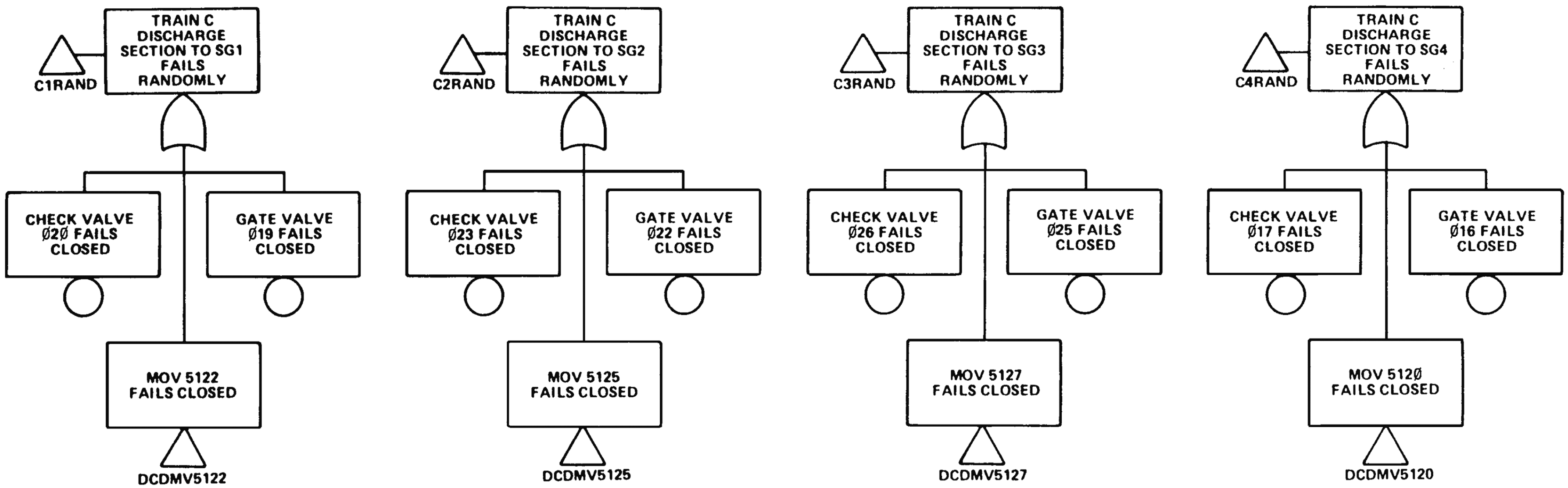
VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2

UNIT 1 AFWS FAULT TREE MODEL

FIGURE 10A-7 (SHEET 13 OF 30)



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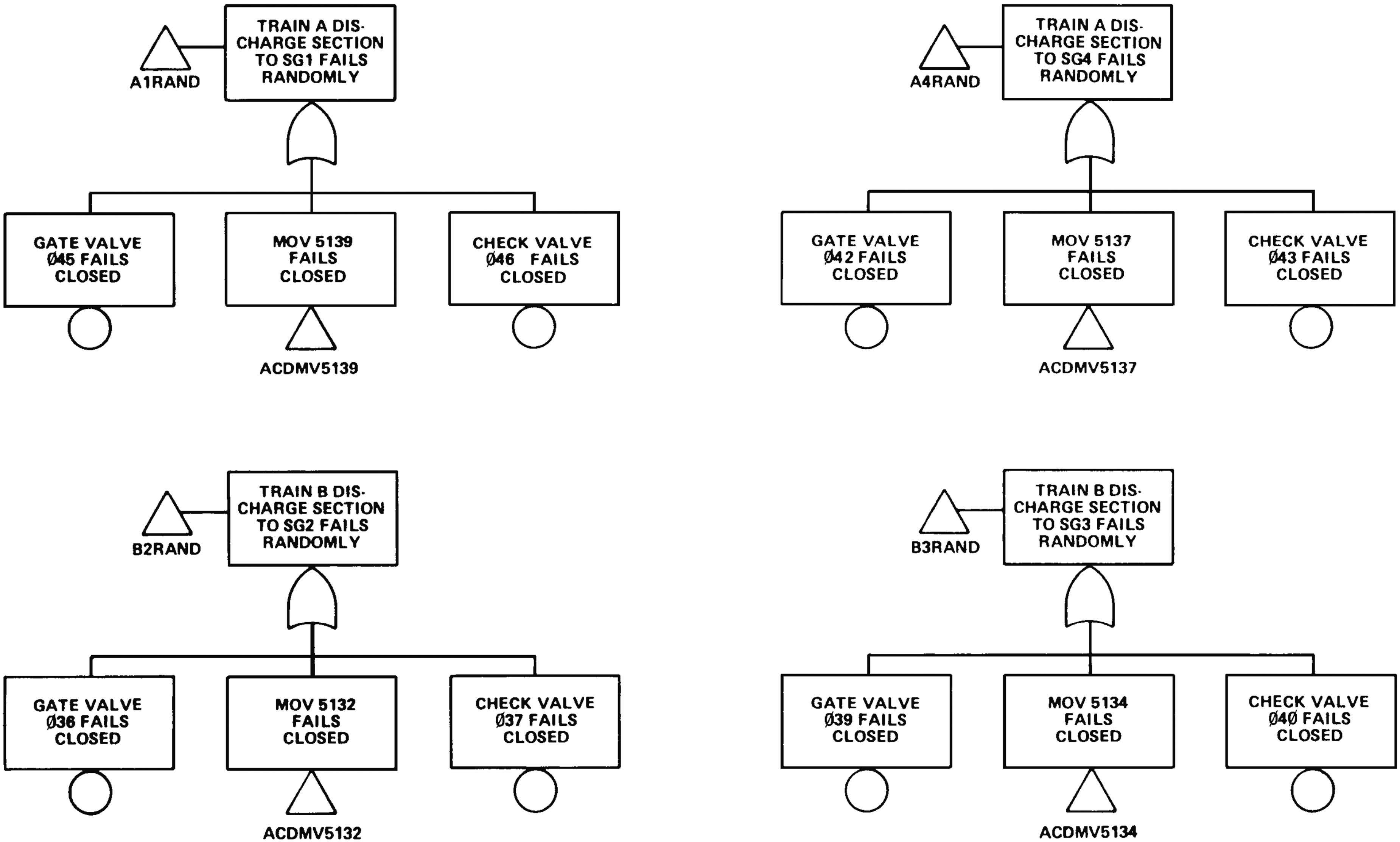
UNIT 1 AND UNIT 2

UNIT 1 AFWS FAULT TREE MODEL

FIGURE 10A-7 (SHEET 14 OF 30)



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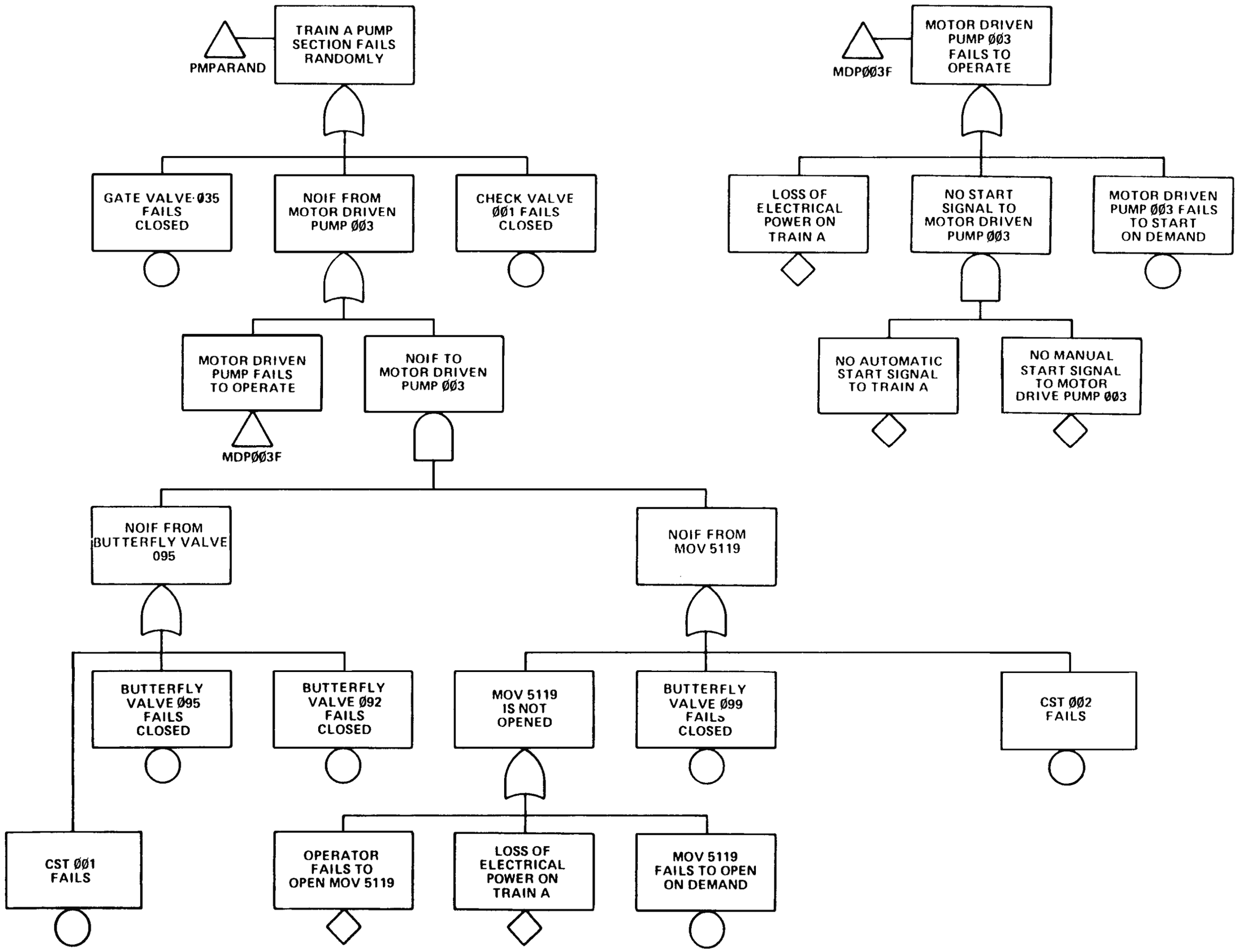
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

UNIT 1 AFWS FAULT TREE MODEL

FIGURE 10A-7 (SHEET 15 OF 30)



NOIF = NO OR INSUFFICIENT FLOW

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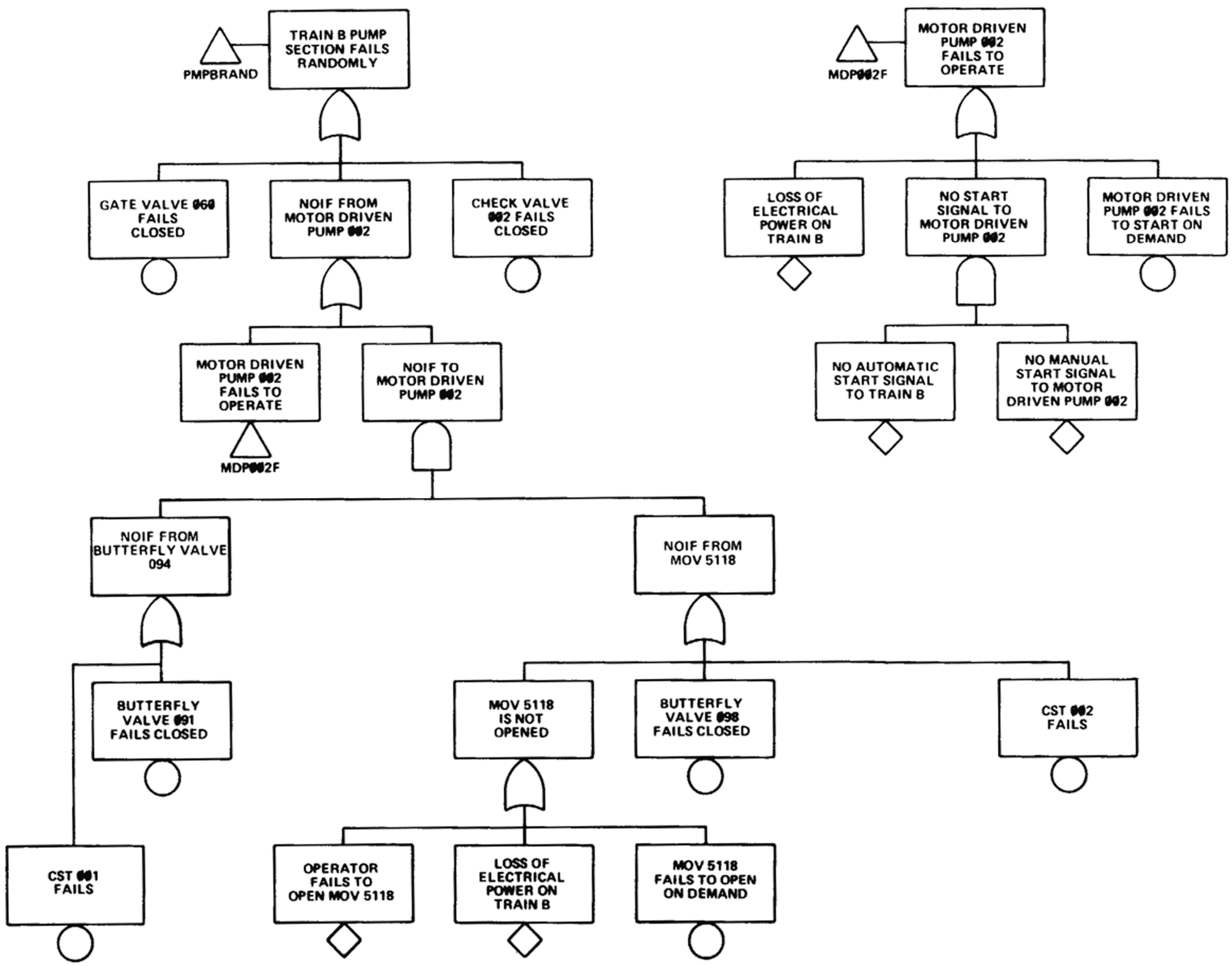
SOUTHERN COMPANY *Energy to Serve Your World®*	**VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2**	UNIT 1 AFWS FAULT TREE MODEL FIGURE 10A-7 (SHEET 16 OF 30)



NOIF = NO OR INSUFFICIENT FLOW

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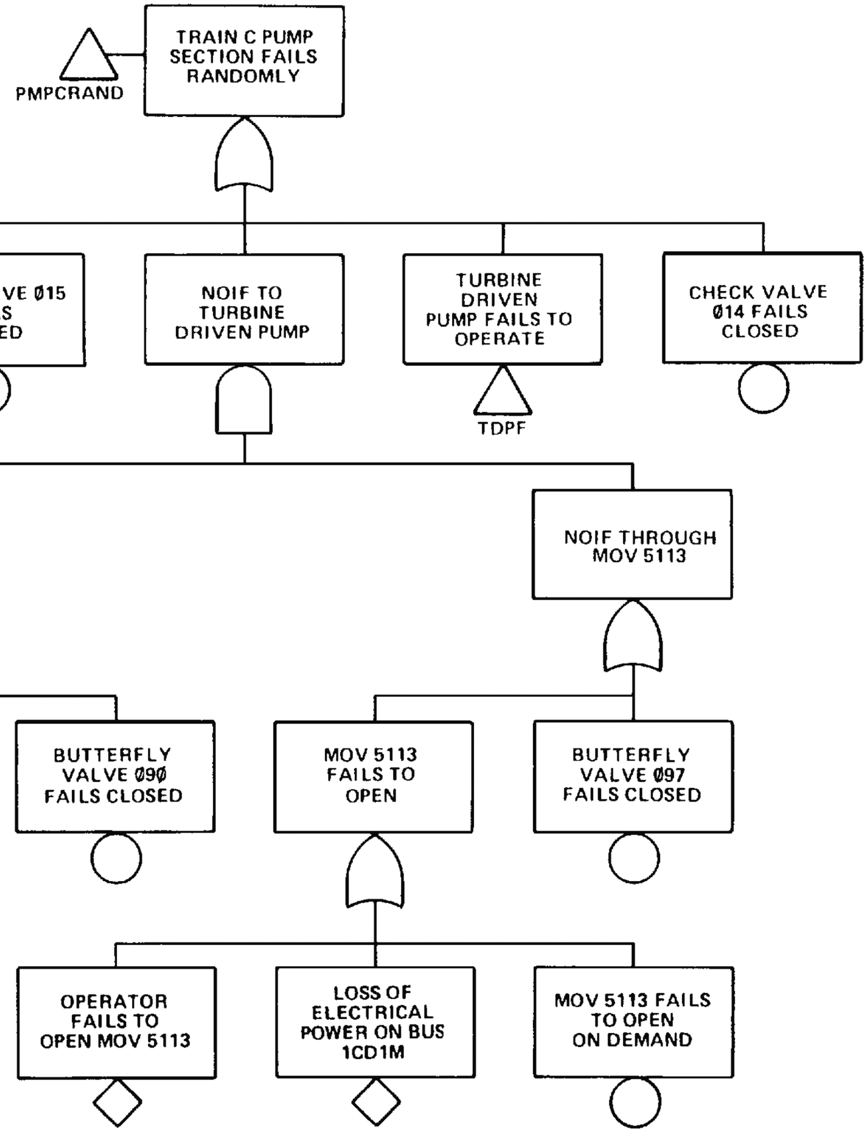
UNIT 1 AND UNIT 2

UNIT 1 AFWS FAULT TREE MODEL

FIGURE 10A-7 (SHEET 17 OF 30)



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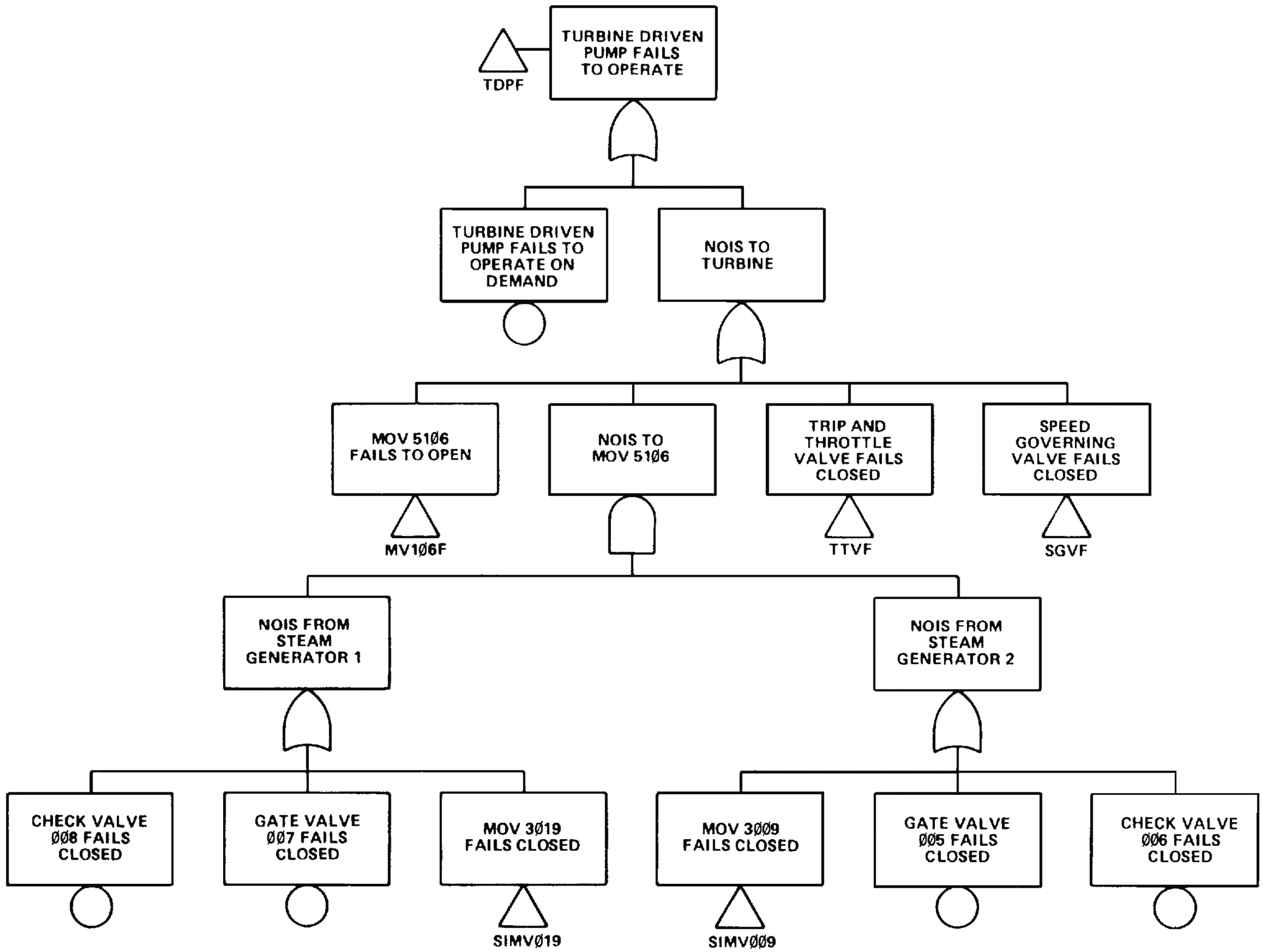
UNIT 1 AND UNIT 2

UNIT 1 AFWS FAULT TREE MODEL

FIGURE 10A-7 (SHEET 18 OF 30)



NOIS = NO OR INSUFFICIENT STEAM

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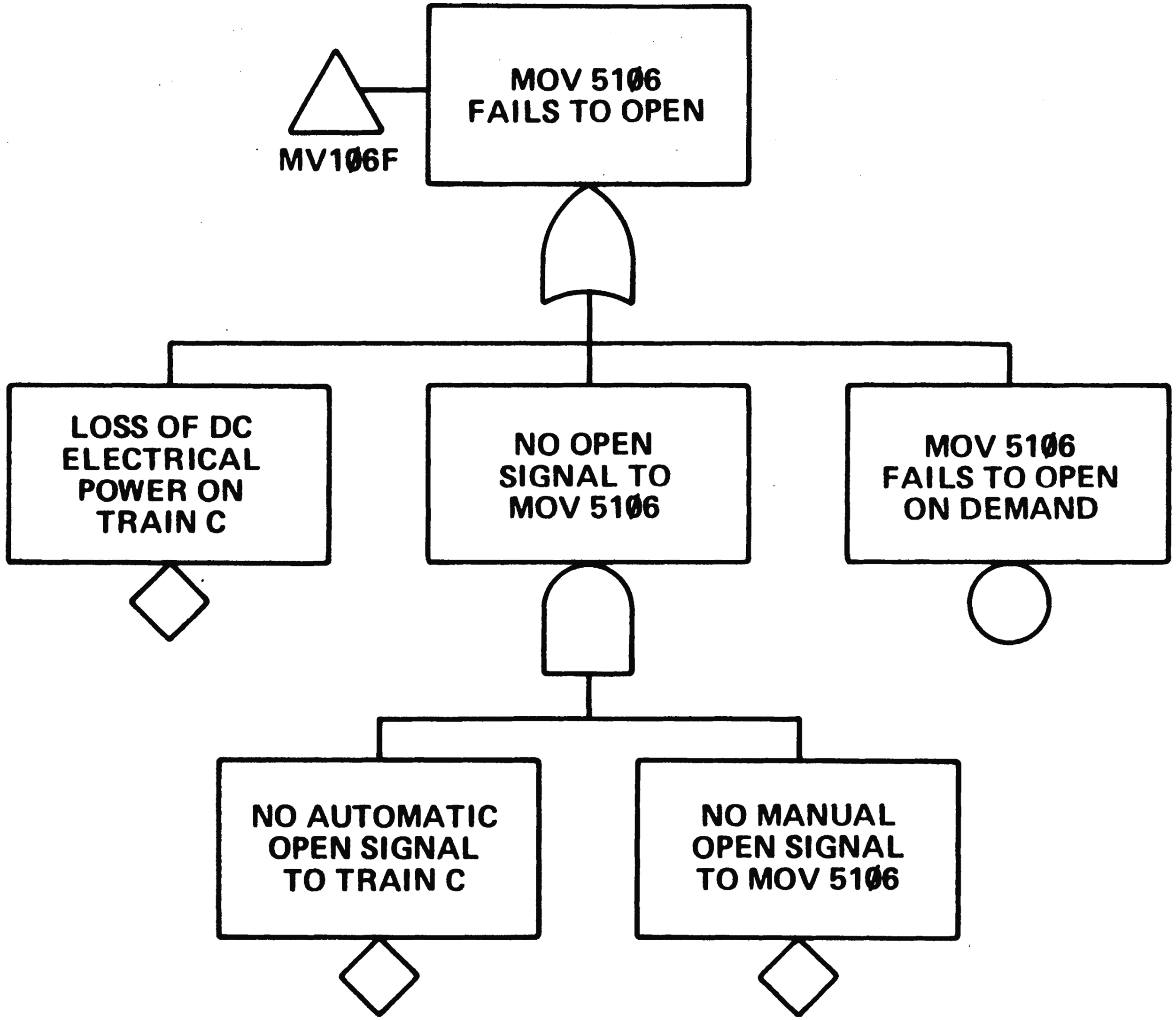
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

UNIT 1 AFWS FAULT TREE MODEL

FIGURE 10A-7 (SHEET 19 OF 30)



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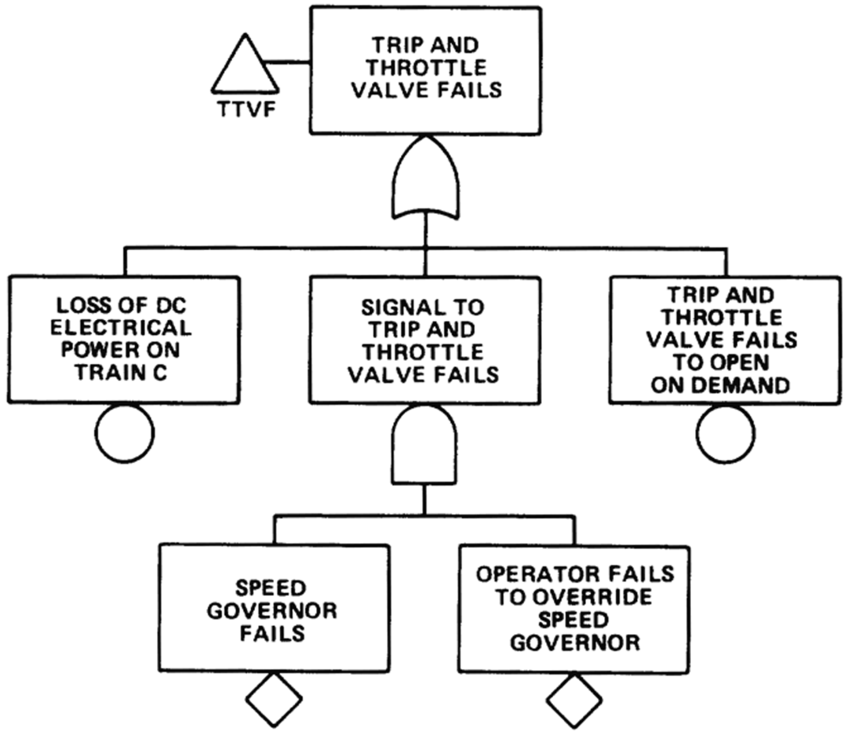
UNIT 1 AND UNIT 2

UNIT 1 AFWS FAULT TREE MODEL

FIGURE 10A-7 (SHEET 20 OF 30)



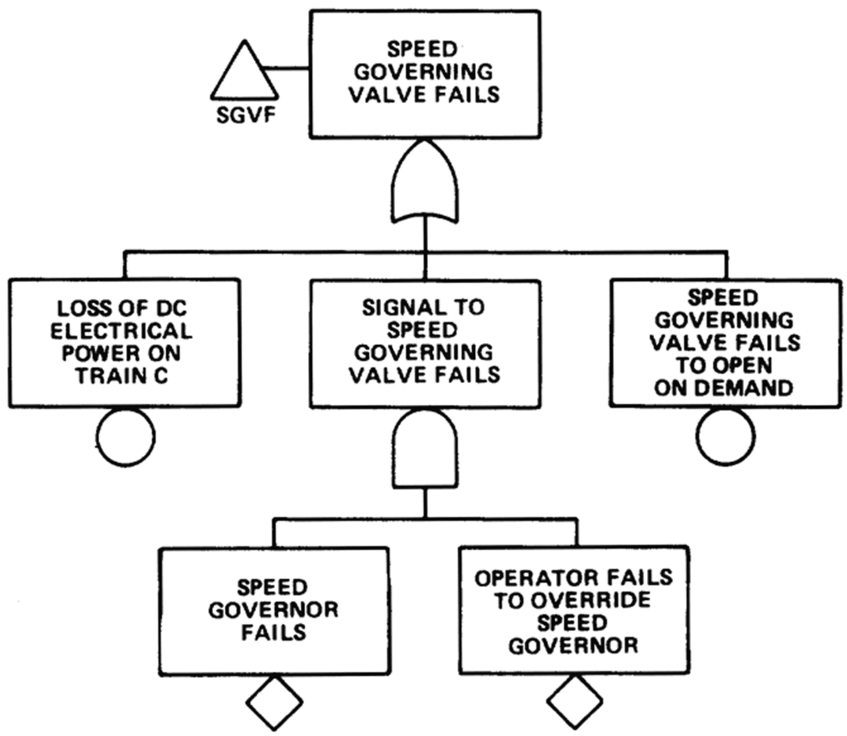


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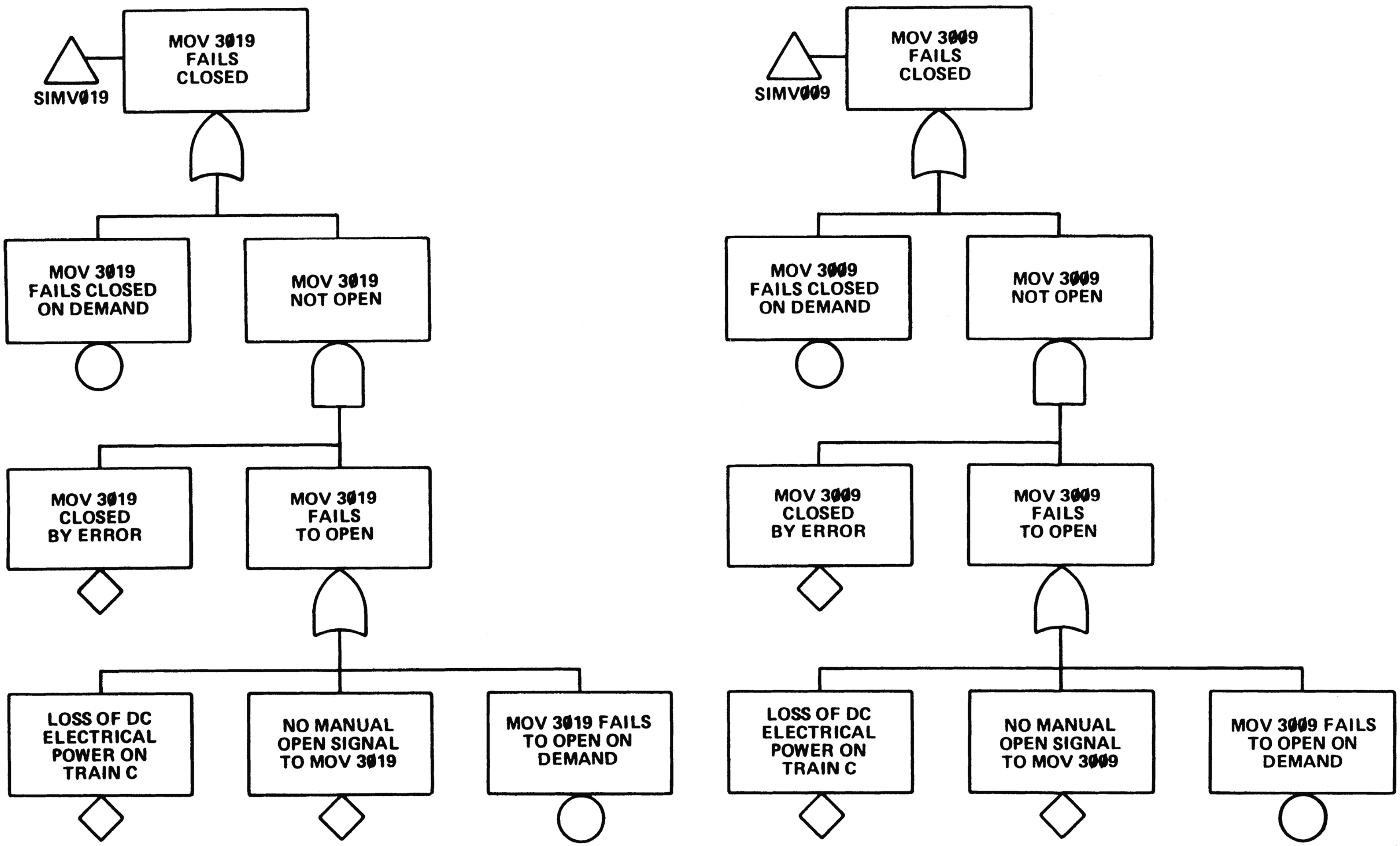
	FIGURE 10A-7 (SHEET 21 OF 30)



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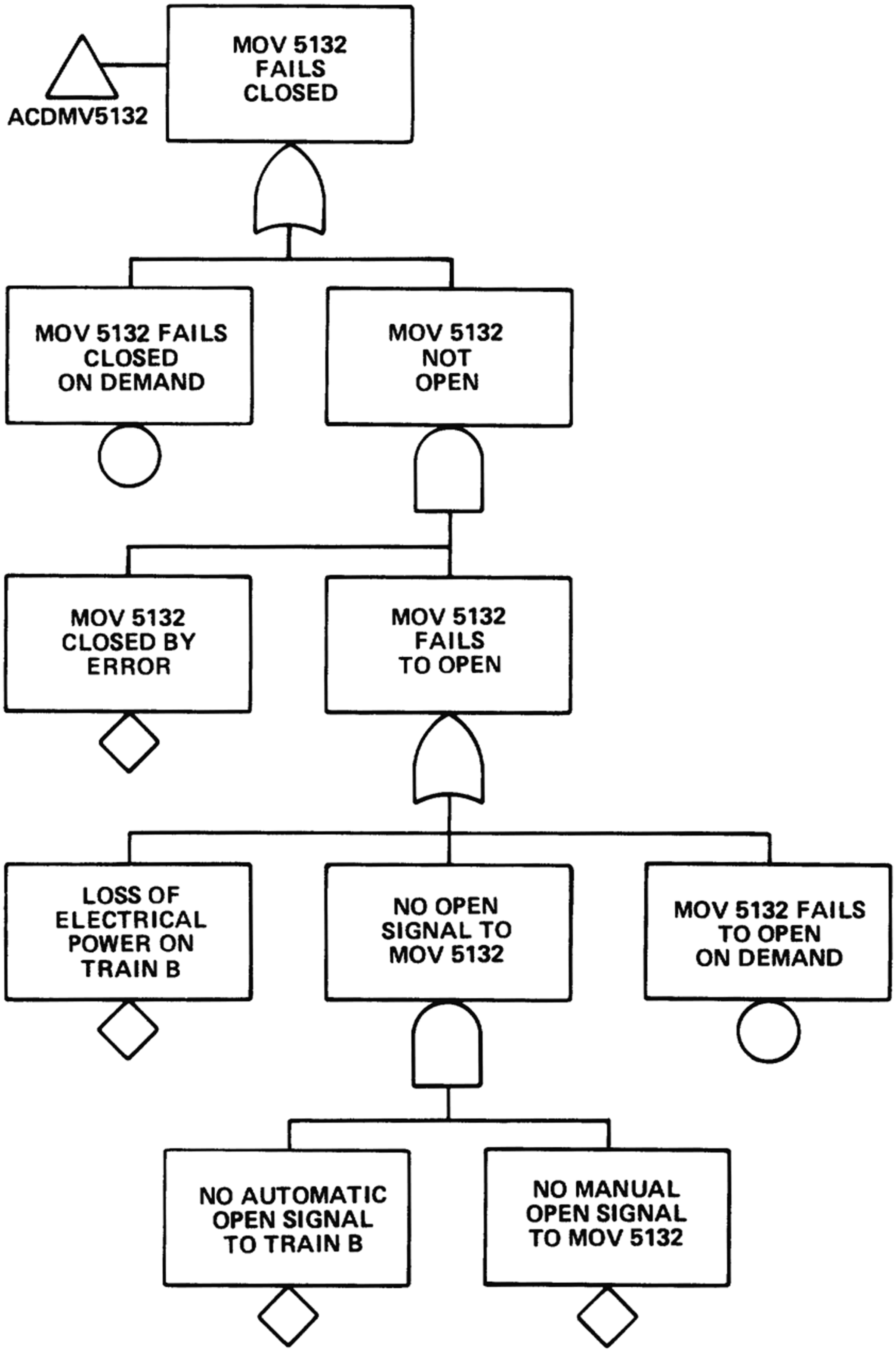
UNIT 1 AND UNIT 2

UNIT 1 AFWS FAULT TREE MODEL

FIGURE 10A-7 (SHEET 22 OF 30)


ACDMV5132
MOV 5132 FAILS CLOSED
MOV 5132 FAILS CLOSED ON DEMAND
MOV 5132 NOT OPEN
MOV 5132 CLOSED BY ERROR
MOV 5132 FAILS TO OPEN
LOSS OF ELECTRICAL POWER ON TRAIN B
NO OPEN SIGNAL TO MOV 5132
MOV 5132 FAILS TO OPEN ON DEMAND
NO AUTOMATIC OPEN SIGNAL TO TRAIN B
NO MANUAL OPEN SIGNAL TO MOV 5132

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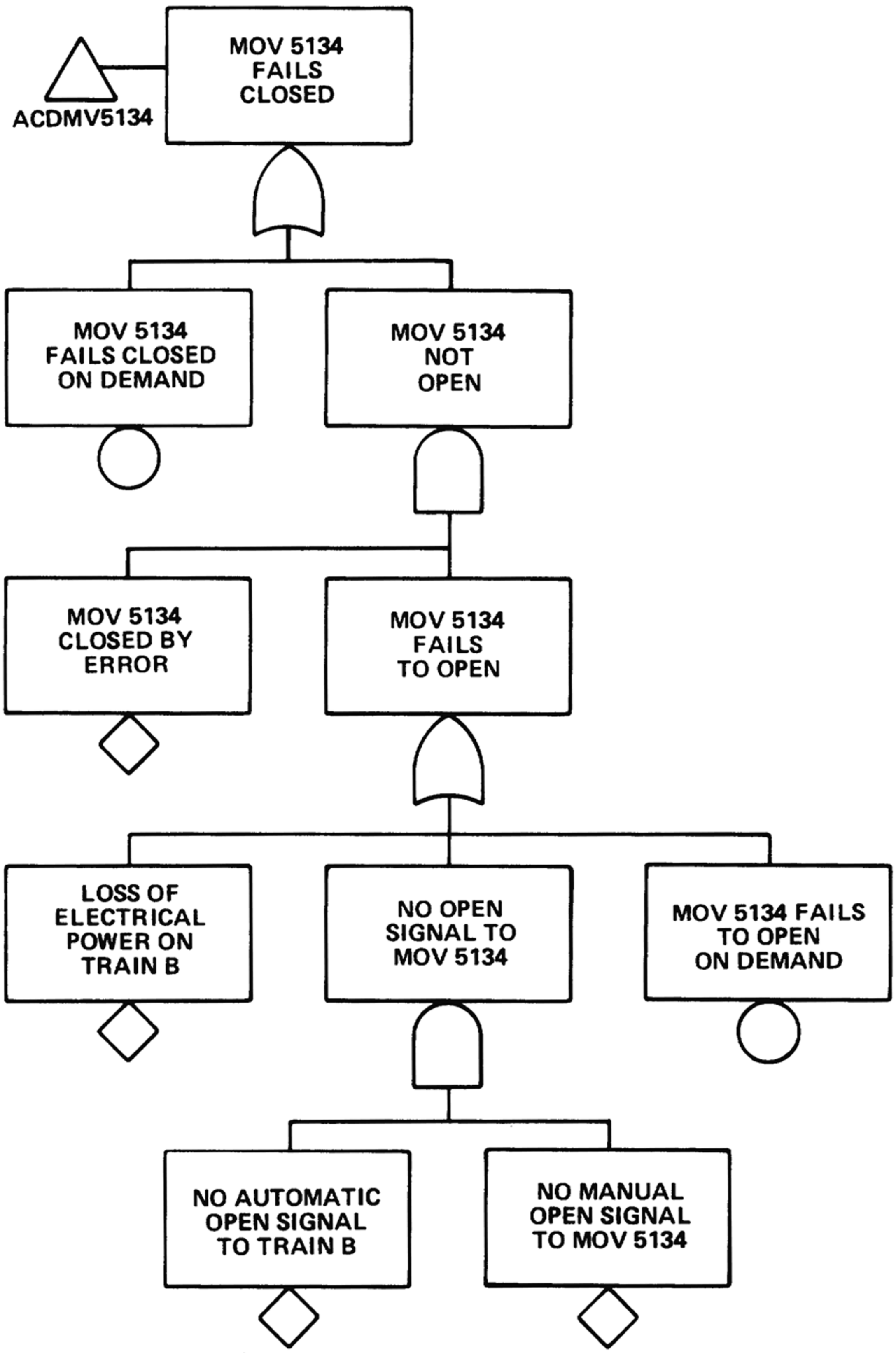

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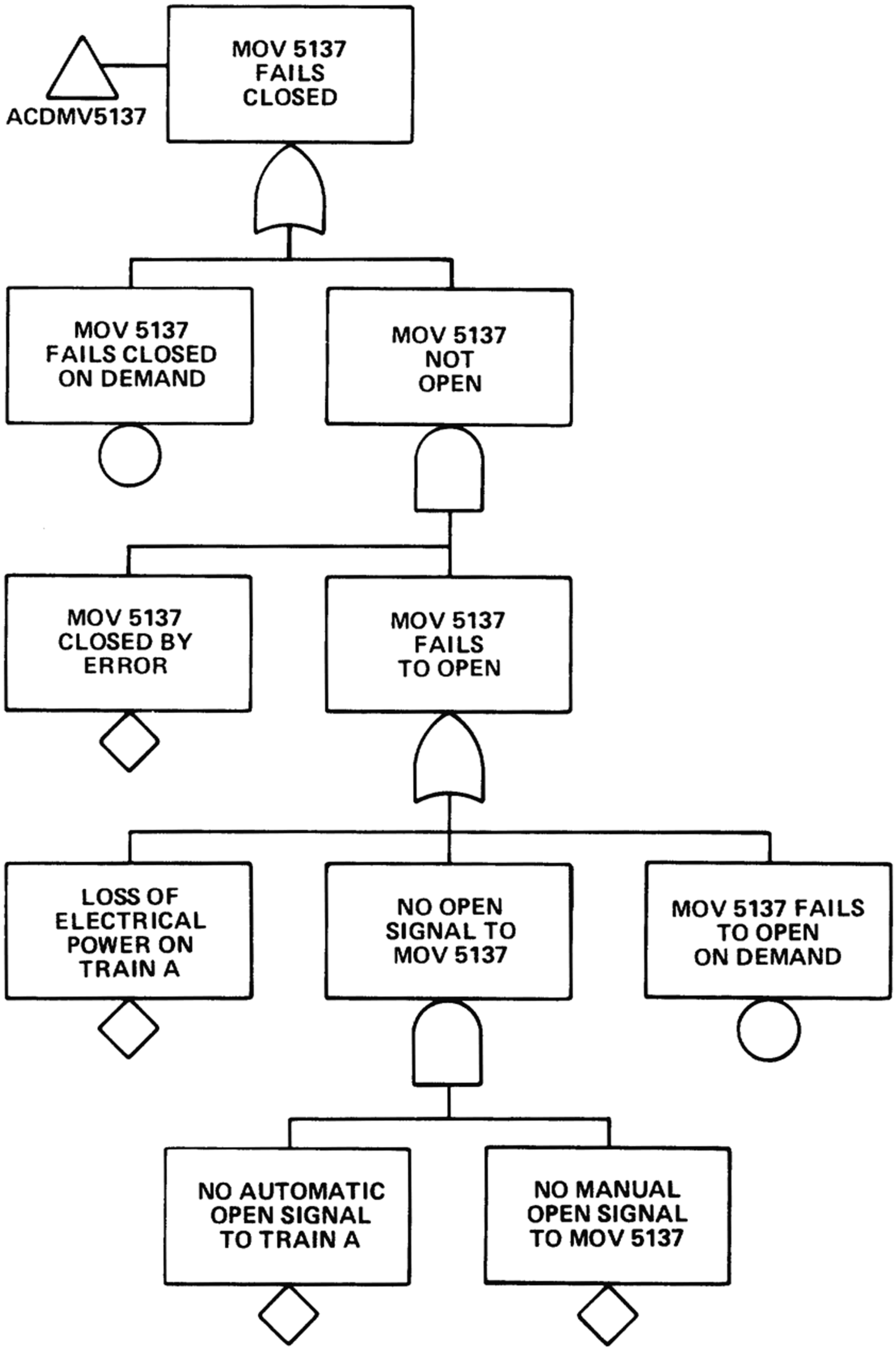
SOUTHERN COMPANY Energy to Serve Your World®	VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2	UNIT 1 AFWS FAULT TREE MODEL
		FIGURE 10A-7 (SHEET 24 OF 30)


MOV 5137
FAILS
CLOSED
ACDMV5137
MOV 5137
FAILS CLOSED
ON DEMAND
MOV 5137
NOT
OPEN
MOV 5137
CLOSED BY
ERROR
MOV 5137
FAILS
TO OPEN
LOSS OF
ELECTRICAL
POWER ON
TRAIN A
NO OPEN
SIGNAL TO
MOV 5137
MOV 5137 FAILS
TO OPEN
ON DEMAND
NO AUTOMATIC
OPEN SIGNAL
TO TRAIN A
NO MANUAL
OPEN SIGNAL
TO MOV 5137

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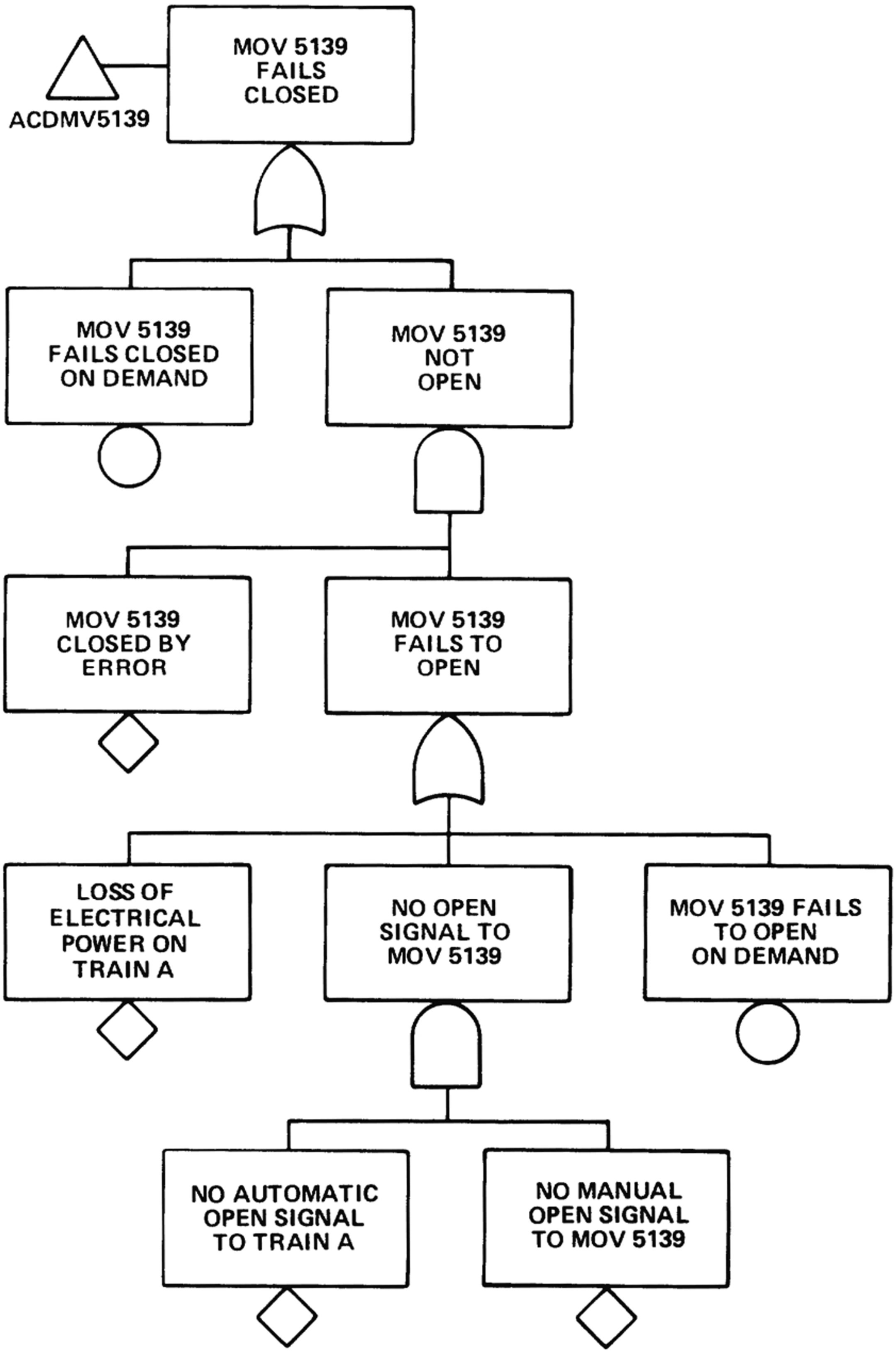
MOV 5139 FAILS CLOSED
ACDMV5139
MOV 5139 FAILS CLOSED ON DEMAND
MOV 5139 NOT OPEN
MOV 5139 CLOSED BY ERROR
MOV 5139 FAILS TO OPEN
LOSS OF ELECTRICAL POWER ON TRAIN A
NO OPEN SIGNAL TO MOV 5139
MOV 5139 FAILS TO OPEN ON DEMAND
NO AUTOMATIC OPEN SIGNAL TO TRAIN A
NO MANUAL OPEN SIGNAL TO MOV 5139

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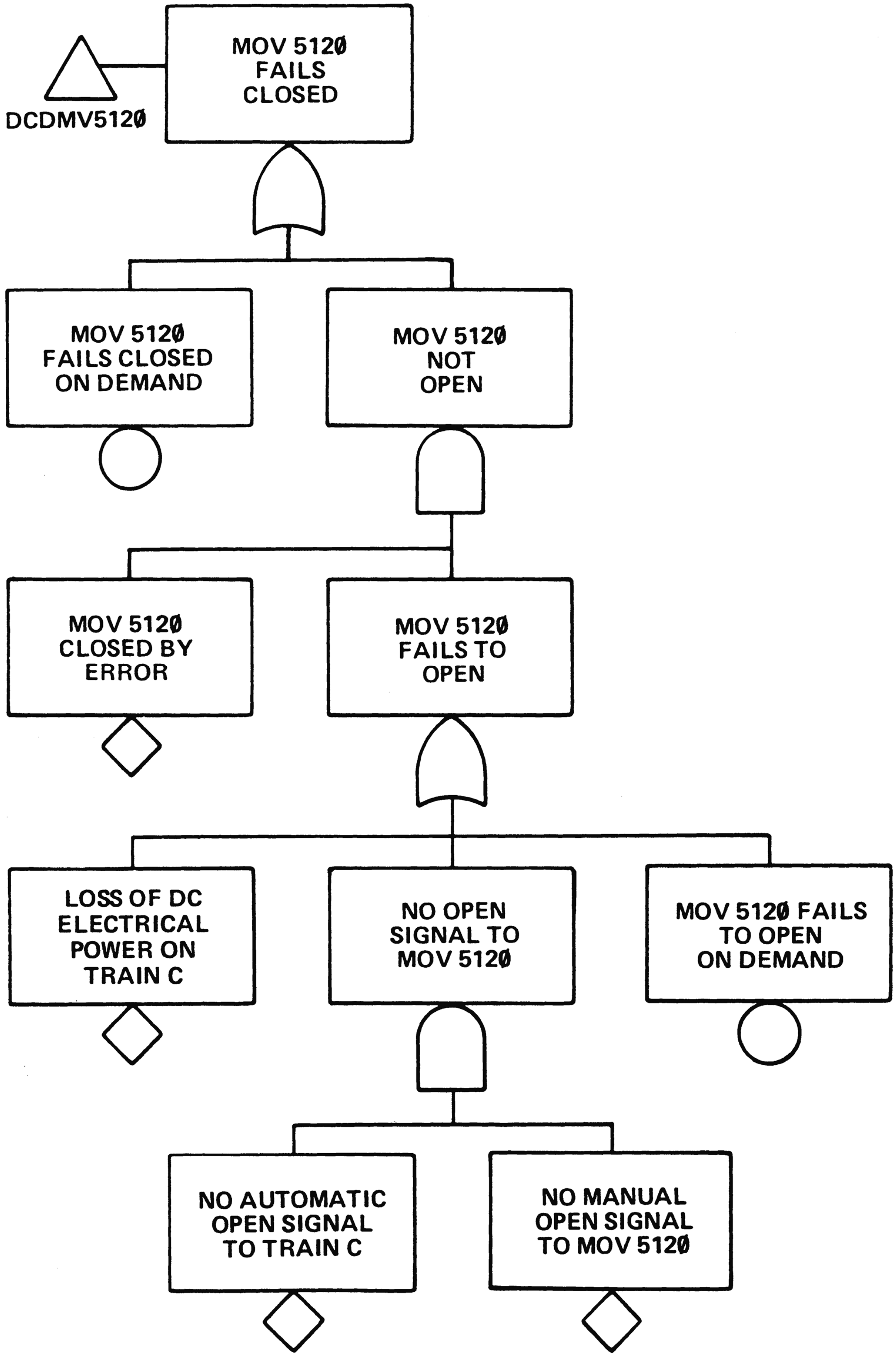
MOV 5120 FAILS CLOSED
DCDMV5120
MOV 5120 FAILS CLOSED ON DEMAND
MOV 5120 NOT OPEN
MOV 5120 CLOSED BY ERROR
MOV 5120 FAILS TO OPEN
LOSS OF DC ELECTRICAL POWER ON TRAIN C
NO OPEN SIGNAL TO MOV 5120
MOV 5120 FAILS TO OPEN ON DEMAND
NO AUTOMATIC OPEN SIGNAL TO TRAIN C
NO MANUAL OPEN SIGNAL TO MOV 5120

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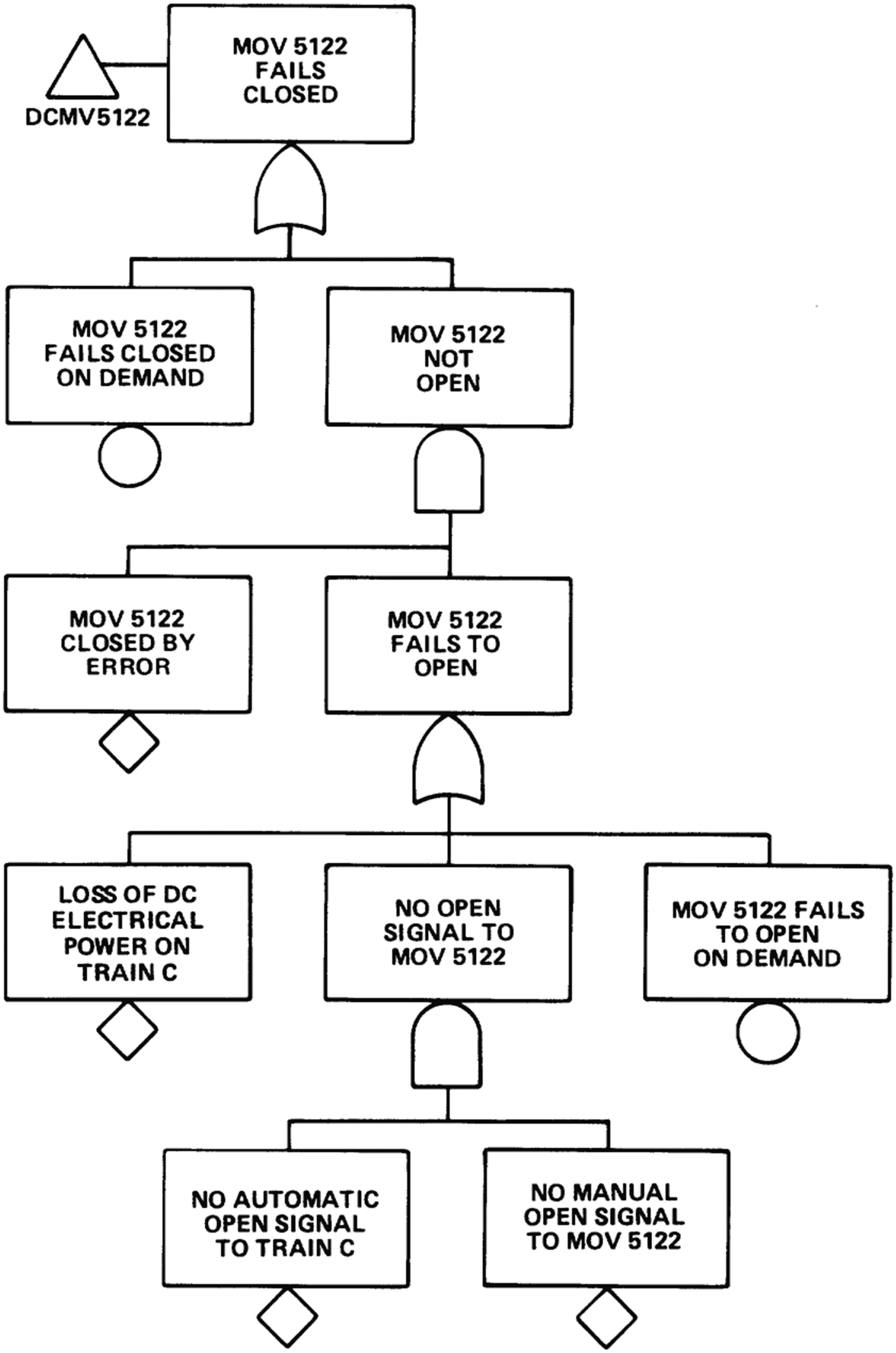

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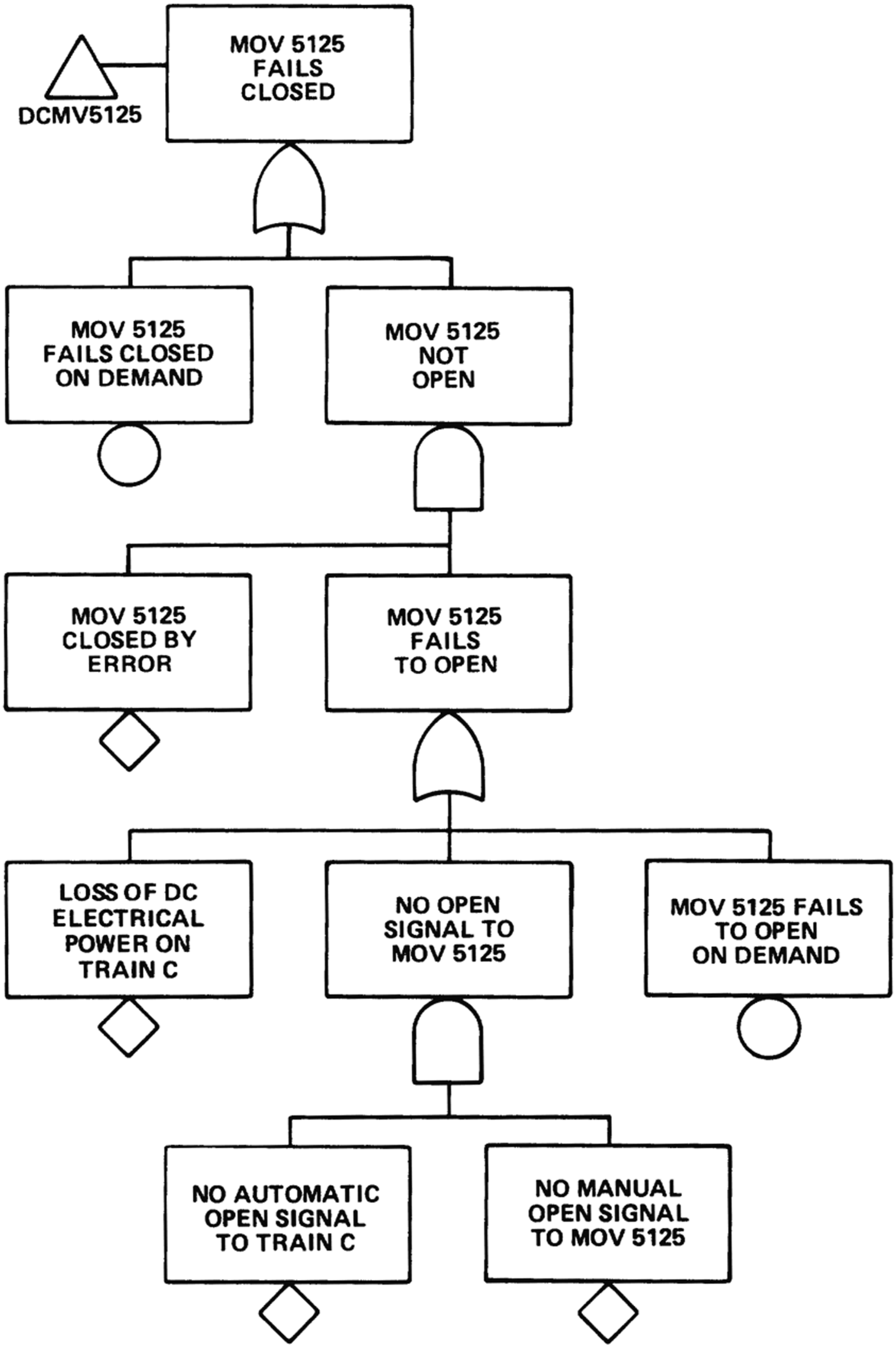
UNIT 1 AND UNIT 2

UNIT 1 AFWS FAULT TREE MODEL

FIGURE 10A-7 (SHEET 28 OF 30)



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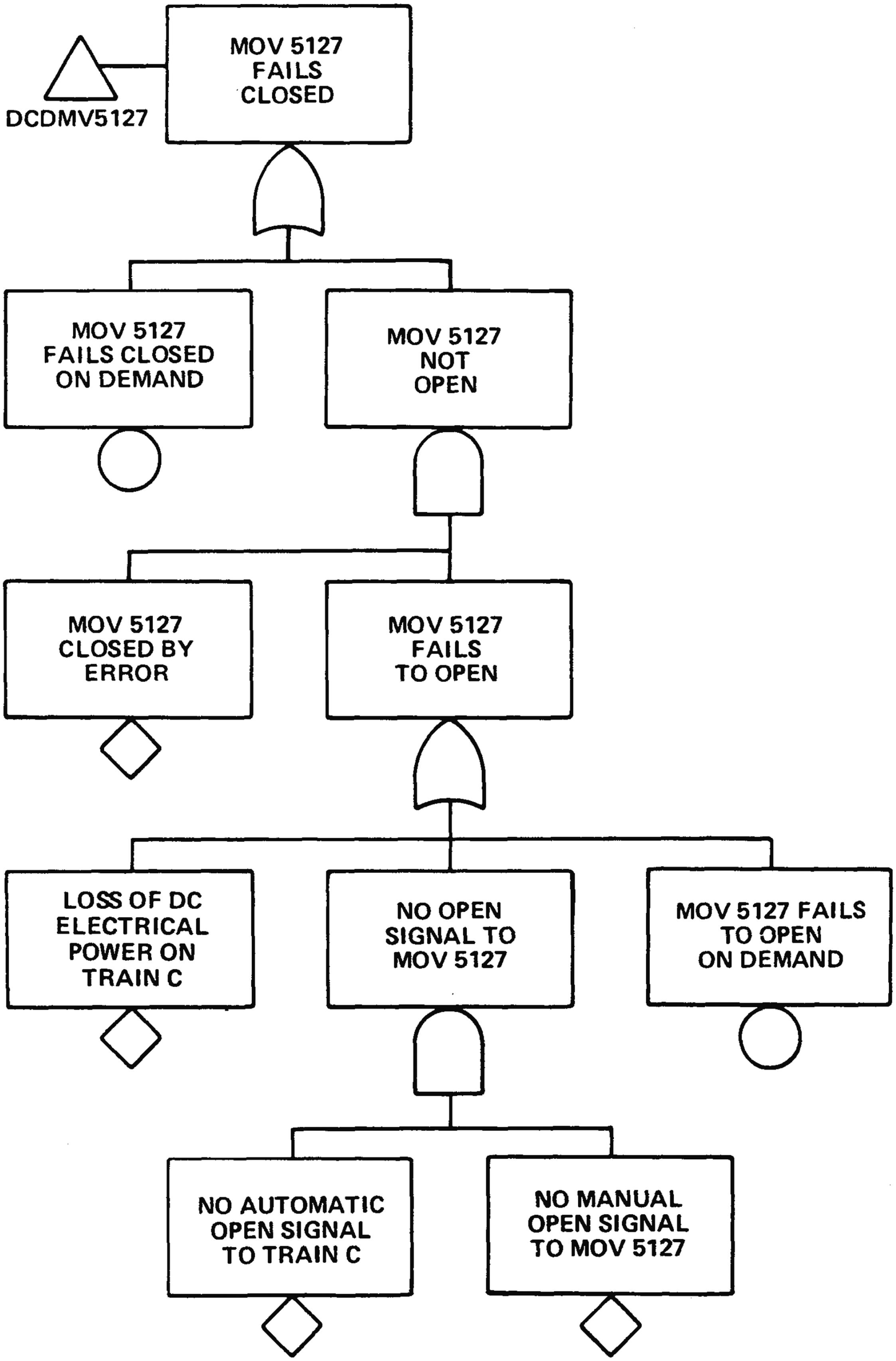
UNIT 1 AFWS FAULT TREE MODEL

FIGURE 10A-7 (SHEET 29 OF 30)


DCDMV5127
MOV 5127 FAILS CLOSED
MOV 5127 FAILS CLOSED ON DEMAND
MOV 5127 NOT OPEN
MOV 5127 CLOSED BY ERROR
MOV 5127 FAILS TO OPEN
LOSS OF DC ELECTRICAL POWER ON TRAIN C
NO OPEN SIGNAL TO MOV 5127
MOV 5127 FAILS TO OPEN ON DEMAND
NO AUTOMATIC OPEN SIGNAL TO TRAIN C
NO MANUAL OPEN SIGNAL TO MOV 5127

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CHAPTER 11.0 RADIOACTIVE WASTE MANAGEMENT

11.1 SOURCE TERMS

A discussion of the fission product inventory in the reactor core and the diffusion to the fuel pellet/cladding gap is presented in chapter 15. Models used in the preparation of the source terms are presented in references 1 and 2 and are based on operating plant data where available.

Two source terms are presented. The first is a conservative design base that utilizes a conventional fuel clad defect model. This design model serves as a basis for system and shielding requirements and calculations of the maximum offsite doses resulting from credible accidents.

The second source term is a realistic model used to predict expected long-term average concentrations of radionuclides in the primary and secondary fluid stream and an average plant's releases over its lifetime. This realistic model, based on available measured nuclide concentrations during normal operation, was formulated as a standard for the American National Standard Source Term Specifications, ANSI N237,[3] and is the source term model used in NUREG-0017.[4]

Recommended design bases for the liquid waste volumes originating from leakage or drainage of nuclear steam supply system (NSSS) components are presented in reference 1.

11.1.1 REACTOR COOLANT AND SECONDARY SIDE ACTIVITY

11.1.1.1 Design Basis Model

The parameters used in the calculation of the reactor coolant fission and corrosion product specific activities together with the pertinent information concerning the expected coolant cleanup flowrate, demineralizer effectiveness, and volume control tank noble gas stripping behavior are summarized in table 11.1-1. No purge is assumed in the volume control tank to the gaseous waste processing system. Calculated reactor coolant radionuclide concentrations are presented in table 11.1-2.

The time dependent fission product core inventories that serve as a source term to the reactor coolant system are calculated by the ORIGEN code[2] using a data library based on ENDF/B-IV.[5]

The specific activity for major nuclides in the pressurizer steam and liquid phases is given in table 11.1-3. The pressurizer liquid specific activity is assumed to be the same as that of the reactor coolant. Table 11.1-3 lists the nuclides that are the major contributors to total source strength.

The pressurizer nitrogen-16 activity calculations are based on an insurge to the pressurizer following a 10-percent step-load power decrease. It is assumed that the incoming reactor coolant mixes only with the pressurizer liquid below the first baffle (109 ft^3) and that the nitrogen-16 concentration is corrected for decay during transit through the surge line. With these assumptions, the pressurizer nitrogen-16 activity is found to be 12.9 μCi/g.

Pressurizer steam phase radiogas concentrations in table 11.1-3 are based on the stripping of radiogases from the continuous 2-gal/min pressurizer spray and the subsequent buildup of these radiogases in the steam space. The buildup time is assumed to be 1 effective full-power year. The radiogases are assumed to be completely stripped from the spray, with the exception of krypton-85, where a stripping fraction of 0.9 is used.

Pressurizer steam phase iodine concentrations in table 11.1-3 are obtained from the liquid phase nuclide activities and measured values of the partition coefficient for iodine-131. A partition coefficient of 100 is used and is assumed to apply to all radioiodines.

The specific activity for major nuclides in the volume control tank liquid and vapor phases is given in table 11.1-4.

The radiation sources in the volume control tank are based on a nominal operating level in the tank of 200 ft^3 in the liquid phase and 200 ft^3 in the vapor phase and on the stripping fractions given in table 11.1-1, assuming no volume control tank purge.

The design basis steam generator secondary side fission and corrosion product concentrations for the steam generator blowdown processing system are given in subsection 10.4.8. Design basis iodine concentrations in the steam generator secondary side based on Technical Specification maximum activity of 0.1 μCi/g dose equivalent iodine-131 are given in table 11.1-5.

11.1.1.2 Realistic Model

The parameters used to describe the realistic model are given in table 11.1-6 together with the range of values utilized by ANSI N237-1976.

Corrections have been made according to the ANSI N237-1976 standard formulas. Operation of a Westinghouse gaseous waste management system is assumed. The Y parameter is interpreted as equal to the stripping fraction and is calculated using the formula in table 11.1-1. These stripping fractions apply to the VEGP gaseous waste management system. Stripping fractions (Y parameter) are listed in table 11.1-7.

Specific activities in the primary coolant, steam generator water, and steam generator steam, based on the parameters of table 11.1-6, are also given in table 11.1-7.

Regulatory Guide 1.112, appendix B, recommends input parameters needed to execute the gaseous and liquid effluents (GALE) computer code[(4)] for pressurized water reactors. These values are listed in table 11.1-8.

11.1.2 TRITIUM PRODUCTION AND RELEASE TO THE REACTOR COOLANT

There are two principal contributors to tritium production within the NSSS: the ternary fission source and the dissolved boron in the reactor coolant. Additional contributions are made by lithium-6, lithium-7, and deuterium in the reactor water and by tritium produced in the B_4C section of the control rods. *(HISTORICAL): Tritium production from various sources is shown in table 11.1-9.*

Additional background information on tritium production is given in reference 1.

11.1.3 SPENT RESIN VOLUMES

The expected volume of spent demineralizer resin supplied to the solid waste management system from demineralizers is presented in table 11.1-10. The demineralizer resin source strengths are given in section 12.2. The information is based on plant experience as further outlined in reference 1.

11.1.4 REFERENCES

1. "Source Term Data for Westinghouse Pressurized Water Reactors," WCAP-8253, Revision 1, February 1976.

2. Bell, M. J., "ORIGEN - The ORNL Isotope Generation and Depletion Code," ORNL-4628, Oak Ridge National Laboratory, Oak Ridge, Tennessee, May 1973.

3. American National Standard Source Term Specification, ANSI N237-1976/AN8-18.1, approved May 11, 1976.

4. U.S. Nuclear Regulatory Commission, "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Pressurized Water Reactors," NUREG-0017 Office of Standard Development, April 1976.

5. "ORIGEN Yields and Cross Sections - Nuclear Transmutation and Decay Data From ENDF/B-IV," RSIC-DLC-38, Oak Ridge National Laboratory, Radiation Shielding Information Center, Oak Ridge, Tennessee.

TABLE 11.1-1 (SHEET 1 OF 3)

PARAMETERS USED IN THE CALCULATION OF DESIGN BASIS FISSION AND CORROSION PRODUCT ACTIVITIES

Ultimate core thermal power (MWt)	3636
Reactor coolant liquid volume (ft^3)	11,500
Reactor coolant full-power average temperature (°F)	590
Purification flowrate, normal (gal/min)[(a)]	75
Effective cation demineralizer flow (gal/min)[(a)]	7.5
Volume control tank	
Vapor volume (ft^3)	200
Liquid volume (ft^3)	200
Pressure (psig)	15
Temperature (°F)	115
Nuclide release coefficients (the product of the failed fuel fraction and the fission product escape rate coefficient)	
Failed fuel fractions:	
Equivalent fraction of core power produced by fuel rods containing small cladding defects	0.01
Fission product escape rate coefficients during full-power operation (s^{-1}):	
Kr and Xe isotopes	6.5×10^{-8}
Br, Rb, I, and Cs isotopes	1.3×10^{-8}
Mo, Tc, and Ag isotopes	2.0×10^{-9}
Te isotopes	1.0×10^{-9}
Sr and Ba isotopes	1.0×10^{-11}
Y, Zr, Nb, Ru, Rh, La, Ce, and Pr isotopes	1.6×10^{-12}
Chemical and volume control system mixed bed demineralizers	
Resin volume (ft^3)	30
Demineralizer isotopic decontamination factors[(b)]:	
Kr and Xe isotopes	1
Br and I isotopes	10
Sr and Ba isotopes	10
Other isotopes	1

TABLE 11.1-1 (SHEET 2 OF 3)

Chemical and volume control system cation bed demineralizer	
Resin volume (ft^3)	20
Demineralizer isotopic decontamination factors[(b)]:	
Kr and Xe isotopes	1
Sr and Ba isotopes	1
Rb-86, Cs-134, and Cs-137	10
Rb-88, Rb-89, Cs-136, and Cs-138	1
Other isotopes	1

Volume control tank noble gas stripping fractions (based on no purge to the gaseous waste management system)

Isotopic	Stripping Fraction[(c)]
Kr-83m	0.73
Kr-85m	0.53
Kr-85	5.3×10^{-5}
Kr-87	0.80
Kr-88	0.64
Kr-89	0.99
Xe-131m	0.012
Xe-133m	0.061
Xe-133	0.027
Xe-135m	0.93
Xe-135	0.27
Xe-137	0.98
Xe-138	0.94

Pressurizer volumes (ft^3)

Liquid volume	1080
Vapor volume	720

Initial boron concentrations (ppm)

First cycle	900
Second cycle	1070
Third cycle	1110
Equilibrium cycle	1110

Operation times (effective full-power hours)

First cycle	9240
Second cycle	6570
Third cycle	6700
Equilibrium cycle	7200

TABLE 11.1-1 (SHEET 3 OF 3)

a. Flow calculated at designed charging pump discharge (2250 psia and 130°F). The 75 gpm flow rate bounds operation at higher letdown flow rates.

b. For all isotopes, except the isotopes of Kr, Xe, Br, I, Rb, Cs, Sr, and Ba, a removal decontamination factor of 10 is assumed. This is done to account for removal mechanisms other than ion exchange, such as plateout, etc.

c. The nuclide stripping fractions are calculated using the following equation:

$$\psi = 1 - \frac{KQ}{KQ + \lambda\ (KL + V) + P}$$

where:

ψ = nuclide volume control tank stripping fraction.

$$K = \frac{RT}{MH}$$

$$R = \text{gas constant} = 45.59\ \frac{\text{atm/cm}^3}{\text{g mol/°R}}$$

T = nominal volume control tank temperature (°R).

M = molecular weight of water = 18.0 g/g mol.

H = Henry's Law constant equal to 3.02×10^4 atm/mol fraction for krypton and 2.04×10^4 atm/mol fraction for xenon, at 115°F.

Q = letdown or purification flowrate (g/s).

λ = nuclide decay constant (s^{-1}).

L = volume control tank liquid mass (g).

V = volume control tank vapor volume (cm^3).

P = volume control tank purge rate to the gaseous waste management system, at volume control tank conditions (cm^3/s).

TABLE 11.1-2 (SHEET 1 OF 2)

REACTOR COOLANT DESIGN BASIS FISSION AND CORROSION PRODUCT SPECIFIC ACTIVITY

Nuclide	Activity (μCi/g)	Nuclide	Activity (μCi/g)
Kr-83m	4.6×10^{-1}	Cs-134	2.3
Kr-85m	2.0	Cs-136	2.9
Kr-85	7.3	Cs-137	1.5
Kr-87	1.3	Ba-137m	1.4
Kr-88	3.6	Cs-138	9.6×10^{-1}
Kr-89	1.1×10^{-1}	H-3	3.5 (maximum)
Xe-131m	2.2	Cr-51	5.5×10^{-3}
Xe-133m	1.7×10^{1}	Mn-54	4.4×10^{-4}
Xe-133	2.7×10^{2}	Mn-56	2.0×10^{-2}
Xe-135m	4.8×10^{-1}	Fe-55	2.0×10^{-3}
Xe-135	7.2	Fe-59	5.2×10^{-4}
Xe-137	1.7×10^{-1}	Co-58	1.5×10^{-2}
Xe-138	6.4×10^{-1}	Co-60	1.9×10^{-3}
Br-83	9.5×10^{-2}	Sr-89	4.3×10^{-3}
Br-84	4.7×10^{-2}	Sr-90	1.2×10^{-4}
Br-85	6.0×10^{-3}	Sr-91	6.2×10^{-3}
I-127[a]	6.2×10^{-11}	Sr-92	1.3×10^{-3}
I-129	4.3×10^{-8}	Y-90	3.4×10^{-5}
I-130	2.1×10^{-2}	Y-91m	3.3×10^{-3}
I-131	2.8	Y-91	5.7×10^{-4}
I-132	2.8	Y-92	1.2×10^{-3}
I-133	4.2	Y-93	3.8×10^{-4}
I-134	5.7×10^{-1}	Zr-95	6.5×10^{-4}
I-135	2.3	Nb-95	6.5×10^{-4}
Rb-86	2.2×10^{-2}	Mo-99	7.5×10^{-1}
Rb-88	4.8	Tc-99m	6.9×10^{-1}
Rb-89	2.1×10^{-1}	Ru-103	5.7×10^{-4}
Ru-106	1.4×10^{-4}	Te-131	1.2×10^{-2}
Rh-103m	5.7×10^{-4}	Te-132	2.9×10^{-1}
Rh-106	1.4×10^{-4}	Te-134	3.0×10^{-2}
Ag-110m	1.4×10^{-3}	Ba-140	4.2×10^{-3}
Te-125m	2.8×10^{-4}	La-140	1.4×10^{-3}
Te-127m	2.9×10^{-3}	Ce-141	6.3×10^{-4}
Te-127	1.2×10^{-2}	Ce-143	5.0×10^{-4}
Te-129m	1.9×10^{-2}	Ce-144	3.9×10^{-4}
Te-129	1.8×10^{-2}	Pr-143	6.3×10^{-4}
Te-131m	2.6×10^{-2}	Pr-144	3.9×10^{-4}

TABLE 11.1-2 (SHEET 2 OF 2)

a. Grams of iodine-127 per gram of coolant.

The above concentrations are based on the assumptions listed below:

1. Reactor coolant mass = 2.3×10^8 g.

2. Operation with small defects in the cladding of fuel rods generating 1 percent of the core-rated power.

3. Reactor coolant purification or letdown rate = 75 gal/min at 130°F and 2250 psia.

4. No volume control tank purge.

TABLE 11.1-3

PRESSURIZER ACTIVITIES

Liquid Phase Specific Activity (1080-ft^3 liquid phase)

Nuclide	Activity (μCi/g)
Kr-88	3.7
Kr-89	1.1×10^{-1}
Xe-133	2.7×10^{2}
Xe-135	7.3
I-132	2.8
I-133	4.2
I-135	2.3
Rb-88	4.8
Cs-134	2.3
Cs-136	2.9
Cs-138	9.6×10^{-1}

Steam Phase Specific Activity (720-ft^3 steam phase)

Nuclide	Activity (μCi/g)
Kr-83m	2.0×10^{-2}
Kr-85m	2.2×10^{-1}
Kr-85	9.3×10^{2}
Kr-87	3.8×10^{-2}
Kr-88	2.4×10^{-1}
Kr-89	1.3×10^{-4}
Xe-131m	1.5×10^{1}
Xe-133m	2.2×10^{1}
Xe-133	8.1×10^{2}
Xe-135m	2.9×10^{-3}
Xe-135	1.6
Xe-137	2.6×10^{-4}
Xe-138	3.6×10^{-3}
I-129	4.3×10^{-10}
I-130	2.1×10^{-4}
I-131	2.8×10^{-2}
I-132	2.8×10^{-2}
I-133	4.2×10^{-2}
I-134	5.7×10^{-3}
I-135	2.3×10^{-2}

TABLE 11.1-4

VOLUME CONTROL TANK ACTIVITIES[a]

Liquid Phase Specific Activity (200-ft^3 liquid phase)

Nuclide	Activity (μCi/g)
Kr-88	1.3
Xe-133	2.6×10^{2}
Xe-135	5.3
Rb-88	4.8
Cs-134	2.3
Cs-136	2.9
Cs-138	9.6×10^{-1}

Vapor Phase Specific Activity (200-ft^3 steam phase)

Nuclide	Activity (μCi/g)
Kr-83m	3.7
Kr-85m	2.1×10^{1}
Kr-85	1.5×10^{2}
Kr-87	5.5
Kr-88	3.0
Kr-89	2.4×10^{-2}
Xe-131m	3.2×10^{1}
Xe-133m	2.5×10^{2}
Xe-133	3.9×10^{3}
Xe-135m	8.5
Xe-135	9.5×10^{1}
Xe-137	4.8×10^{-2}
Xe-138	6.2×10^{-1}

a. Based on parameters given in table 11.1-1.

TABLE 11.1-5

DESIGN BASIS IODINE CONCENTRATIONS IN SECONDARY SIDE WATER AT TECHNICAL SPECIFICATION LIMITS

Isotope	Concentration (μCi/g)
I-131	8.0×10^{-2}
I-132	2.8×10^{-2}
I-133	9.0×10^{-2}
I-134	2.0×10^{-3}
I-135	3.3×10^{-2}

TABLE 11.1-6

PARAMETERS USED TO DESCRIBE THE REACTOR SYSTEM - REALISTIC BASIS

Parameter	Symbol	Units	Value	ANSI N237 Range Maximum	ANSI N237 Range Minimum
Thermal power	P	MWt	3636	3800	3000
Steam flowrate	FS	lb/h	1.5×10^7	1.7×10^7	1.3×10^7
Weight of water in reactor coolant system	WP	lb	5.6×10^5	6.0×10^5	5.0×10^5
Weight of water in all steam generators	WS	lb	4.7×10^5	5.0×10^5	4.0×10^5
Reactor coolant letdown flow (purification)	FD	lb/h	3.8×10^4	4.2×10^4	3.2×10^4
Reactor coolant letdown flow (yearly average for boron control)	FB	lb/h	100	1.0×10^3	2.5 x 10
Steam generator blowdown flow (total)	FBD	lb/h	1.8×10^5	1.0×10^5	5.0×10^4
Fraction of radioactivity in blowdown steam that is not returned to the secondary coolant system	NBD	-	1.0	1.0	0.9
Flow through the purification system cation demineralizer	FA	lb/h	3.8×10^3	7.5×10^4	0.0
Ratio of condensate demineralizer flowrate to the total stream flowrate	NC	-	0.0	0.01	0.0
Ratio of the total amount of noble gases routed to gaseous radwaste from the purification system to the total amount routed from the primary coolant system (not including the boron recycle system)	Y	-	See table 11.1-7	0.01	0.0
Primary-to-secondary leak rate	-	lb/day	100	-	100

TABLE 11.1-7 (SHEET 1 OF 3)

SPECIFIC ACTIVITIES IN PRINCIPAL FLUID STREAMS - REALISTIC BASIS

Normal Plant Operation Source Terms
(based on ANSI N237)

Volume Control Tank Purge of 1.736×10^{2} cm^3/s

Group I - Noble Gases

Nuclide	Y Parameter	Reactor Coolant Activity (μCi/g)	Steam Gen. Liq. Activity (μCi/g)	Steam Gen. Steam Activity (μCi/g)
Kr-83m	7.8×10^{-1}	2.1×10^{-2}	Nil	5.7×10^{-9}
Kr-85m	6.5×10^{-1}	9.6×10^{-2}	Nil	2.7×10^{-8}
Kr-85	4.3×10^{-1}	4.8×10^{-3}	Nil	1.3×10^{-9}
Kr-87	8.2×10^{-1}	6.2×10^{-2}	Nil	1.6×10^{-8}
Kr-88	7.2×10^{-1}	1.9×10^{-1}	Nil	5.1×10^{-8}
Kr-89	9.9×10^{-1}	5.7×10^{-3}	Nil	1.6×10^{-9}
Xe-131m	3.5×10^{-1}	1.5×10^{-2}	Nil	4.1×10^{-9}
Xe-133m	3.7×10^{-1}	8.6×10^{-2}	Nil	2.4×10^{-8}
Xe-133	3.5×10^{-1}	4.1×10^{-1}	Nil	1.1×10^{-7}
Xe-135m	9.3×10^{-1}	1.4×10^{-2}	Nil	4.0×10^{-9}
Xe-135	4.7×10^{-1}	2.8×10^{-1}	Nil	7.6×10^{-8}
Xe-137	9.8×10^{-1}	1.0×10^{-2}	Nil	2.8×10^{-9}
Xe-138	9.4×10^{-1}	4.9×10^{-2}	Nil	1.3×10^{-8}

Group II - Halogens

Nuclide	Reactor Coolant Activity (μCi/g)	Steam Gen. Liq. Activity (μCi/g)	Steam Gen. Steam Activity (μCi/g)
Br-83	5.4×10^{-3}	1.2×10^{-7}	1.2×10^{-9}
Br-84	3.0×10^{-3}	2.0×10^{-8}	2.0×10^{-10}
Br-85	3.4×10^{-4}	2.2×10^{-10}	2.2×10^{-12}
I-130	2.3×10^{-3}	1.1×10^{-7}	1.1×10^{-9}
I-131	2.8×10^{-1}	2.2×10^{-5}	2.2×10^{-7}
I-132	1.1×10^{-1}	9.1×10^{-6}	9.1×10^{-8}
I-133	4.0×10^{-1}	2.4×10^{-5}	2.4×10^{-7}
I-134	5.3×10^{-2}	5.5×10^{-8}	5.5×10^{-9}
I-135	2.1×10^{-1}	8.3×10^{-6}	8.3×10^{-8}

TABLE 11.1-7 (SHEET 2 OF 3)

Group III - Rubidium and Cesium

Nuclide	Reactor Coolant Activity (μCi/g)	Steam Gen. Liq. Activity (μCi/g)	Steam Gen. Steam Activity (μCi/g)
Rb-86	8.9×10^{-5}	7.4×10^{-9}	7.4×10^{-12}
Rb-88	2.3×10^{-1}	8.3×10^{-7}	8.3×10^{-10}
Cs-134	2.6×10^{-2}	1.7×10^{-6}	1.7×10^{-9}
Cs-136	1.4×10^{-2}	9.5×10^{-7}	9.5×10^{-10}
Cs-137	1.9×10^{-2}	1.4×10^{-6}	1.4×10^{-9}
Ba-137m	1.8×10^{-2}	1.3×10^{-6}	1.3×10^{-9}

Group IV - Nitrogen-16

Nuclide	Reactor Coolant Activity (μCi/g)	Steam Gen. Liq. Activity (μCi/g)	Steam Gen. Steam Activity (μCi/g)
N-16	4.0×10^{1}	9.5×10^{-7}	9.5×10^{-7}

Group V - Tritium

Nuclide	Reactor Coolant Activity (μCi/g)	Steam Gen. Liq. Activity (μCi/g)	Steam Gen. Steam Activity (μCi/g)
H-3	1.0	1.0×10^{-3}	1.0×10^{-3}

TABLE 11.1-7 (SHEET 3 OF 3)

Group VI - Miscellaneous Isotopes

Nuclide	Reactor Coolant Activity (μCi/g)	Steam Gen. Liq. Activity (μCi/g)	Steam Gen. Steam Activity (μCi/g)
Cr-51	1.9×10^{-3}	1.4×10^{-7}	1.4×10^{-10}
Mn-54	3.1×10^{-4}	3.4×10^{-8}	3.4×10^{-11}
Fe-55	1.6×10^{-3}	1.2×10^{-8}	1.2×10^{-10}
Fe-59	1.0×10^{-3}	8.7×10^{-8}	8.7×10^{-11}
Co-58	1.6×10^{-2}	1.2×10^{-6}	1.2×10^{-9}
Co-60	2.0×10^{-3}	1.5×10^{-7}	1.5×10^{-10}
Sr-89	3.6×10^{-4}	3.5×10^{-8}	3.5×10^{-11}
Sr-90	1.0×10^{-5}	8.4×10^{-10}	8.4×10^{-13}
Sr-91	7.0×10^{-4}	3.1×10^{-8}	3.1×10^{-11}
Y-90	1.2×10^{-6}	4.8×10^{-10}	4.8×10^{-13}
Y-91m	4.1×10^{-4}	2.9×10^{-8}	2.9×10^{-11}
Y-91	6.5×10^{-5}	5.2×10^{-9}	5.2×10^{-12}
Y-93	3.7×10^{-5}	2.0×10^{-9}	2.0×10^{-12}
Zr-95	6.1×10^{-5}	5.2×10^{-9}	5.2×10^{-12}
Nb-95	5.1×10^{-5}	5.2×10^{-9}	5.2×10^{-12}
Mo-99	8.7×10^{-2}	7.2×10^{-6}	7.2×10^{-9}
Tc-99m	5.2×10^{-2}	1.8×10^{-5}	1.8×10^{-8}
Ru-103	4.6×10^{-5}	3.5×10^{-9}	3.5×10^{-12}
Ru-106	1.0×10^{-5}	8.5×10^{-10}	8.5×10^{-13}
Rh-103m	5.1×10^{-5}	1.9×10^{-8}	1.9×10^{-11}
Rh-106	1.1×10^{-5}	5.4×10^{-9}	5.4×10^{-12}
Te-125m	2.9×10^{-5}	1.6×10^{-9}	1.6×10^{-12}
Te-127m	2.8×10^{-4}	1.5×10^{-8}	1.5×10^{-11}
Te-127	9.2×10^{-4}	1.0×10^{-7}	1.0×10^{-10}
Te-129m	1.4×10^{-3}	1.1×10^{-7}	1.1×10^{-10}
Te-129	1.8×10^{-3}	5.6×10^{-7}	5.6×10^{-10}
Te-131m	2.6×10^{-3}	1.6×10^{-7}	1.6×10^{-10}
Te-131	1.2×10^{-3}	5.1×10^{-7}	5.1×10^{-10}
Te-132	2.8×10^{-2}	1.8×10^{-6}	1.8×10^{-9}
Ba-140	2.2×10^{-4}	1.7×10^{-8}	1.7×10^{-11}
La-140	1.6×10^{-4}	2.2×10^{-8}	2.2×10^{-11}
Ce-141	7.1×10^{-5}	5.3×10^{-9}	5.3×10^{-12}
Ce-143	4.2×10^{-5}	2.7×10^{-9}	2.7×10^{-12}
Ce-144	3.3×10^{-5}	3.4×10^{-9}	3.4×10^{-12}
Pr-143	5.1×10^{-5}	3.7×10^{-9}	3.7×10^{-12}
Pr-144	3.8×10^{-5}	2.1×10^{-8}	3.1×10^{-11}
Np-239	1.2×10^{-3}	7.5×10^{-8}	7.5×10^{-11}

TABLE 11.1-8 (SHEET 1 OF 3)

PARAMETERS SPECIFIED BY REGULATORY GUIDE 1.112 APPENDIX B
(INPUT PARAMETERS FOR THE GALE COMPUTER CODE)

Description	Value
Thermal power level (MWt)	3626
Mass of primary coolant (lb)	5.1×10^5
Primary system letdown rate (gal/min)	75[(a)]
Letdown cation demineralizer flowrate (gal/min)	7.5
Number of steam generators	4
Total steam flow (lb/h)	16.25×10^6
Mass of steam in each steam generator (lb)	6.4×10^3
Mass of liquid in each steam generator (lb)	1.11×10^5
Total mass of secondary coolant	2.022×10^6
Total blowdown rate (lb/h)	1.8×10^5
Condensate demineralizer regeneration time	0.0
Condensate demineralizer flow fraction	0.0
Maximum radwaste dilution flow (gal/min)	15.0×10^3
Shim Bleed	
Shim bleed flowrate (gal/day)	1.7×10^3
Decontamination factor for I	10^6
Decontamination factor for Cs and Rb	1.6×10^4
Decontamination factor for others	10^8
Collection time (day)	22.4
Process and discharge time (day)	0.31
Fraction discharged	1.0
Equipment Drains	
Equipment drains flowrate (gal/day)	300
Fraction of reactor coolant activity	1.0
Decontamination factor for I	10^5
Decontamination factor for Cs and Rb	8×10^3
Decontamination factor for others	1×10^7
Collection time (day)	22.4
Process and discharge time (day)	2.1
Fraction discharged	0.25
Clean Waste	
Clean waste input flowrate (gal/day)	7.13×10^2
Fraction of reactor coolant activity	0.051
Decontamination factor for I	10^4
Decontamination factor for Cs and Rb	4×10^3

TABLE 11.1-8 (SHEET 2 OF 3)

Description	Value
Decontamination factor for others	1×10^6
Collection time (day)	5.6
Process and discharge time (day)	0.03
Fraction discharged	1.0
Dirty Waste	
Dirty waste input flowrate (gal/day)	2.047×10^3
Fraction of reactor coolant activity	0.02
Decontamination factor for I	10^4
Decontamination factor for Cs and Rb	4×10^3
Decontamination factor for others	1×10^6
Collection time (day)	1.95
Process and discharge time (day)	0.19
Fraction discharged	1.0
Blowdown Waste	
Blowdown fraction processed	1.0
Decontamination factor for I	10^3
Decontamination factor for Cs and Rb	10^2
Decontamination factor for others	10^3
Collection time	0.0
Process and discharge time	0.0
Fraction discharged	0.1
Regenerant flowrate	
Decontamination factor for I	N/A
Decontamination factor for Cs and Rb	N/A
Decontamination factor for others	N/A
Collection time	N/A
Process and discharge time	N/A
Fraction discharged	N/A

TABLE 11.1-8 (SHEET 3 OF 3)

Description	Value
Gaseous Waste System	
Holdup time for xenon (day)	90
Holdup time for krypton (day)	90
Fill time of decay tanks for gas stripper	0.0
Gas waste system: HEPA?	Yes
Auxiliary building: charcoal?	Yes
Auxiliary building: HEPA?	Yes
Containment volume (ft^3)	2.75×10^6
Containment atmosphere cleanup rate (ft^3/min)	30×10^3
Containment shutdown purge: charcoal?, HEPA?	Yes, yes
Number purge per year	4
Containment normal purge rate (ft^3/min); charcoal?, HEPA?	5000; yes, yes
Fraction of iodine released from blowdown tank vent	0
Fraction of iodine released from main condenser air ejector	1.0
Detergent waste decontamination factor	1.0

a. The primary system letdown rate of 75 gpm results in effluent releases and doses which bound those for higher letdown rates to the maximum design flow of 130 gpm.

TABLE 11.1-9

TRITIUM PRODUCTION[a] (HISTORICAL)

Tritium Source	*Total Produced (Ci/cycle)*	*Release Expected to Reactor Coolant (Ci/cycle)*
Ternary fissions		
Initial cycle	*14,000*	*1400*
Equilibrium cycle	*10,900*	*1090*
Burnable poison rods		
Initial cycle	*1950*	*195*
Control rods	*70*	*70*
Coolant (soluble boron)		
Initial cycle	*388*	*388*
Equilibrium cycle	*285*	*285*
Coolant (lithium, deuterium)		
Initial cycle	*141*	*141*
Equilibrium cycle	*109*	*109*
Total initial cycle	*16,500*	*2130*
Total equilibrium cycle tabs; reset	*11,400*	*1560*

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a. The following parameters were used:

Power level, 3565 MWt.
Release fraction from fuel, 10 percent.
Release fraction from burnable poison rods, 10 percent.
Weight of boron-10 in burnable poison rods, 6160 g.
Initial cycle boron, 900 ppm.
Equilibrium cycle boron, 1110 ppm.
Lithium concentration (99.9 atom-percent lithium-7), 2.2 ppm.
Initial cycle operating time, 9240 effective full-power h.
Equilibrium cycle operating time, 7200 effective full-power h.
Ag-In-Cd tip length in hybrid B_4C/Ag-In-Cd control rods, 40 in.
Release fraction from control rods to the coolant, 100 percent.
Production in control rods is based on continuous daily load follow (12, 3, 6, 3 cycle).
During baseload full-power operation, the production would be negligible.

TABLE 11.1-10

EXPECTED SPENT RESIN VOLUME

Item	Number	Resin Volume Each (ft^3)	Replacement Frequency (number/year)	Annual Average Spent Resin Volume (ft^3)
CVCS mixed bed demineralizers	2 per unit	30	2 beds/year (per unit)	60
CVCS cation bed demineralizer	1 per unit	20	1 bed/4 years (per unit)	5
Recycle evaporator feed demineralizers	2 for 2 units	30	1 bed/year (for both units)	30
Recycle evaporator condensate demin-eralizer	1 for 2 units	30	1 bed/5 years (for both units)	6
Thermal regeneration demineralizers	5 per unit	90	1 bed/year (per unit)	90
Steam generator blowdown mixed bed demineralizer	2 per unit	75	2 beds/year (per unit)	150
Waste monitor tank demineralizer	1 per unit	30	5 beds/6 years (per unit)	25
Spent fuel pool demineralizer	1 per unit	30	4 beds/3 years (per unit)	40

11.2 LIQUID WASTE MANAGEMENT SYSTEMS

The liquid waste management systems include all systems that may be used to process for disposal liquids containing radioactive material. These include:

A. Steam generator blowdown processing system (subsection 10.4.8).

B. Turbine building floor drain system (subsection 9.3.3).

C. Liquid waste processing system (LWPS) (section 11.2).

This section primarily addresses the LWPS. The other systems are also addressed in subsection 11.2.3, which discusses the expected releases from all liquid waste management systems.

The LWPS is designed to control, collect, process, handle, store, and dispose of liquid radioactive waste generated as the result of normal operation, including anticipated operational occurrences.

The LWPS services both units; most equipment is not shared but is associated with either Unit 1 or Unit 2. The shared portions of the LWPS are identified in subsection 11.2.2.

11.2.1 DESIGN BASES

The conformance of the LWPS design with the criteria of Regulatory Guide 1.143 is discussed in section 1.9.

11.2.1.1 Capacity

The projected flows of various liquid waste streams to the LWPS are specified in table 11.2.1-1. The LWPS provides adequate capacity to meet the anticipated processing requirements of the plant.

The LWPS design can accept equipment malfunctions without affecting the capability of the system to handle both anticipated liquid waste flows and possible surge load due to excessive leakage. Surge capacity of individual tanks is discussed in paragraph 11.2.2.6.2.

Portions of the LWPS may become unavailable as a result of the malfunctions listed in paragraphs 11.2.1.1.1 and 11.2.1.1.2.

Ample surge capacity of the system and the low load factor of the processing equipment permit the system to accommodate waste until failures can be repaired and normal plant operation resumed. In addition, the LWPS is designed to accommodate the anticipated operational occurrences described in paragraphs 11.2.1.1.3 through 11.2.1.1.5.

As an additional measure of assurance that liquid wastes can be processed during all plant conditions, piping connections have been installed in the LWPS process stream to allow use of a portable filtration system and the portable demineralizer system, which are both located in the radwaste processing facility.

11.2.1.1.1 Pump Failure

Where operation is not essential and surge capacity is available, a single pump is provided. Two reactor coolant drain tank (RCDT) pumps are provided because the relative inaccessibility of the containment during plant operation would hinder maintenance. Pump repair and replacement is facilitated by using three standard pump designs for the eight applications in the LWPS. To protect the pumps from damage due to loss of suction, each pump is interlocked to stop on a low level condition in the tank feeding the pump.

11.2.1.1.2 Filter, Strainer, or Demineralizer Plugging

Instrumentation is provided to give local indication of the pressure drop across all filters, strainers, and demineralizers. Periodic checks of the pressure drops provide indication of the equipment fouling, thus permitting corrective action to be taken before an excessive pressure drop is reached.

11.2.1.1.3 High Leakage Rate

The system is designed to handle a 1-gal/min primary coolant system leak in addition to the expected leakage during normal operation. Operation of the system is the same as for normal operation except that the load on the system is increased.

The extra leakage may be directed to the waste holdup tank or the floor drain tank. The floor drain tank contents are not necessarily processed; they may be discharged without treatment if sample analysis indicates that the water quality is acceptable for discharge.

11.2.1.1.4 High Use of Decontamination Water

Should large quantities of water be used to decontaminate areas or equipment, the load on the LWPS will be increased. However, the LWPS is designed to handle a 6-gal/min continuous and indefinite input to the waste holdup and/or floor drain tanks. If the flow to the floor drain tank can be discharged without processing, the overall LWPS capacity is increased. Laundry and hot shower wastes are in addition to the 6-gal/min capacity.

The 6-gal/min process capacity is equivalent to almost 5000 gal/day above what is normally sent to the waste holdup and floor drain tanks.

11.2.1.1.5 Refueling

During refueling the load on the LWPS is expected to increase, but operation is the same as for normal plant operation, and there is no significant effect on the performance capability of the LWPS.

11.2.1.2 Controlled Release of Radioactivity

The LWPS provides the capability to reduce the amounts of radioactive nuclides released in the liquid wastes through the use of demineralization and recycling of clean tritiated water for reuse in the reactor plant when possible and time delay for decay of short-lived nuclides.

The assumed equipment decontamination factors are included in table 11.2.1-2. The estimated LWPS isotopic inventories, concentrations, and annual average flowrates that will be processed in the LWPS or discharged to the environment during normal operation are estimated in table 11.2.1-3.

Before any liquid radioactive waste is discharged, it is pumped to a monitor tank. A sample of the monitor tank contents is analyzed and the results logged. In this way, a record is kept of all planned releases of radioactive liquid waste. The liquid waste is discharged from the monitor tank in a batch operation, and the discharge flowrate is restricted as necessary to maintain an acceptable concentration when diluted by the circulating water discharge flow. These provisions preclude uncontrolled releases of radioactivity. In addition, the discharge line contains a stop valve interlocked with a radiation monitor on the system discharge line. The valve automatically closes, and an alarm is actuated if the activity in the discharge stream reaches the monitor setpoint. The stop valve is also interlocked to isolate discharge flow if sufficient dilution flow is not available.

To minimize leakage from the LWPS, the system is of all-welded construction except where flanged connections are required to facilitate component maintenance. The use of canned rotor design pumps for most applications minimizes system leakage and the release of radioactive gas that might be entrained in the leaking fluid to the building atmosphere.

Provisions are made to preclude uncontrolled spills of radioactive liquids due to tank overflows. Tables 11.2.1-4 and 11.4.5-1 lists the provisions for tank level indication, level annunciation, and overflow disposition for tanks located outside the containment building that potentially could contain radioactive liquids.

11.2.1.2.1 Expected Releases

The LWPS design ensures that the annual average concentration limits established by Appendix B, Table II, column 2 of 10 CFR 20.1 - 20.601 for liquid releases are not exceeded during plant operation with expected levels of fuel cladding defects. Subsection 11.2.3 discusses the calculated releases of radioactive materials from the LWPS and other portions of the liquid waste management systems resulting from normal plant operation.

An evaluation is provided in subsection 11.2.3 to demonstrate that the doses to individuals, at or beyond the site boundary, resulting from the expected releases from the liquid waste management systems are within the numerical design objectives of Appendix I of 10 CFR 50.

11.2.1.2.2 Off-Normal Operation

Paragraph 11.2.1.1 discusses the capability of the LWPS to accommodate various equipment failures and anticipated operational occurrences. During these anticipated occurrences, the effectiveness of the LWPS in controlling releases of radioactivity remains essentially unaffected, so releases are limited to approximately the same as during normal operation.

The GALE code,[(1)] used to calculate the release concentration described in paragraph 11.2.1.2.1, contains an adjustment factor of 0.15 Ci/year per reactor to account for abnormal occurrences resulting in unplanned releases.

11.2.1.3 <u>Equipment Design</u>

The LWPS equipment design parameters are provided in table 11.2.1-2.

The seismic design classification and safety classification for the LWPS components and structures are listed in table 3.2.2-1. Safety class designations are also indicated on the LWPS piping and instrumentation diagram, drawings 1X4DB124, 1X4DB125, 1X4DB126, 1X4DB127, AX4DB124-2, AX4DB124-3, AX4DB124-4, and AX4DB124-5.

11.2.1.4 Reference

1. U.S. Nuclear Regulatory Commission, "Calculation of Releases from Pressurized Water Reactors," NUREG-0017, April 1976.

11.2.2 SYSTEM DESCRIPTIONS

The liquid waste processing system (LWPS) collects and processes potentially radioactive wastes for recycling or release to the environment. Provisions are made to sample and analyze fluids before discharge. Based on the laboratory analysis, these wastes are either retained for further processing or released under controlled conditions through the cooling water system, which dilutes the discharge flow. A permanent record of liquid releases is provided by analyses of known volumes of effluent.

The radioactive liquid discharged from the reactor coolant system (RCS) is processed by the radwaste processing facility systems and may be discharged or recycled.

The LWPS is arranged to recycle reactor grade water if desired. This is implemented by the segregation of equipment drains and waste streams to prevent intermixing of liquid wastes. The LWPS can be divided into the following subsystems:

A. Reactor Coolant Drain Tank (RCDT) Subsystem

   This portion of the LWPS collects nonaerated, reactor grade effluent from sources inside the containment.

B. Drain Channel A

   This portion of the LWPS collects aerated, reactor grade effluent that can be recycled.

C. Drain Channel B

   This portion of the LWPS processes all effluent that is not suitable for recycling.

D. Radwaste Processing Facility Demineralizers

   The radwaste processing facility demineralizer systems consist of portable demineralizers installed in subterranean enclosures inside the radwaste processing facility. The radwaste processing facility is described in paragraph 11.4.2.4. The radwaste processing facility demineralizers can be aligned to process any of the three waste drain streams.

E. The radwaste processing facility filtration system consists of a portable, vendor supplied system located within a shielded area inside the radwaste processing facility. The filtration system associated tanks, pumps, accumulator, piping, valves, and controls located within a shielded area inside the radwaste processing facility. The peripheral equipment is located adjacent to the filter assembly. The filter system can be aligned to process any of the three waste drain streams. Details of this equipment are shown on drawing AX4DB124-1.

In addition, the LWPS provides capability for handling and storage of spent ion exchange resins.

The LWPS does not include provisions for processing secondary system wastes. Secondary system effluent is handled by the steam generator blowdown processing system (SGBPS), as described in subsection 10.4.8, and by the turbine building drain system. Estimated releases from these systems are discussed in subsection 11.2.3. The LWPS design, which segregates primary and secondary wastes, minimizes the amount of water that must be processed by discharging low activity wastes directly, where permissible, with no treatment.

Instrumentation and controls necessary for the operation of the LWPS are located on a control board in the auxiliary building. Any alarm on this control board (except for the waste processing holdup control panel) is relayed to the main control board in the control room.

The LWPS piping and instrumentation diagrams are shown in drawings 1X4DB124, 1X4DB125, 1X4DB126, 1X4DB127, AX4DB124-1, AX4DB124-2, AX4DB124-3, AX4DB124-4, and AX4DB124-5 and process flow diagram for the LWPS is shown on figure 11.2.2-1. Table 11.2.1-1 lists the assumptions regarding flows and activity levels that were used in preparation of table 11.2.1-3, which gives nuclide concentrations for key locations within the LWPS as shown on figure 11.2.2-1. The process flow data is calculated using the data in table 11.2.1-1, the flow paths indicated on figure 11.2.2-1, realistic primary coolant activity levels from section 11.1, and decontamination factors as given in reference 1 of subsection 11.2.1.

11.2.2.1 Reactor Coolant Drain Tank Subsystem

Recyclable reactor grade effluents enter this subsystem from valve leakoffs, reactor coolant pump No. 2 seal leakoffs, reactor vessel flange leakoff, and other deaerated, tritiated water sources inside the containment. Connections are provided for draining the RCS loops[a] and the safety injection system (SIS) accumulators and for cooling the pressurizer relief tank. In addition, refueling canal drains can be routed to the refueling water storage tank using the RCDT pumps.

The RCDT contents are continuously recirculated through the RCDT heat exchanger to maintain the desired temperature. Level is prevented from varying significantly by a control valve which automatically opens a path from the recirculation line to the BRS when normal tank level is exceeded. The RCDT is also connected to the gaseous waste processing system (GWPS) vent header. Hydrogen gas bottles connected to the RCDT ensure a hydrogen blanket. Maintaining a constant level minimizes the amount of gas sent to the GWPS and minimizes the amount of hydrogen used. Provisions for sampling the gas are provided.

Details of the RCDT subsystem are shown on drawing 1X4DB127. A separate RCDT subsystem is provided for each of the two units.

11.2.2.2 Drain Channel A Subsystem

Aerated, tritiated liquid enters drain channel A through lines connected to the waste holdup tank. Sources of this aerated liquid are as follows:

A. Accumulator drainage (via RCDT pump suction).

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[a] (Unit 1 only) Flow path from Loop 3 RCL intermediate leg to liquid waste processing system drain header into the reactor coolant drain tank has been removed.

B. Sample room sink drains (excess primary sample volume only).

C. Ion exchanger, filter, pump, and other equipment drains.

The containment sump or auxiliary building sump may be directed to the waste holdup tank or the floor drain tank for processing as necessary.

The collected aerated drainage is pumped or flows to the waste holdup tank prior to processing through the radwaste processing facility filtration system and/or the radwaste processing facility demineralizers before reuse or discharge. Details of this equipment are shown on drawings AX4DB124-2, AX4DB124-3, AX4DB124-4, and AX4DB124-5.

The basic composition of the liquid collected in the waste holdup tank is boric acid and water with some radioactivity.

A separate drain channel A subsystem is provided for each of the two units. Details are shown on drawings 1X4DB124 and 1X4DB127. Table 11.2.1-1 lists the estimated flows entering the waste holdup tank.

11.2.2.3 Drain Channel B Subsystem

Drain channel B is provided to collect and process nonreactor grade liquid wastes. These include:

- Wastes from floor drains.
- Equipment drains containing nonreactor grade water.
- Laundry and hot shower drains.
- Other nonreactor grade sources.

Drain channel B is comprised of three drain subchannels, each associated with one of the following tanks.

A. Laundry and Hot Shower Tank

The laundry and hot shower tank is provided to collect and process waste effluents from the plant laundry and personnel decontamination showers and hand sinks.

Laundry and hot shower drains normally need no treatment for removal of radioactivity. This water is transferred to a waste monitor tank through the laundry and hot shower tank filter for eventual discharge. If sample analysis indicates that decontamination is necessary, the water can be directed through the Unit 1 or Unit 2 waste monitor tank demineralizer or the radwaste processing facility for cleanup.

The laundry and hot shower tank and filter are shared by the two units. Details of this portion of the LWPS are shown on drawing 1X4DB126. Table 11.2.1-1 lists estimated flows entering the laundry and hot shower tank.

B. Floor Drain Tank

Water may enter the floor drain tank from system leaks inside the containment through the containment sump, from system leaks in the auxiliary building through auxiliary building sumps and the floor drains, and floor drains in the radwaste facilities. Sources of water to the containment sump and auxiliary building sumps and floor drains are the following:

1. Fan cooler leaks.
2. Secondary side steam and feedwater leaks.
3. Primary side process leaks.
4. Decontamination water.

The containment sump or auxiliary building sumps may be directed to the waste holdup tank.

Another source of water to the floor drain tank is the chemical laboratory drains. Excess nonreactor grade samples that are not chemically contaminated and laboratory equipment rinse water are drained to the floor drain tank.

The contents of the floor drain tank are processed through the radwaste processing facility demineralizers and/or the radwaste processing facility filtration system and then pumped to a waste monitor tank for ultimate discharge.

If the activity in the floor drain tank liquid is such that the discharge limits cannot be met without cleanup, the liquid can be processed by the waste monitor tank demineralizer, the radwaste processing facility demineralizers, or the radwaste processing facility filtration system.

A separate floor drain tank and associated equipment are provided for each of the two units. Details of this portion of the LWPS are shown on drawing 1X4DB126. Table 11.2.1-1 lists the estimated flows entering the floor drain tank.

C. Chemical Drain Tank

Laboratory samples which contain reagent chemicals (and possibly tritiated liquid) are discarded through a sample room sink which drains to the chemical drain tank. Chemical drains requiring radwaste processing are sent to the solid waste management system or may be processed through the radwaste processing facility demineralizers and/or the radwaste processing facility filtration system.

The chemical drain tank and associated equipment are shared by Units 1 and 2. Details of this portion of the LWPS are shown on drawing 1X4DB125. Table 11.2.1-1 lists the estimated flow directed to the chemical drain tank.

Any liquids released to the environment by the LWPS are first directed to a waste monitor tank. Before releasing the contents of a waste monitor tank, a sample is taken for analysis. The findings are logged, and, if the activity level is within acceptable limits, the tank contents are released to the discharge canal. The discharge valve is interlocked with a process radiation monitor and closes automatically when the radioactivity concentration in the liquid discharge exceeds a preset limit. The radiation element is located upstream of the discharge valve at a distance sufficient to close the valve before passing the fluid that activated the detector trip signal. The isolation valve also blocks flow if sufficient dilution water is not available. The radiation monitor is described in section 11.5. A permanent record of the radioactive releases is provided by a sample analysis of the known volumes of waste effluent released. Liquid waste discharge flow and volume are also recorded.

If the monitor tank contents are not acceptable for discharge, the fluid can be held for a time to allow activity to decay to acceptable levels, or it can be further processed by the waste monitor tank demineralizer, the radwaste processing facility filtration system, or radwaste processing facility demineralizers.

11.2.2.4 Spent Resin Handling Subsystem

This subsystem collects, handles, and processes spent resins from the primary fluid systems prior to their disposal.

Spent resin from the primary system demineralizers is transported to and stored in the spent resin storage tank prior to being drummed. The spent resin sluice portion of the LWPS consists of a spent resin sluice filter, spent resin sluice pump, and the spent resin storage tank. The resin sluice water, after being directed to an ion exchange vessel by the sluice pump, returns to the spent resin storage tank for reuse.

Thus, sluicing of spent resin from primary plant demineralizers is accomplished without generating a large volume of liquid waste.

Reactor makeup water storage tank system pressure may also be used as a motive force for sluicing demineralizers to the spent resin storage tank.

Resin from the SGBPS demineralizers is handled in a similar manner to that described for the primary system demineralizers. The SGBPS resin handling is discussed in subsection 10.4.8. The primary system resins and the SGBPS resins are segregated during all phases of handling so that no cross-contamination occurs.

Resin slurry from both spent resin storage tanks is sent to the radwaste processing facility by pressurizing the tank with nitrogen.

A separate resin handling subsystem is provided for each of the two units. Details of this portion of the LWPS are shown on drawing 1X4DB125.

11.2.2.5 Liquids From Sources Other Than the Liquid Waste Processing System

11.2.2.5.1 Steam Generator Blowdown Processing System

Blowdown from the steam generators of each unit is cooled, filtered, and demineralized. Normally it is then returned to the condensers for reuse as condensate makeup, but it may be discharged to the environment. The SGBPS is described in subsection 10.4.8.

11.2.2.5.2 Turbine Building Drain System

The function of the turbine building drain system is:

A. To collect the floor drains and sampling wastes in the turbine building and other miscellaneous drains.

B. To treat these wastes, if necessary, to meet the requirements of the State of Georgia Environmental Protection Agency (EPA) regulations and Nuclear Regulatory Commission regulations prior to discharge to the Savannah River.

All the wastes generated in the turbine building including drains, leakages, and the sampling wastes are collected in the sumps in the turbine building. The sump pump discharge, combining with the auxiliary building clean water sump discharge, auxiliary feedwater pumphouse drain, maintenance building drain, and the treated turbine building drain, normally passes through a radiation monitor before entering into an oil water separator to meet the EPA oil discharge limit. If the radioactivity level of this combined waste stream exceeds the setpoint of the radiation monitor, a signal is sent to stop the flow to the nonradioactive oil water separator. This stream is then sent to one of the turbine building drain tanks before processing. The amount of waste delivered to the turbine building drain tanks is kept minimal by promptly locating the specific contaminated (radioactive) stream(s) and pumping the remaining (nonradioactive) streams via their normal route. The turbine building drain system and associated components are discussed in detail in subsection 9.3.3.

11.2.2.6 Equipment Description

Principal design parameters for the LWPS equipment are given in table 11.2.1-2. All parts or components in contact with borated water are fabricated from or clad with austenitic stainless steel. Pumps are provided with vent and drain connections.

Since this system performs no function related to the safe shutdown of the plant, many components of the system are classified as nonnuclear safety. Component safety classes, seismic design, and principal codes are shown in table 3.2.2-1.

11.2.2.6.1 Pumps

Pumps in the LWPS have been standardized wherever possible. Where operation is not critical and surge capacity is available, a single pump has been provided. Spare pumps can be kept onsite in case any pump should fail. Quick replacement is possible because:

- The pumps are flanged, not welded.
- The system has surge capacity.
- Adequate vent, flush, and drain capabilities are provided.
- The pumps are standardized.

Three types of standard pumps are required, as described below.

Canned rotor pumps are used wherever possible to minimize fluid leakage and to minimize the release of entrained radioactive gases in the leaking fluid to the atmosphere.

Canned rotor pumps with similar head-flow characteristics are used for the following applications:

- Spent resin sluice pumps.
- RCDT pumps.

Another canned rotor pump design is used for the following applications:

- Waste evaporator feed pumps.
- Waste evaporator condensate tank pumps.

- Chemical drain tank pump.
- Waste monitor tank pumps.

Mechanical seal pumps are also used in some applications because these pumps have an open impeller, which will not be damaged by large particles in the water. Mechanical seal pumps with similar head-flow characteristics are used for the following applications:

- Floor drain tank pump.
- Laundry and hot shower tank pump.

Globe valves are installed in pump discharge lines where necessary to prevent pump runout. Pump miniflow lines have locked-in-position globe valves to ensure that the minimum pump flow requirements are met.

A. RCDT Pump

Two pumps are furnished because of the relative inaccessibility of the containment during plant operation. Both pumps are operated to meet the draining time requirement. One pump provides sufficient flow for normal operation of the RCDT portion of the LWPS. The liquid is sent to the recycle holdup tanks.

B. Waste Evaporator Feed Pump

This pump supplies feed to the radwaste processing facility for processing from the waste holdup tank, and it can be used to transfer waste holdup tank contents to the floor drain tank, if desired.

C. Waste Evaporator Condensate Pump

The waste evaporator condensate tank pump is used to transfer the contents of the waste evaporator condensate tank.

D. Chemical Drain Tank Pump

This pump may be used to transfer the contents of the chemical drain tank.

E. Spent Resin Sluice Pump

One pump is provided to sluice resins from primary side demineralizers to the spent resin storage tank. Its delivery flow is based on the velocity required to sluice resin in a 3-in. pipe.

F. Laundry and Hot Shower Tank Pump

This pump is used to transfer the water from the laundry and hot shower tank to a waste monitor tank.

G. Floor Drain Tank Pump

This pump is used to transfer water from the floor drain tank to the radwaste processing facility. From the radwaste processing facility, processed waste is transferred to the waste monitor tank.

H. Waste Monitor Tank Pumps

Two pumps are provided for each unit. One pump is used for each monitor tank to discharge water from the LWPS or for recycling if further processing is required.

The pump may also be used for circulating the water in the waste monitor tank to obtain uniform tank contents, and therefore a representative sample, before discharge. These pumps can be throttled to achieve the desired discharge rate.

I. Auxiliary Waste Monitor Tank Pumps

Two pumps are provided. They are installed in Unit 2 but serve both units. One pump is used for each auxiliary waste monitor tank to discharge water from LWPS or for recycling if further processing is required. A mixer may be used for circulating the water in the auxiliary waste monitor tank to obtain uniform tank contents, thereby assuring a representative sample is acquired prior to discharge of the tank contents. The pumps can be throttled to achieve the desired discharge rate.

11.2.2.6.2 Tanks

A. Reactor Coolant Drain Tank

One tank is provided for each unit. The purpose of the RCDT is to collect leakoff-type drains inside the containment at a central collection point for further disposition through a single penetration via the RCDT pumps. The tank provides surge volume and net positive suction head (NPSH) to the pumps.

Only water which can be directed to the boron recycle holdup tanks enters the RCDT. The water is compatible with reactor coolant and does not contain dissolved air during normal plant operation, by engineering design.

A constant level is maintained in the tank to minimize the amount of gas sent to the GWPS and also to minimize the amount of hydrogen cover gas required. The level is maintained by one continuously running pump and by a control valve in the discharge line. This valve operates on a signal from a level controller to limit the flow out of the system. The remainder of the flow is recirculated to the tank.

Continuous flow is maintained through the heat exchanger in order to prevent loss of pump NPSH resulting from a sudden inflow of hot liquid into the RCDT.

B. Waste Holdup Tank

One atmospheric pressure tank is provided for each unit to collect:

1. Equipment drains.
2. Valve and pump seal leakoffs (outside the containment).
3. Boron recycle holdup tank overflows.
4. Other water from tritiated, aerated sources.

The tank size is adequate to accommodate 11 days of expected influent during normal operation.

C. Waste Evaporator Condensate Tank

One tank with a diaphragm to exclude air is provided for each unit to collect water from processing systems.

D. Chemical Drain Tank

One tank is provided to collect chemically contaminated, tritiated water from the laboratories. This tank is shared by the two units and has sufficient capacity to accept a month's laboratory waste during normal operation of both units.

E. Spent Resin Storage Tank

The purpose of the spent resin storage tank is to provide a collection point for spent resin and to allow for decay of short-lived radionuclides before disposal. The tank also serves as a head tank for the spent resin sluice pump.

One vertical, cylindrical tank with sufficient capacity to handle the spent resin storage needs is provided. A vertical, cylindrical tank is used because the symmetrical bottom facilitates the removal of resin. The tank is designed so that sufficient pressure can be applied in the gas space of the tank to move the resin slurry to the radwaste processing facility.

The spent resin storage tank and associated equipment, which can contain radioactive material, are shielded to limit the dose to personnel.

The level indicating system in the spent resin storage tank shows only total level and not the amount of resin and water separately. However, since the resin volumes flushed from demineralizers and the resin volumes transferred to the radwaste processing facility are known, the resin level in the tank is also known.

F. Laundry and Hot Shower Tank

One atmospheric pressure tank is provided to collect laundry and hot shower drains for the two units. The tank size is sufficient to furnish a 10-day surge capacity for the two units during normal operation of both units and a 2-day surge capacity during refueling of a single unit.

G. Floor Drain Tank

One atmospheric pressure tank is used to collect floor drains from the controlled areas of each unit's primary system. The tank provides sufficient surge capacity for the floor drains within the collection area and, in connection with the waste holdup tank, provides surge capacity for abnormal primary system leaks. The tank size is adequate to accommodate 3.5 days of expected influent during normal operation or 1.8 days of expected influent during shutdown operation.

H. Waste Monitor Tanks

Two atmospheric pressure waste monitor tanks are provided for each unit to monitor liquid discharged from the plant site. Each tank is sized to hold a volume large enough that sampling requirements are minimized, thereby minimizing laboratory effluent.

Two additional atmospheric pressure tanks have been provided to augment the plant capacity to handle large surges of water or to accommodate conditions in which release of the processed water is not feasible. These 20,000 gallon tanks are installed on level D of the Unit 2 auxiliary building. They are shared by both units, with one tank normally aligned to each unit.

11.2.2.6.3 Reactor Coolant Drain Tank Heat Exchanger

This heat exchanger is located in the discharge line of the RCDT pumps and is in constant service as part of the RCDT recirculation path. Continuous auxiliary component cooling water flow is maintained to the heat exchanger to accommodate, without operator action, sudden flow of hot liquid to the RCDT. The heat exchanger can also be used to cool the contents of the pressurizer relief tank in the RCS.

The heat exchanger is sized for several modes of operation:

A. It can maintain the RCDT liquid below 170°F, assuming a 10-gal/min input of 600°F reactor coolant.

B. It can cool the contents of the pressurizer relief tank from 200°F to 120°F in less than 8 h.

C. It can maintain the contents of the RCDT at less than 170°F, assuming a 25-gal/min input from the excess letdown heat exchanger (chemical and volume control system).

One RCDT heat exchanger is provided per unit.

11.2.2.6.4 Demineralizers

As part of a program of continuous pressurized water reactor operating plant followup, Westinghouse has obtained operational data on demineralizer decontamination factors for selected isotopes. The measured range of decontamination factors for these isotopes is given in table 11.2.2-1.

These values were observed across mixed bed demineralizers containing cation resin in the lithium-7 form and anion resin in the borated form. The minimum values in table 11.2.2-1 were generally observed just before resin flushing and recharging; during the operating life of the demineralizer, decontamination factors were consistently closer to the maximum values.

Although specific operating decontamination factors have not yet been measured for other isotopes, their behavior in a mixed bed demineralizer may be inferred from this data.

The process decontamination factor used for the demineralizers in the analysis of system performance is taken from reference 1, subsection 11.2.1. These decontamination factors are given in table 11.2.2-1.

A. Waste Monitor Tank Demineralizer

One mixed bed demineralizer is provided upstream of the waste monitor tanks to remove, if desired, trace ionic contaminants. The laundry and hot shower tank contents can also be processed through the demineralizer if such processing is necessary.

B. Radwaste Processing Facility Demineralizers

The radwaste processing facility houses a series of demineralizers which can be operated in various configurations. The type and loadings of resins or other filter media utilized in these vessels can be changed as necessary to optimize the performance of the system.

11.2.2.6.5 Filters

The following filters are provided in the LWPS:

- Laundry and hot shower tank filter.

- Floor drain tank filter.
- Spent resin sluice filter.
- Waste evaporator condensate filter.
- Waste monitor tank filter.

The laundry and hot shower tank filter element is normally removed since installation of the filtration system in the radwaste processing facility.

11.2.2.6.6 Strainers

Basket-type strainers of mesh construction are provided to prevent clogging of filters and lines downstream because of large particles being sluiced through the lines during liquid transfer operations.

The following strainers are provided in the LWPS:

- Laundry and hot shower tank strainer.
- Floor drain tank pump suction strainer.

11.2.2.7 Instrumentation Design

The system instrumentation is shown on the LWPS drawings 1X4DB124, 1X4DB125, 1X4DB126, 1X4DB127, AX4DB124-2, AX4DB124-3, and AX4DB124-4.

Instrumentation readout is located mainly on the waste processing system panel in the auxiliary building. Some instruments are read at the equipment location.

All alarms are shown separately on the waste processing system panel and are further relayed to one common waste processing system annunciator on the main control board in the control room. The waste processing holdup control panel does not relay signals to the control room annunciator.

All pumps are protected against loss of suction pressure by a control setpoint on the level instrumentation for the respective vessels feeding the pumps. In addition, the RCDT pumps and the spent resin sluice pump are interlocked with flowrate instrumentation to stop the pumps when the delivery flows reach minimum setpoints. The RCDT pumps have a keyswitch allowing the bypass of the flowrate instrumentation control logic. During certain intermittent outage flowpaths, flowrates less than those recommended by the pump vendor may be established. Administrative controls require continuous monitoring of pump performance during such flowpaths to ensure the pumps' availability.

Pressure indicators are provided to give local indication of pressure drops across demineralizers, filters, and strainers.

All releases to the environment are monitored for radioactivity. This instrumentation is described in section 11.5.

Each tank is provided with level indication instrumentation that actuates an alarm on high liquid level in the tank, thus warning of potential tank overflow.

11.2.2.8 Operating Procedures

The LWPS is manually operated except for some functions of the RCDT circuit. The system includes adequate control equipment to protect components and adequate instrumentation and alarm functions to provide operator information to ensure proper system operation.

Operation of the LWPS is essentially the same during all phases of normal and defined off-normal reactor plant operation; the only differences are in the load on the system. The term "normal operation," as used here, means all phases of plant operation except operation under emergency or accident conditions. The LWPS is not regarded as an engineered safety features system.

11.2.2.8.1 Reactor Coolant Drain Tank Subsystem Operation

A. Reactor Coolant Drain Tank Recirculation

Reactor coolant is continuously circulated through the RCDT heat exchanger to maintain ≤170°F in the event of a hot reactor coolant leak. Level is maintained by a control valve which automatically opens a path from the recirculation line to the recycle holdup tanks. Normal operation of this mode is automatic and requires no operator action. The system can be put into the manual mode, if desired.

Leakage into the RCDT can be estimated by putting the system in the manual mode, stopping the pump, and monitoring the level change. A venting system is provided to prevent wide pressure variations in the RCDT. A hydrogen blanket is automatically maintained between 2 and 6 psig. Hydrogen is supplied from bottles which must be replaced when the pressure drops to ~100 psig.

During all other operations, the recirculation mode is stopped and the RCDT is isolated.

B. Pressurizer Relief Tank Cooling

The pressurizer relief tank may be cooled by a feed- and-bleed method, by spraying cold makeup water and pumping the water to the recycle holdup tanks with an RCDT pump through the RCDT heat exchanger. This is a rapid cooldown and can be used even if the heat exchanger is out of service. However, to minimize the addition of water to the system when a rapid cooldown is not necessary, the pressurizer relief tank may be cooled from 200°F to 120°F in less than 8 h by using the RCDT heat exchanger and one pump in a recirculation mode without adding makeup water.

C. Loop Draining

Unit 2 only - Four RCS loops may be drained simultaneously. The loops are vented to 100 psig, then a spectacle flange is positioned to the pumps' suction. The water may be sent to the recycle holdup tanks, refueling water storage tank, waste holdup tank, or the spent fuel pool. Since the RCDT heat exchanger is not needed for cooling, it may be bypassed or used in parallel with the bypass.

Unit 1 only - Flow path from Loop 3 RCL intermediate leg to liquid waste processing system drain header into the reactor coolant drain tank has been removed. As such, three RCS loops may be drained simultaneously. The loops are vented to 100 psig, then a spectacle flange is positioned to the pumps' suction. The water may be sent to the recycle holdup tanks, refueling water storage tank, waste holdup tank, or the spent fuel pool. Since the RCDT heat exchanger is not needed for cooling, it may be bypassed or used in parallel with the bypass.

D. Refueling Canal Draining and Cleanup

Refueling canal water is transferred, using the RCDT pumps, to the refueling water storage tank (draining), to the canal via the spent fuel pool cooling system (cleanup), or to the waste holdup tank (disposal).

Since the RCDT heat exchanger is not needed for cooling, it may be bypassed or used in parallel with the bypass.

E. Accumulator Draining

This mode is available for accumulator maintenance. After the accumulator is vented to 90 psig, the spool piece is connected to the RCDT pump suction for transfer of accumulator water to the reactor water storage tank or recycle holdup tank.

F. Excess Letdown Header

During normal plant heatup operations, excess letdown to the RCDT will permit faster heatup rate. Excess letdown flow is directed from the RCDT to the recycle holdup tanks.

11.2.2.8.2 Drain Channel A Subsystem Operation

Water is accumulated in the waste holdup tank until sufficient quantity exists to warrant starting the processing systems for a batch process.

If it is not desired to recycle the water in the waste holdup tank and analysis indicates that decontamination is not necessary, the water may be sent to the floor drain tank for eventual discharge.

The waste holdup tank contents may be processed by the radwaste processing facility filtration system and/or the radwaste processing facility demineralizers.

11.2.2.8.3 Drain Channel B Subsystem Operation

Drain channel B of the LWPS consists of the laundry and hot shower system, the floor drain tank system, and the chemical drain tank system.

Laundry and hot shower water enters the laundry and hot shower tank for holdup; it is sampled, filtered, and transferred to the monitor tank for discharge, or processed through the radwaste processing facility filtration system or the radwaste processing facility demineralizer. If demineralization is required, the resin must be replaced with clean resin thereafter.

The floor drain tank contents are recirculated, and then samples are taken and analyzed. If the floor drain tank is overloaded or the water is recyclable, the water can be transferred to the waste holdup tank.

Water leaving this system to the discharge canal is monitored for radiation. This radiation monitor is described in section 11.5. If the radiation monitor closes the discharge valve, it must be cleared before the valve can be reopened. The monitor element can be cleared by flushing it with demineralized water from the temporary connection back to the waste monitor tank. During refueling, the load on this portion of the LWPS is increased, but there is no change in operation.

Spent samples with high chemical concentrations are held up in the chemical drain tank, then sampled. The contents are drained to the auxiliary building sumps or the radioactive drain sumps.

11.2.2.8.4 Spent Resin Handling Subsystem

A. Resin Fluffing

The demineralizer is valved out of service, and the flow path is aligned from the spent resin sluice pump (or the reactor makeup water system header) through the process line of the demineralizer, through the Johnson screen at the top of the demineralizer, and back to the spent resin storage tank. The resin bed is backflushed for about 10 min to loosen it for sluicing. This operation may also be used to recover pressure drop caused by bed fouling by backwashing particulates from the top layer of the resin into the spent resin storage tank. Such a recovery is useful when the resin is not ionically depleted.

B. Resin Sluicing

The sluice pump is shut off after fluffing. The valves in the backflush circuit are closed, the sluice line to the bed screen is opened, and the sluice pump is started. The resin flows to the spent resin storage tank, initially at a slow rate; it continues for about 10 min until sluicing is completed. Finally, the pump is stopped, and the sluice inlet and outlet valves are closed. If the pump suction line screen in the tank becomes resin plugged at any time, it can be cleared with a blast of nitrogen.

C. Resin Fill

After sluicing is completed, fresh resin must be added. The path to the drain header from the demineralizer is opened to allow overflow. The resin fill line is opened and resin added. After fresh resin is added, the fill line valve is closed and the flow path is realigned for normal demineralizer operation.

D. Resin Disposal

The resin in the spent resin storage tank is loosened before disposal by sending pressurized nitrogen or sluice water through the six sparging nozzles in the tank. The valves in the resin transfer line are opened to direct the spent resin to the radwaste processing facility.

The tank is then pressurized with nitrogen to force the resin up through the resin transfer line to the disposal area. A single nozzle in the spent resin storage tank is provided to allow local fluidization with sluice water at the opening of the discharge pipe. During resin transfer, this nozzle is used to ease the flow of the resin slurry. After resin transfer is complete, the tank is vented to the plant vent and returned to atmospheric pressure. The resin transfer line is then backflushed to the spent resin storage tank to clear it of resin.

Since a certain amount of resin remains in the tank after a disposal operation, it may hinder the backflush operation. Therefore, the fluidizing nozzle is again used to facilitate the backwash operation.

11.2.3 RADIOACTIVE RELEASES

11.2.3.1 Criteria for Discharge, Recycle, or Further Treatment of Liquid Waste

Processed liquids are recycled for reuse within the plant whenever possible, provided that the following criteria are satisfied:

- The plant water inventory requires makeup.
- The water to be reused satisfies system water quality requirements.
- Tritium buildup is less than plant operating limits.

Processed liquids are discharged under the following conditions:

- The processed water does not satisfy plant operating requirements for water quality and tritium buildup.
- The effluent concentrations are within the limits specified by the Offsite Dose Calculation Manual.
- The discharge does not cause the limits of 10 CFR 50, Appendix I, to be exceeded.

Processed liquids are recycled within their respective treatment systems for additional processing when system water quality requirements are not satisfied and reuse within the plant is desirable, or when discharge of the processed liquid is planned but the discharge would result in exceeding limits in the Offsite Dose Calculation Manual.

11.2.3.2 Estimated Releases

The equipment utilized during liquid waste processing is at the discretion of the operator; therefore, the calculated releases conservatively do not address all possible treatment processes but only the expected process. Liquid releases from VEGP are calculated using the PWR-GALE computer code[(1)] and parameters listed in table 11.1-8, which are discussed in more detail below. Releases calculated assuming operation with expected levels of fuel cladding defects of 0.12 percent are presented in table 11.2.3-1. Primary and secondary coolant activity levels are given in section 11.1 for the realistic case. In agreement with reference 1, the total releases include an adjustment factor of 0.15 Ci/year, using the same isotopic distribution as the calculated release, to account for anticipated operational occurrences.

The tables list the calculated annual release from each of the process paths discussed below as well as the total annual release. A comparison of annual average effluent concentrations with Appendix B, Table II, column 2 values of 10 CFR 20.1 -20.601 is provided in table 11.2.3-2 for operation with expected fuel leakage.

The releases are calculated for one unit, assuming that both units are operating. This is done to reflect the impact of the second unit's operation on the operation of systems and components shared between the two units. To obtain the combined releases of the two units, simply double the values listed in table 11.2.3-1.

A survey has been performed of liquid discharges from different Westinghouse pressurized water reactor plants, with results presented in table 11.2.3-3. The data includes radionuclides released on an unidentified basis and are all within the permissible concentration for release of liquid containing an unidentified radionuclide mixture. The data in table 11.2.3-4 clearly indicate that actual releases are highly dependent upon the actual operation of the plant and can vary significantly from year to year for a given plant as well as from plant to plant.

11.2.3.2.1 Boron Recycle System (BRS)

Primary coolant is withdrawn from the reactor coolant system (RCS) and processed through the chemical and volume control system (CVCS). A side stream of 1700 gal/day of the letdown stream is assumed to be diverted to the BRS as shim bleed. The shim bleed is combined with a conservatively estimated 300 gal/day of other reactor grade wastes that are collected by the reactor coolant drain tank (equipment drain wastes). Since the BRS is shared by both units, the total process flow is 4000 gal/day. The equipment drains and shim bleed flows have an activity level equivalent to primary coolant activity (PCA).

The combined shim bleed and equipment drain wastes streams are routed to one of the recycle holdup tanks. The contents of the recycle holdup tank are then processed through the radwaste processing facility filtration system and the radwaste processing facility demineralizers. The water is either pumped to the reactor makeup water storage tank for reuse in the plant or to a waste monitor tank for monitoring and discharge. The BRS has sufficient capacity to allow total reuse of the combined shim bleed and equipment drain wastes.

Radioactive decay during collection in the recycle holdup tanks is calculated using a collection time of 22.4 days. This value is based upon filling one of the recycle holdup tanks to 80 percent of capacity using the combined shim bleed and reactor coolant drain tank flows. Radioactive decay during processing and discharge is calculated using a process time of 0.31 days. This value is based upon processing the combined shim bleed and reactor coolant drain tank flows at the design flowrate of the recycle evaporator.

The decontamination factors used in calculating radionuclide removal for iodine, cesium, rubidium, and other nuclides, are determined by applying the methodology and parameters of reference 1 to the processing capabilities of the BRS and CVCS as shown in figure 11.2.3-1. No credit is taken for the recycle evaporation condensate demineralizer since its main function is to remove boron carryover.

11.2.3.2.2 Liquid Waste Processing System (LWPS)

A. Clean Wastes (Drain Channel A - Miscellaneous Wastes)

Clean wastes are collected in the waste holdup tank for processing. Based on sample analysis, the water in the monitor tank would either be discharged or processed further by recirculation through the waste monitor tank demineralizer until sample analysis indicated that it was acceptable for discharge. The flow to the waste holdup tank is 713 gal/day at 0.051 times PCA. The LWPS has sufficient capacity to allow total reuse of the processed clean wastes. However, in the release calculations a discharge fraction of 1.0 is used as specified by reference 1.

Radioactive decay during collection in the waste holdup tank is calculated using a collection time of 5.6 days. This value is based upon filling the waste holdup tank to 40 percent of capacity. Radioactive decay during processing and discharge is calculated using a process time of 0.028 days. This value is based upon processing 40 percent of the waste holdup tank capacity at the design flowrate of the waste evaporator (no longer in system) and the time required to pump this volume of water out of the monitor tank at the maximum pumping rate.

The decontamination factors used in calculating radionuclide removal are based upon demineralization as shown in figure 11.2.3-1.

B. Dirty Wastes (Drain Channel B - Miscellaneous Wastes)

Dirty wastes are collected in the floor drain tank. A sample is analyzed to determine whether the water can be discharged without processing. If cleanup is required, the liquid is processed by the radwaste processing facility filtration system and/or the radwaste processing facility demineralizer and the effluent directed to a waste monitor tank. Based on sample analysis, the monitor tank contents are either discharged or processed further by recirculation through the monitor tank demineralizer until sample analysis indicates that the water is acceptable for discharge. The flow to the floor drain tank is 2047 gal/day at 0.02 times PCA. Since all of the dirty wastes are normally discharged, a discharge fraction of 1.0 is used in the release calculations.

Radioactive decay during collection in the floor drain tank is calculated using a collection time of 1.43 days. This value is based upon filling the floor drain tank to 40 percent of capacity. Radioactive decay during processing is calculated using process time of 0.03 days. This value is based upon processing 40 percent of the floor drain tank capacity at the design flowrate of the waste evaporator (no longer in service) and the time required to pump this volume of water out of the monitor tank at the maximum pumping rate.

The decontamination factors used in calculating radionuclide removal are based upon the decontamination factors given in reference 1 for demineralizers as shown in figure 11.2.3-1.

C. Detergent Wastes (Laundry and Hot Shower Tank)

Detergent wastes are normally released without treatment. The releases through this path are assumed to be the same as listed in reference 1, table 2-20.

11.2.3.2.3 Steam Generator Blowdown Processing System (SGBPS)

Blowdown from the steam generators is normally processed by the two generator blowdown demineralizers (in series) and recycled back to the main condenser. If discharge of blowdown is desired, the demineralizers can be bypassed. A valve in the discharge line automatically stops discharge on a high radioactivity signal. In the event of a primary to secondary side leak, the demineralizers would be in use. For the release calculation, a 100 lb/day primary to secondary side leakage is assumed. Also, a design basis total blowdown rate of 180,096 lb/h is assumed with 100 percent of the flow being discharged.

No credit is taken for radioactive decay of the isotopes in the blowdown stream since the SGBPS design utilizes a through-flow process with no significant decay.

The decontamination factors used in calculating radionuclide removal are shown in figure 11.2.3-1. These values are based upon the decontamination factors given in reference 1 for two steam generator blowdown mixed bed demineralizers (in series).

11.2.3.2.4 Turbine Building Floor Drains

The processing of the low level radioactive water in the turbine building dirty drain tank is manually initiated by the operator based on the predetermined water level in the tank. This processing system, located in the auxiliary building, consists of an oil water separator, an activated charcoal filter, two demineralizers in series, and a discharge filter. The treated water is returned to the clean drain tank in the turbine building. After monitoring it is combined with other waste streams for disposal.

11.2.3.3 Release Points

The natural draft cooling towers are not radioactive release points. However, the following information regarding location is offered.

Blowdown from both cooling towers and flow from the waste water retention basin is collected in a common blowdown sump and is discharged to the river via the waste water effluent pipe. The radioactive release line discharges into the waste water effluent pipe at a point downstream of the blowdown sump and is then discharged into the Savannah River downstream of the river intake structure.

11.2.3.4 Dilution Factors

At 100-percent capacity factor and design basis conditions for cooling tower operation, blowdown from the cooling tower to the blowdown sump is approximately 5000 gal/min per unit. Furthermore, 10,000 gal/min per unit river water can be provided to the sump through a dilution flow transfer line takeoff on the river water makeup line to the natural draft cooling towers. Thus, a total of 15,000 gal/min per unit dilution flow is available during the liquid radwaste discharge operation. Minimum dilution flow is as specified in the release permit which is prepared in accordance with the ODCM.

There will be some additional dilution following discharge due to the effect of the near field mixing zone in the Savannah River. For 15,000 gal/min (33.4 ft^3/s) effluent discharge into the Savannah River with 5800 ft^3/s minimum flow, the VEGP thermal plume analysis utilizes a dilution factor of 10 for summer discharge conditions and 20 for winter conditions.(2) For conservatism, the lower dilution factor of 10 was chosen for the LADTAP II analysis.

11.2.3.5 Estimated Doses

Release of radioactive effluents to the Savannah River during normal plant operation and anticipated operational occurrences will result in a minimal radiological exposure to individuals as noted in table 11.2.3-4. The estimated annual average doses to the maximum exposed individual were calculated using the LADTAP II computer code.(3)

The location of maximum exposure is in that area of the river allowing for initial near field dilution of the discharge with essentially zero travel time. Since crop irrigation from the Savannah River has not been observed in the vicinity of the plant site, this pathway has not been considered in the evaluation of doses. The anticipated dose due to drinking water could be considered insignificant because the nearest location of potable use of river water is in Beaufort County, South Carolina, approximately 103 river miles downstream of the plant site. However, for conservatism, this pathway was evaluated assuming the maximum exposed individual obtains drinking water in the immediate area of the discharge plume. Shoreline use in the vicinity of the plant is very limited, with essentially no fishing from the bank, swimming, or sunbathing, and, consequently, is expected to be an insignificant pathway in comparison with the pathway of aquatic foods. Nevertheless, for purposes of conservatism, this pathway has been included in the evaluation of doses for the maximum exposed individual.

Furthermore, a single dilution factor (10.0) for the initial near field dilution of the discharge with a travel time of zero was used for the pathway's consideration. In lieu of site specific data, the pathway usage factors and shorewidth factor (0.2) outlined in Regulatory Guide 1.109, Revision 1, October 1977, were utilized in this evaluation. As noted in table 11.2.3-4, based on the criteria as outlined above, the maximum exposed individual annual doses from the discharge of radioactive materials in liquid effluent from each of the VEGP units meet the guideline of Appendix I, 10 CFR 50, and Docket RM-50-2, Annex to Appendix I.

11.2.3.6 References

1. U.S. Nuclear Regulatory Commission, "Calculation of Releases from Pressurized Water Reactors," PWR-GALE Computer Code, NUREG-0017, April 1976.
2. Georgia Power Company, "Waste Water Effluent Discharge Structure Thermal Plume Analysis," Vogtle Electric Generating Plant Units 1 and 2, Revised March 1981.
3. U.S. Nuclear Regulatory Commission, "Calculation of Radiation Exposure to Man from Routine Release of Nuclear Reactor Liquid Effluents," LADTAP II Computer Code, NUREG/CR-1276, March 1980.

TABLE 11.2.1-1 (SHEET 1 OF 2)

PARAMETERS USED IN THE CALCULATION OF ESTIMATED ACTIVITY IN LIQUID WASTES

Collection Tank and Sources	Expected Input Rates[(a)]	Activity[(b)]	Basis	Disposition
1. Reactor coolant drain tank	300 gal/day	1.0 x (RCS) tritium; 0.1 x (RCS) other nuclides	RC pump No. 2 seal leakoff of 0.05 gal/min per pump	Directed to BRS recycle holdup tanks for process and transfer to the waste monitor tank
2. Waste holdup tank	913 gal/day	0.051 x (RCS)		Processed and transferred to the waste monitor tank
a. Equipment drains	156 gal/day	0.1 x (RCS)	Drains from tanks, filters, demineralizers, and heat exchangers containing reactor grade water	
b. Containment sump[(c)]	530 gal/day	0.036 x (RCS)	ANSI/ANS-55.6-1979	
c. Sample room drains	27 gal/day	0.05 x (RCS)	(d)	
3. Floor drain tank	2560 gal/day (normal) 5520 gal/day (shutdown)			Processed and transferred to the waste monitor tank
a. Auxiliary building sump	200 gal/day	0.1 x (RCS)	NUREG-0017	
b. Miscellaneous	700 gal/day	0.01 x (RCS)	NUREG-0017	
c. Lab equipment rinse water	164 gal/day	0.05 x (RCS)	(d)	
d. Area decontamination	40 gal/day (normal) 3000 gal/day (shutdown)	0.01 x (RCS) 0.01 x (RCS)	ANSI/ANS-55.6-1979	
e. Spent fuel pool liner leakage	700 gal/day	0.001 x (RCS)	ANSI/ANS-55.6-1979	
4. Chemical drain tank	9 gal/day	0.148 x (RCS)	(d)	Drained to auxiliary building sumps

TABLE 11.2.1-1 (SHEET 2 OF 2)

Collection Tank and Sources	Expected Input Rates[a]	Activity[b]	Basis	Disposition
5. Laundry and hot shower tank	500 gal/day (normal) 3900 gal/day (shutdown)	(f) (f)		Discharged to environment after sampling
a. Laundry waste	300 gal/day (normal) 2000 gal/day (shutdown)	(f) (f)	ANSI/ANS-55.6-1979 ANSI/ANS-55.6-1979	
b. Hot shower waste	Negligible (normal) 400 gal/day (shutdown)	(f) (f)	ANSI/ANS-55.6-1979 ANSI/ANS-55.6-1979	
c. Hand washes	200 gal/day (normal) 1500 gal/day (shutdown)	(f) (f)	ANSI/ANS-55.6-1979 ANSI/ANS-55.6-1979	
6. Boron recycle system	500 gal/day	(g)	Two RCS volumes purged per year for tritium control.	Discharged to environment after sampling

a. Input rates are for one unit. These rates do not represent system capacity. During operation of both units, shared tanks (chemical drain tank and the laundry and hot shower tank) have double the indicated input rates.

b. Activity of the liquid wastes entering the LWPS is given in terms of a fraction of primary coolant activity as given in section 11.1 unless stated otherwise.

c. Each sump may be directed to either the waste holdup or floor drain tank, depending upon water quality.

d. Expected input rates are based on a total rate of 200 gal/day in accordance with ANSI/ANS-55.6-1979, Reactor Systems Sampling. The total rate is divided among sample room drains, lab rinse water, and chemically contaminated samples.

e. Shutdown is assumed to last 30 days.

f. Laundry and hot shower waste activity is taken from NUREG-0017, April 1976.

g. The 500 gal/day coming from the boron recycle system is a fraction of the liquid processed by that system. The activity of the water is primary coolant activity in accordance with section 11.1 but reduced by the processing performed by the boron recycle system.

TABLE 11.2.1-2 (SHEET 1 OF 8)

EQUIPMENT DESIGN PARAMETERS

<u>Pumps</u>

RCDT pumps	
Number	2 (per unit)
Type	Canned centrifugal
Design pressure (psig)	150
Design temperature (°F)	200
Design flow (gal/min)	100/140[(a)]
Design head (ft)	300/250[(a)]
Material	Stainless steel (SS)
Waste evaporator feed pump	
Number	1 (per unit)
Type	Canned centrifugal
Design pressure (psig)	150
Design temperature (°F)	200
Design flow (gal/min)	35/100[(a)]
Design head (ft)	250/200[(a)]
Material	SS
Waste evaporator condensate pump	
Number	1 (per unit)
Type	Canned centrifugal
Design pressure (psig)	150
Design temperature (°F)	200
Design flow (gal/min)	35/100[(a)]
Design head (ft)	250/200[(a)]
Material	SS
Chemical drain tank pump	
Number	1 (shared)
Type	Canned centrifugal
Design pressure (psig)	150
Design temperature (°F)	200
Design flow (gal/min)	35/100[(a)]
Design head (ft)	250/200[(a)]
Material	SS

TABLE 11.2.1-2 (SHEET 2 OF 8)

Spent resin sluice pump	
Number	1 (per unit)
Type	Canned centrifugal
Design pressure (psig)	240
Design temperature (°F)	200
Design flow (gal/min)	100/140[a]
Design head (ft)	300/250[a]
Material	SS
Laundry and hot shower tank pump	
Number	1 (shared)
Type	Mechanical seal, horizontal centrifugal
Design pressure (psig)	150
Design temperature (°F)	200
Design flow (gal/min)	35/100[a]
Design head (ft)	250/200[a]
Material	SS
Floor drain tank pump	
Number	1 (per unit)
Type	Mechanical seal, horizontal centrifugal
Design pressure (psig)	150
Design temperature (°F)	200
Design flow (gal/min)	35/100[a]
Design head (ft)	250/200[a]
Material	SS
Waste monitor tank pumps	
Number	2 (per unit)
Type	Canned centrifugal
Design pressure (psig)	150
Design temperature (°F)	200
Design flow (gal/min)	35/100[a]
Design head (ft)	250/200[a]
Material	SS

TABLE 11.2.1-2 (SHEET 3 OF 8)

Filters

Waste evaporator feed filter[h]	
Number	1 (per unit)
Waste evaporator condensate filter	
Number	1 (per unit)
Type	Disposable cartridge
Design pressure, (psig)	150[b]
Design temperature (°F)	200[b]
Design flow (gal/min)	35
Wp at design flow (psi)	5
Size of particles, 98% retention (μm) (nominal)	25
Materials	
Housing	SS
Filter element	Epoxy impregnated cellulose fiber
Spent resin sluice filter	
Type	Backflushable
Number	1 (per unit)
Design pressure (psig)	375
Design temperature (°F)	200
Design flow (gal/min)	150
Wp at design flow (psi) (unfouled)	5
Size of particles, 98% retention (μm) (nominal)	25
Materials	
Housing	SS
Filter element	Epoxy impregnated cellulose fiber or fine diameter 304L SS wires
Laundry and hot shower tank filter	
Number	1
Type	Disposable cartridge
Design pressure (psig)	150[b]
Design temperature (°F)	200[b]
Design flow (gal/min)	35
Wp at design flow (psi) (unfouled)	5
Size of particles, 98% retention (μm) (nominal)	25
Materials	
Housing	SS
Filter element	Epoxy impregnated[h] cellulose fiber

TABLE 11.2.1-2 (SHEET 4 OF 8)

Floor drain tank filter[h]	
Number	1 (per unit)
Type	Backflushable
Design pressure (psig)	375
Design temperature (°F)	200
Design flow (gal/min)	35
Wp at design flow (psi) (unfouled)	5
Size of particles, 98% retention (μm) (nominal)	25
Materials	
Housing	SS
Filter element	Chemically etched SS discs
Waste monitor tank filter	
Type	Backflushable
Number	1 (per unit)
Design pressure (psig)	375
Design temperature (°F)	200
Design flow (gal/min)	35
Wp at design flow (psi) (unfouled)	5
Size of particles, 98% retention (μm) (nominal)	25
Materials	
Housing	SS
Filter element	Chemically etched SS discs

<u>Strainers</u>

Laundry and hot shower tank strainer	
Number	1 (shared)
Type	Basket
Design pressure (psig)	150
Design temperature (°F)	200
Design flow (gal/min)	35
Δp at design flow (psi)	Negligible
Strainer mesh number	40
Materials	SS
Floor drain tank pump suction strainer	
Number	1 (per unit)
Type	Basket
Design pressure (psig)	150
Design temperature (°F)	200
Design flow (gal/min)	50
Δp at design flow (psi)	Negligible
Basket Perforation Size	5/32
Materials	SS

TABLE 11.2.1-2 (SHEET 5 OF 8)

Heat Exchanger

RCDT heat exchanger

Number		1
Type		U-tube
Estimated UA[d] (Btu/h/°F)		70,000
	Shell Side	Tube Side
Design pressure (psig)	150	240
Design temperature (°F)	250	200
Design flow (lb/h)	112,000	44,600
Temperature inlet (°F)	105	180
Temperature outlet (°F)	125	130
Material	Carbon steel	SS

Demineralizers

Waste evaporator condensate demineralizer

Number	1 (per unit)
Type	Flushable
Design pressure (psig)	150[b]
Design temperature (°F)	200[b]
Design flow (gal/min)	35
Resin volume (ft^3)	30[e]
Material	SS
Process decontamination factors©	100 for iodine; 2 for cesium; 100 for all other nuclides[f]

Waste monitor tank demineralizer

Number	1 (per unit)
Type	Flushable
Design pressure (psig)	150[b]
Design temperature (°F)	200[b]
Design flow (gal/min)	35
Resin volume (ft^3)	30[e]
Material	SS
Process decontamination factors[c]	100 for iodine; 2 for cesium; 100 for all other nuclides[f]

TABLE 11.2.1-2 (SHEET 6 OF 8)

Tanks

RCDT	
Number	1 (per unit)
Usable volume (gal)	350
Type	Horizontal
Internal design pressure (psig)	100
External design pressure (psig)	60
Design temperature (°F)	250
Material	SS
Waste holdup tank	
Number	1 (per unit)
Usable volume (gal)	10,000
Type	Vertical
Design pressure (psig)	Atmospheric
Design temperature (°F)	200
Material	SS
Waste evaporator condensate tank	
Number	1 (per unit)
Usable volume (gal)	5000
Type	Vertical with diaphragm
Design pressure (psig)	Atmospheric
Design temperature (°F)	200
Material	SS
Chemical drain tank	
Number	1 (shared)
Usable volume (gal)	600
Type	Vertical
Design pressure (psig)	Atmospheric
Design temperature (°F)	200
Material	SS
Spent resin storage tank	
Number	1 (per unit)
Usable volume (ft^3)[g]	350
Type	Vertical
Design pressure (psig)	150
Design temperature (°F)	200
Material	SS

TABLE 11.2.1-2 (SHEET 7 OF 8)

Laundry and hot shower tank	
Number	1 (shared)
Usable volume (gal)	10,000
Type	Vertical
Design pressure (psig)	Atmospheric
Design temperature (°F)	200
Material	SS
Floor drain tank	
Number	1 (per unit)
Usable volume (gal)	10,000
Type	Vertical
Design pressure (psig)	Atmospheric
Design temperature (°F)	200
Material	SS
Waste monitor tank	
Number	2 (per unit)
Usable volume (gal)	5000
Type	Vertical
Design pressure (psig)	Atmospheric
Design temperature (°F)	200
Material	SS
Auxiliary waste monitor tank	
Number	2 (shared)
Nominal volume (gal)	20,000
Type	Vertical
Design pressure (psig)	Atmospheric
Design temperature (°F)	200
Material	SS

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a. Denotes two design conditions.

b. These are minimum conditions required. Design conditions supplied are 300 psig and 250°F.

c. Decontamination factors are taken from "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Pressurized Water Reactors (PWR-GALE Code)," Office of Standards Development, U.S. Nuclear Regulatory Commission, NUREG-0017, April 1976.

TABLE 11.2.1-2 (SHEET 8 OF 8)

d. U is the overall heat transfer coefficient. A is the heat transfer area.

e. This is the design maximum loading. Smaller volumes may be used based on operational needs and type of resin used. Media loaded into the demineralizer are evaluated through the VEGP Chemical Control Program.

f. The process decontamination factor is 10 for iodine, cesium, and all other nuclides when used as a second demineralizer in a series.

g. Total for resin and liquid.

h. The internals have been removed from these filters. They remain in place as part of the piping system.

TABLE 11.2.1-3 (SHEET 1 OF 10)

ESTIMATED LIQUID WASTE PROCESSING SYSTEM ISOTOPIC INVENTORIES AND CONCENTRATIONS[(a)]

Line Number	Liquid Process Line Description	Annual Average Flowrate (gal/day)	Isotopic Concentration (μCi/g) Cr-51	Mn-54	Fe-55	Fe-59	Co-58
1	Reactor coolant pump seal leakoff to reactor coolant drain tank	300	1.90E-04	3.10E-05	1.60E-04	1.00E-04	1.60E-03
2	Reactor coolant drain tank outflow	300	1.88E-04	3.10E-05	1.60E-04	9.94E-05	1.59E-03
3	Drain header flow to waste holdup tank	222	1.64E-04	2.23E-05	2.67E-04	1.07E-04	1.66E-03
4	Sample sink flow to waste holdup tank	27	9.50E-05	1.55E-05	8.00E-05	5.00E-05	8.00E-04
5	Sum of lines 3 and 4	249	1.57E-04	2.16E-05	2.47E-04	1.01E-05	1.57E-03
6	Flow to waste holdup tank from containment sump	530	6.84E-05	1.12E-05	5.76E-05	3.60E-05	5.76E-04
12	Floor drain flow to floor drain tank	1137	1.66E-05	2.71E-06	1.40E-05	8.73E-06	1.40E-04
13	Building sump flow to floor drain tank	900	4.37E-05	7.13E-06	3.68E-05	2.30E-05	3.68E-04
14	Sum of lines 12 and 13	2037	2.86E-05	4.66E-06	2.41E-05	1.50E-05	2.41E-04
15	Lab rinse water flow to floor drain tank	164	9.50E-05	1.55E-05	8.00E-05	5.00E-05	8.00E-04
16	Flow from floor drain tank	2800	2.66E-05	4.35E-06	2.24E-05	1.40E-05	2.25E-04
19	Waste monitor tank demineralizer outlet flow	2800	2.66E-07	4.35E-08	2.24E-07	1.40E-07	2.25E-06
20	Flow from boron recycle system	500	1.57E-08	3.25E-09	1.71E-08	9.10E-09	1.55E-07
21	Sum of lines 19 and 20	3300	2.26E-07	3.90E-08	1.93E-07	1.20E-07	1.73E-06
22	Discharge flow from waste monitor tank No. 2	3300	2.26E-07	3.90E-08	1.93E-07	1.20E-07	1.93E-06
23	Flow of contaminated samples to chemical drain tank	9	2.81E-04	4.59E-05	2.37E-04	1.48E-04	2.37E-03
24	Chemical drain tank outflow	9	2.39E-04	4.52E-05	2.36E-04	1.34E-04	2.22E-03
25	Flow to laundry and hot shower tank	500	0.00E+00	1.61E-06	0.00E+00	0.00E+00	6.44E-06
26	Discharge from waste monitor tank No. 1	500	0.00E+00	1.61E-06	0.00E+00	0.00E+00	6.44E-06
27	Sum of lines 22 and 26	3800	1.96E-07	2.46E-07	1.68E-07	1.04E-07	2.53E-06

TABLE 11.2.1-3 (SHEET 2 OF 10)

Isotopic Concentration (μCi/g)

Line Number	Co-60	Br-83	Br-84	Br-85	Rb-86	Rb-88	Sr-89	Sr-90	Y-90
1	2.00E-04	5.40E-04	3.00E-04	3.40E-05	4.45E-05	1.15E-01	3.60E-05	1.00E-06	1.20E-07
2	2.00E-04	1.38E-04	2.11E-05	2.42E-07	4.38E-05	4.68E-03	3.58E-05	1.00E-06	2.07E-07
3	3.22E-04	3.80E-04	2.11E-04	2.39E-05	6.30E-06	1.62E-02	3.84E-05	1.83E-06	5.26E-07
4	1.00E-04	2.70E-04	1.50E-04	1.70E-05	4.45E-06	1.15E-02	1.80E-05	5.00E-07	6.00E-08
5	2.98E-04	3.68E-04	2.04E-04	2.32E-05	6.10E-06	1.57E-02	3.62E-05	1.69E-06	4.75E-07
6	7.20E-05	1.94E-04	1.08E-04	1.22E-05	3.20E-06	8.28E-03	1.30E-05	3.60E-07	4.32E-08
12	1.75E-05	4.67E-05	2.59E-05	2.94E-06	7.77E-07	2.00E-03	3.14E-06	8.73E-08	1.09E-08
13	4.60E-05	1.24E-04	6.90E-05	7.82E-06	2.05E-06	5.29E-03	8.28E-06	2.30E-07	2.76E-08
14	3.01E-05	8.09E-05	4.49E-05	5.10E-06	1.34E-06	3.45E-03	5.41E-06	1.50E-07	1.83E-08
15	1.00E-04	2.70E-04	1.50E-04	1.70E-05	4.45E-06	1.15E-02	1.80E-05	5.00E-07	6.00E-08
16	2.80E-05	7.47E-05	4.15E-05	4.70E-06	1.25E-06	6.61E-04	5.05E-06	1.40E-07	1.77E-08
19	2.80E-07	7.47E-07	4.15E-07	4.70E-08	6.25E-07	3.31E-04	5.05E-08	1.40E-09	1.77E-10
20	2.15E-08	3.32E-10	3.95E-11	4.18E-13	1.62E-08	4.22E-08	3.35E-09	1.08E-10	9.17E-11
21	2.41E-07	6.33E-07	7.52E-07	4.00E-08	5.33E-07	2.81E-04	4.33E-08	1.20E-09	1.64E-10
22	2.41E-07	6.33E-07	7.52E-07	4.00E-08	5.33E-07	2.81E-04	4.33E-08	1.20E-09	1.64E-10
23	2.96E-04	7.99E-04	4.44E-04	5.03E-05	1.32E-05	3.40E-02	5.33E-05	1.48E-06	1.78E-07
24	2.95E-04	8.64E-06	1.06E-06	1.13E-08	1.04E-05	4.55E-05	4.88E-05	1.48E-06	1.12E-06
25	1.45E-05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
26	1.45E-05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
27	2.12E-06	5.50E-07	3.06E-07	3.47E-08	4.63E-07	2.44E-04	3.76E-08	1.04E-09	1.41E-10

TABLE 11.2.1-3 (SHEET 3 OF 10)

Isotopic Concentration (μCi/g)

Line Number	Sr-91	Y-91m	Y-91	Y-93	Zr-95	Nb-95	Mo-99	Tc-99m	Ru-103
1	7.00E-05	4.10E-05	6.50E-06	3.70E-06	6.10E-06	5.10E-06	8.70E-03	5.20E-03	4.60E-06
2	4.07E-05	2.59E-05	6.68E-06	2.19E-06	6.07E-06	5.11E-06	7.88E-03	6.64E-03	4.57E-06
3	4.94E-05	2.89E-05	4.76E-06	2.60E-06	4.41E-06	3.62E-06	6.12E-03	3.65E-03	1.15E-05
4	3.50E-05	2.05E-05	3.25E-06	1.85E-06	3.05E-06	2.55E-06	4.35E-03	2.60E-03	2.30E-06
5	4.78E-05	2.80E-05	4.60E-06	2.52E-05	4.26E-06	3.50E-06	5.93E-03	3.54E-03	1.05E-05
6	2.52E-05	1.48E-05	2.34E-06	1.33E-06	2.20E-06	1.84E-06	3.13E-03	1.87E-03	1.66E-06
12	6.06E-06	3.55E-06	5.68E-07	3.20E-07	5.33E-07	4.45E-07	7.56E-04	4.93E-04	4.02E-07
13	1.61E-05	9.43E-06	1.49E-06	8.51E-07	1.40E-06	1.17E-06	2.00E-03	1.20E-03	1.06E-06
14	1.05E-05	6.15E-06	9.75E-07	5.55E-07	9.16E-07	7.65E-07	1.31E-03	8.05E-04	6.92E-07
15	3.50E-05	2.05E-05	3.25E-06	1.85E-06	3.05E-06	2.55E-06	4.35E-03	2.60E-03	2.30E-06
16	9.66E-06	5.68E-06	9.10E-07	5.12E-07	8.56E-07	7.16E-07	1.21E-03	8.20E-04	6.45E-07
19	9.66E-08	5.68E-08	9.10E-09	5.12E-09	8.56E-09	7.16E-09	1,21E-05	8.20E-06	6.45E-05
20	1.84E-10	1.18E-10	6.37E-10	1.02E-11	5.84E-10	5.60E-10	1.66E-07	1.75E-07	4.10E-10
21	8.20E-08	4.85E-08	7.82E-07	4.34E-09	7.37E-09	6.15E-09	1.03E-05	6.97E-06	5.52E-04
22	8.20E-08	4.85E-08	7.82E-07	4.34E-09	7.37E-09	6.15E-09	1.03E-05	6.97E-06	5.52E-04
23	1.04E-04	6.07E-05	9.62E-06	5.48E-06	9.03E-06	7.55E-06	1.29E-02	7.70E-03	6.81E-06
24	4.54E-06	2.91E-06	9.20E-06	2.52E-07	8.42E-06	7.68E-06	3.75E-03	3.95E-03	6.07E-06
25	0.00E+00	0.00E+00	0.00E+00	0.00E+00	2.25E-06	3.22E-06	0.00E+00	0.00E+00	2.25E-07
26	0.00E+00	0.00E+00	0.00E+00	0.00E+00	2.25E-06	3.22E-06	0.00E+00	0.00E+00	2.25E-07
27	7.12E-08	4.21E-08	6.79E-09	3.77E-09	3.03E-07	4.29E-07	8.94E-06	6.05E-06	3.44E-08

TABLE 11.2.1-3 (SHEET 4 OF 10)

Isotopic Concentration (μCi/g)

Line Number	Rh103m	Ru-106	Rh-106	Te125m	Te127m	Te-127	Te129m	Te-129	I-130
1	5.10E-06	1.00E-06	1.10E-06	2.90E-06	2.80E-05	9.20E-05	1.40E-04	1.80E-04	2.30E-04
2	4.63E-06	9.99E-07	9.99E-07	2.89E-06	2.79E-05	6.44E-05	1.39E-04	1.02E-04	1.48E-04
3	3.60E-06	7.16E-07	1.11E-06	2.05E-06	1.99E-05	6.49E-05	9.87E-05	1.29E-04	1.63E-04
4	2.55E-06	5.00E-07	5.50E-07	1.45E-06	1.40E-05	4.60E-05	7.00E-05	9.00E-05	1.15E-04
5	3.49E-06	6.93E-07	1.05E-06	1.98E-06	1.93E-05	6.29E-05	9.56E-05	1.25E-04	1.58E-04
6	1.84E-06	3.60E-07	3.96E-07	1.04E-06	1.01E-05	3.31E-05	5.04E-05	6.48E-05	8.28E-05
12	4.45E-07	8.73E-07	9.60E-08	2.53E-07	2.45E-06	7.98E-06	1.22E-05	1.56E-05	1.99E-05
13	1.17E-06	2.30E-07	2.53E-07	6.67E-07	6.44E-06	2.12E-05	3.22E-05	4.14E-05	5.29E-05
14	7.65E-07	1.50E-07	1.65E-07	4.36E-07	4.21E-06	1.38E-05	2.10E-05	2.70E-05	3.45E-05
15	2.55E-06	5.00E-07	5.50E-07	1.45E-06	1.40E-05	4.60E-05	7.00E-05	9.00E-05	1.15E-04
16	7.14E-07	1.59E-07	1.52E-07	4.07E-07	3.93E-06	1.28E-05	1.96E-05	2.51E-05	3.19E-05
19	7.14E-09	1.59E-09	1.52E-09	4.07E-09	3.93E-06	1.28E-07	1.96E-07	2.51E-05	3.19E-07
20	4.10E-10	1.05E-10	1.05E-10	2.74E-10	2.81E-09	2.96E-09	1.21E-08	7.78E-09	7.86E-09
21	6.13E-06	1.37E-09	1.30E-09	3.49E-09	3.38E-08	1.09E-07	1.68E-07	2.14E-07	2.72E-07
22	6.13E-06	1.37E-09	1.30E-09	3.49E-09	3.38E-08	1.09E-07	1.68E-07	2.14E-07	2.72E-07
23	7.55E-06	1.48E-06	1.63E-06	4.29E-06	4.14E-05	1.36E-04	2.07E-04	2.66E-04	3.40E-04
24	6.08E-06	1.46E-06	1.46E-06	3.97E-06	3.97E-05	4.35E-05	1.82E-04	1.17E-04	1.92E-05
25	0.00E+00	3.86E-06	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
26	0.00E+00	3.86E-06	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
27	5.32E-06	5.09E-07	1.13E-09	3.03E-09	2.94E-08	9.47E-08	1.46E-07	1.86E-07	2.36E-07

TABLE 11.2.1-3 (SHEET 5 OF 10)

Isotopic Concentration (μCi/g)

Line Number	Te131m	Te-131	I-131	Te-132	I-132	I-133	I-134	I-135	Cs-134
1	2.60E-04	1.20E-04	2.80E-02	2.80E-03	1.10E-02	4.00E-02	5.30E-03	2.10E-02	1.30E-02
2	2.11E-04	4.26E-05	2.70E-02	2.57E-03	4.66E-03	2.99E-02	5.92E-04	1.03E-02	1.30E-02
3	1.83E-04	8.43E-04	2.10E-02	1.97E-03	7.74E-03	2.84E-02	3.43E-03	1.48E-02	2.04E-03
4	1.30E-04	6.00E-05	1.40E-02	1.40E-03	5.50E-03	2.00E-02	2.65E-03	1.05E-02	1.30E-03
5	1.77E-04	8.17E-05	2.02E-02	1.91E-03	7.50E-03	2.75E-02	3.61E-03	1.43E-02	1.96E-03
6	9.36E-05	4.32E-05	1.01E-02	1.01E-03	3.96E-03	1.44E-02	1.91E-03	7.56E-03	9.36E-04
12	2.25E-05	1.04E-05	2.44E-03	2.44E-04	9.53E-03	3.47E-03	4.58E-04	1.82E-03	2.27E-04
13	5.98E-05	2.76E-05	6.44E-03	6.44E-04	2.53E-03	9.20E-03	1.22E-03	4.83E-03	5.98E-04
14	3.90E-05	1.80E-05	4.21E-03	4.21E-04	1.65E-03	6.00E-03	7.95E-04	3.15E-03	3.91E-04
15	1.30E-04	6.00E-05	1.40E-02	1.40E-03	5.50E-03	2.00E-02	6.65E-03	1.05E-02	1.30E-03
16	3.61E-05	1.66E-05	3.92E-03	3.91E-04	1.52E-03	1.54E-03	7.33E-04	2.90E-03	3.65E-44
19	3.61E-07	1.66E-07	3.92E-05	3.91E-06	1.52E-05	1.54E-05	7.33E-06	2.90E-05	1.83E-04
20	2.19E-09	4.07E-10	1.33E-05	6.18E-08	6.84E-07	2.31E-06	1.17E-08	3.73E-07	6.93E-06
21	3.07E-07	1.41E-07	3.53E-05	3.33E-06	1.30E-05	1.34E-05	6.22E-06	2.47E-05	1.56E-04
22	3.07E-07	1.41E-07	3.53E-05	3.33E-06	1.30E-06	1.34E-05	6.22E-06	2.47E-05	1.56E-04
23	3.85E-04	1.78E-04	4.14E-02	4.14E-03	1.63E-02	5.92E-02	7.84E-03	3.11E-02	3.85E-03
24	5.20E-05	9.69E-06	2.47E-02	1.37E-03	1.54E-03	5.55E-03	3.11E-05	9.34E-04	3.82E-03
25	0.00E+00	0.00E+00	9.66E-09	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	2.09E-05
26	0.00E+00	0.00E+00	9.66E-09	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	2.09E-05
27	2.67E-07	1.22E-07	3.05E-05	2.89E-06	1.13E-05	1.16E-05	5.40E-06	2.14E-05	1.38E-04

TABLE 11.2.1-3 (SHEET 6 OF 10)

Isotopic Concentration (μCi/g)

Line Number	Cs-136	Cs-137	Ba137m	Ba-140	La-140	Ce-141	Ce-143	Pr-143	Ce-144
1	7.00E-03	9.50E-03	1.80E-03	2.20E-05	1.60E-05	7.10E-06	4.20E-06	5.10E-06	3.30E-06
2	6.85E-03	9.50E-03	8.84E-03	2.15E-05	1.68E-05	7.04E-06	3.47E-06	5.07E-06	3.30E-06
3	9.94E-04	1.51E-03	1.43E-03	1.74E-05	2.08E-05	5.01E-06	2.95E-06	3.59E-06	2.36E-06
4	7.00E-04	9.50E-04	9.00E-04	1.10E-05	8.00E-06	3.55E-06	2.10E-06	2.55E-06	1.65E-06
5	9.62E-04	1.45E-03	1.37E-03	1.67E-05	1.94E-05	4.85E-06	2.86E-06	3.48E-06	2.28E-06
6	5.04E-04	6.84E-04	6.48E-04	7.92E-06	5.76E-06	2.56E-06	1.51E-06	1.84E-06	1.19E-06
12	1.22E-04	1.66E-04	1.57E-04	1.92E-06	1.40E-06	6.20E-07	3.64E-07	4.45E-07	2.88E-07
13	3.22E-04	4.37E-04	4.14E-04	5.06E-06	3.68E-06	1.63E-06	9.66E-07	1.17E-06	7.59E-07
14	2.10E-04	2.86E-04	2.71E-04	3.31E-06	2.41E-06	1.07E-06	6.30E-07	7.65E-07	4.96E-07
15	7.00E-04	9.50E-04	9.00E-04	1.10E-05	8.00E-06	3.55E-06	2.10E-06	2.55E-06	1.65E-06
16	1.96E-04	2.67E-04	2.53E-04	3.08E-06	2.25E-06	9.95E-07	5.84E-07	7.15E-07	4.63E-07
19	9.80E-05	1.34E-04	2.53E-06	3.08E-08	2.25E-08	9.95E-09	5.84E-09	7.15E-09	4.63E-09
20	2.19E-06	5.11E-06	4.78E-06	1.37E-09	1.47E-09	6.07E-10	3.90E-11	3.52E-10	3.46E-10
21	8.35E-05	1.14E-04	2.87E-06	2.63E-08	1.93E-08	8.53E-09	4.96E-09	6.12E-09	3.98E-09
22	8.35E-05	1.14E-04	2.87E-06	2.63E-08	1.93E-08	8.53E-09	4.96E-09	6.12E-09	3.98E-09
23	2.07E-03	2.81E-03	2.66E-03	3.26E-08	2.37E-05	1.05E-05	6.22E-05	7.55E-06	4.88E-06
24	1.48E-03	2.81E-03	2.63E-03	2.32E-05	2.42E-05	9.15E-06	9.24E-07	5.88E-06	4.81E-06
25	0.00E+00	3.86E-05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	8.05E-06
26	0.00E+00	3.86E-05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	8.05E-06
27	7.25E-05	1.04E-04	2.49E-06	2.28E-08	1.68E-08	7.41E-09	4.31E-09	5.31E-09	1.06E-06

TABLE 11.2.1-3 (SHEET 7 OF 10)

Isotopic Concentration (μCi/g)

Line Number	Pr-144	Np-239	TOTAL	H-3
1	3.80E-06	1.20E-04	2.73E-01	1.00E+00
2	3.32E-06	1.07E-04	1.36E-01	1.00E+00
3	2.72E-06	8.43E-05	1.13E-01	7.03E-02
4	1.90E-06	6.00E-05	7.86E-02	5.00E-02
5	2.63E-06	8.17E-05	1.10E-01	6.81E-02
6	1.37E-06	4.32E-05	5.66E-02	3.60E-02
12	3.31E-07	1.04E-05	1.37E-02	8.87E-03
13	8.74E-07	2.76E-05	3.62E-02	2.30E-02
14	5.71E-07	1.80E-05	2.36E-02	1.51E-02
15	1.90E-06	6.00E-05	7.86E-02	5.00E-02
16	5.32E-07	5.25E-07	1.54E-02	1.38E-02
19	5.32E-09	5.25E-09	8.85E-04	1.38E-04
20	3.46E-10	1.92E-09	3.64E-05	9.98E-01
21	4.57E-09	4.75E-09	7.85E-04	1.51E-01
22	4.57E-09	4.75E-09	7.85E-04	1.51E-01
23	5.62E-06	1.78E-04	2.33E-01	1.48E-01
24	4.81E-06	4.42E-05	5.65E-02	1.48E-01
25	0.00E+00	0.00E+00	0.00E+00	0.00E+00
26	0.00E+00	0.00E+00	1.01E-04	0.00E+00
27	3.97E-09	4.12E-09	6.94E-04	1.31E-01

TABLE 11.2.1-3 (SHEET 8 OF 10)

Tank Number	Tank Description	Liquid[c] Volume (gal)	Tank Inventories (Ci) Cr-51	Mn-54	Fe-55	Fe-59	Co-58
1	Reactor coolant drain tank	126	8.96E-05	1.48E-05	7.62E-05	4.74E-05	7.60E-04
2	Waste holdup tank	4000	3.00E-04	1.07E-02	2.82E-03	5.51E-04	6.60E-03
3	Waste evaporator condensate tank	4000	3.42E-04	5.90E-07	2.92E-06	1.82E-06	2.92E-05
4	Floor drain tank	4000	4.03E-04	6.59E-05	3.39E-04	2.12E-04	3.41E-03
5	Waste monitor tank No. 2	4000	3.42E-04	5.90E-07	2.92E-06	1.82E-06	2.92E-05
6	Chemical drain tank	510	2.17E-04	4.10E-05	2.14E-04	1.21E-04	2.02E-03
7	Laundry and hot shower tank	4000	0.00E+00	2.44E-05	0.00E+00	0.00E+00	9.74E-05
8	Waste monitor tank No. 1	4000	0.00E+00	2.44E-05	0.00E+00	0.00E+00	9.74E-05

Tank Inventories (Ci)

Tank Number	Co-60	Br-83	Br-84	Br-85	Rb-86	Rb-88	Sr-89	Sr-90	Y-90
1	9.53E-05	6.58E-05	1.01E-05	1.15E-07	2.09E-05	2.23E-03	1.71E-05	4.77E-07	9.85E-08
2	2.57E-03	6.75E-04	8.40E-05	8.58E-07	1.24E-05	1.57E-06	9.64E-03	2.66E-08	7.66E-06
3	3.65E-06	9.58E-06	5.33E-06	6.06E-07	8.07E-06	4.25E-03	6.56E-07	1.82E-08	2.48E-09
4	4.24E-04	1.13E-03	6.28E-04	7.12E-05	1.89E-05	1.00E-02	7.65E-05	2.12E-06	2.68E-07
5	3.65E-06	9.58E-06	5.33E-06	6.06E-07	8.07E-06	4.25E-03	6.56E-07	1.82E-08	2.48E-09
6	2.68E-04	7.84E-06	9.61E-07	1.03E-08	9.42E-06	4.13E-05	4.43E-05	1.34E-06	1.01E-06
7	2.19E-04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
8	2.19E-04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00

Tank Inventories (Ci)

Tank Number	Sr-91	Y-91m	Y-91	Y-93	Zr-95	Nb-95	Mo-99	Tc-99m	Ru-103
1	1.94E-05	1.23E-05	3.18E-06	1.05E-06	2.89E-06	2.43E-06	3.76E-03	3.16E-03	2.18E-06
2	4.45E-07	3.07E-05	1.33E-04	4.33E-07	3.98E-05	2.38E-07	2.62E-05	4.45E-05	9.39E-05
3	1.24E-06	7.34E-07	1.18E-07	6.57E-08	1.13E-07	9.31E-08	1.56E-04	1.06E-04	8.36E-08
4	1.46E-04	8.60E-05	1.38E-05	7.75E-06	1.30E-05	1.08E-05	1.83E-02	1.24E-02	9.77E-06
5	1.24E-06	7.34E-07	1.18E-07	6.57E-08	1.13E-07	9.31E-08	1.56E-04	1.06E-04	8.36E-08
6	4.12E-06	2.64E-06	8.35E-06	2.28E-07	7.64E-06	6.97E-06	3.41E-03	3.58E-03	5.51E-06
7	0.00E+00	0.00E+00	0.00E+00	0.00E+00	3.41E-05	4.87E-05	0.00E+00	0.00E+00	3.41E-06
8	0.00E+00	0.00E+00	0.00E+00	0.00E+00	3.41E-05	4.87E-05	0.00E+00	0.00E+00	3.41E-06

TABLE 11.2.1-3 (SHEET 9 OF 10)

Tank Inventories (Ci)

Tank Number	Rh103m	Ru-106	Rh-106	Te125m	Te127m	Te-127	Te129m	Te-129	I-130
1	2.21E-06	4.76E-07	4.76E-07	1.38E-06	1.33E-05	3.07E-05	6.62E-05	4.85E-05	7.03E-05
2	2.97E-03	3.12E-05	4.16E-05	8.07E-08	7.34E-06	1.82E-05	8.66E-06	1.71E-03	9.02E-04
3	9.28E-05	2.07E-08	1.97E-08	5.28E-08	5.12E-07	1.65E-06	2.54E-06	3.24E-06	4.12E-06
4	1.08E-05	2.41E-06	2.30E-06	6.16E-06	5.95E-05	1.94E-04	2.97E-04	3.80E-04	4.82E-04
5	9.28E-05	2.07E-08	1.97E-08	5.28E-08	5.12E-07	1.65E-06	2.54E-06	3.24E-06	4.12E-06
6	5.52E-06	1.33E-06	1.33E-06	3.60E-06	3.61E-05	3.95E-05	1.65E-04	1.06E-04	1.74E-05
7	0.00E+00	5.84E-05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
8	0.00E+00	5.84E-05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00

Tank Inventories (Ci)

Tank Number	Te131m	Te-131	I-131	Te-132	I-132	I-133	I-134	I-135	Cs-134
1	1.01E-04	2.03E-05	1.29E-02	1.22E-03	2.22E-03	1.43E-02	2.82E-04	4.89E-03	6.19E-03
2	1.45E-05	6.16E-06	2.09E-01	2.80E-05	1.32E-02	1.85E-01	4.01E-07	1.53E-02	1.39E-01
3	4.65E-06	2.13E-06	5.34E-04	5.04E-05	1.97E-04	2.02E-04	9.42E-05	3.74E-04	2.36E-03
4	5.47E-04	2.51E-04	5.93E-02	5.92E-03	2.30E-02	2.33E-02	1.11E-02	4.39E-02	5.53E-03
5	4.65E-06	2.13E-06	5.34E-04	5.04E-05	1.97E-04	2.02E-04	9.42E-05	3.74E-04	2.36E-03
6	4.72E-05	8.80E-06	2.24E-02	1.25E-03	1.40E-03	5.04E-03	2.82E-05	8.48E-04	3.47E-03
7	0.00E+00	0.00E+00	1.46E-05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	3.17E-04
8	0.00E+00	0.00E+00	1.46E-05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	3.17E-04

Tank Inventories (Ci)

Tank Number	Cs-136	Cs-137	Ba137m	Ba-140	La-140	Ce-141	Ce-143	Pr-143	Ce-144
1	3.26E-03	4.53E-03	4.21E-03	1.03E-05	8.02E-06	3.35E-06	1.65E-06	2.41E-06	1.57E-06
2	1.53E-02	2.10E-07	4.50E-05	2.29E-04	2.45E-04	5.45E-04	1.11E-09	1.06E-03	8.37E-04
3	1.26E-03	1.73E-03	4.35E-05	3.98E-07	2.92E-03	1.29E-07	7.51E-08	9.27E-08	6.03E-08
4	2.97E-03	4.04E-03	3.83E-03	4.66E-05	3.41E-05	1.31E-05	8.84E-06	1.08E-05	7.01E-06
5	1.26E-03	1.73E-03	4.35E-05	3.98E-07	2.92E-03	1.29E-07	7.51E-08	9.27E-08	6.03E-08
6	1.35E-03	2.55E-03	2.39E-03	2.11E-05	2.20E-05	8.30E-06	8.39E-07	5.34E-06	4.36E-06
7	0.00E+00	5.84E-04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.22E-04
8	0.00E+00	5.84E-04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.22E-04

TABLE 11.2.1-3 (SHEET 10 OF 10)

Tank Inventories (Ci)

Tank Number	Pr-144	Np-239	TOTAL	H-3
1	1.58E-06	5.09E-05	6.48E-02	4.77E-01
2	3.95E-05	6.50E-04	7.45E-01	4.24E-05
3	6.92E-08	7.19E-08	1.19E-02	2.29E+00
4	8.05E-06	7.95E-06	2.33E-01	2.09E-01
5	6.92E-08	7.19E-08	1.19E-02	2.29E+00
6	4.36E-06	4.01E-05	5.13E-02	1.34E-01
7	0.00E+00	0.00E+00	1.52E-03	0.00E+00
8	0.00E+00	0.00E+00	1.52E-03	0.00E+00

a. Locations of lines and tanks are shown in the liquid waste processing system process flow diagram (drawings 1X4DB124, 1X4DB125, 1X4DB126, 1X4DB127, AX4DB124-2, and AX4DB124-3). Activities are based on expected reactor coolant activity levels given in section 11.1.

b. Optional flow path.

c. This volume does not represent the capacity of the tank but represents the expected liquid volume.

TABLE 11.2.1-4

SUMMARY OF TANK LEVEL INDICATION, LEVEL ANNUNCIATORS, AND OVERFLOWS

Tank	Level Indication Location	Alarm Location	Alarm	Overflow To
Volume control	MCP Local	MCP	Low, High	None
Batching	Local	MCP	Low	Waste holdup tank
Boric acid	MCP	MCP	Empty, Low-Low	Waste holdup tank
Recycle holdup	BRP Local	BRP	Low, High	Waste holdup tank
Boron injection (Unit 1 only)	None	None	None	None
Refueling water storage	MCP	MCP	Empty, Low-Low, Low, High	RWST Dike
Condensate storage	MCP AFP	MCP	High, Low, Low-Low	CST Dike
Reactor makeup water storage	MCP	MCP	High, Low, Low-Low	RMWST Dike
Waste holdup	LWPP Local	LWPP	High, High-High	Local drains
Waste evaporator condensate	LWPP Local	LWPP	Low, High	Waste holdup tank
Spent resin storage	LWPP	LWPP	Low, High	None
Chemical drain	LWPP DRP Local	LWPP	Low, High	Local drains
Floor drain	LWPP Local	LWPP	Low, High	Local drains
Waste monitor	LWPP Local	LWPP	Low, High	None
Auxiliary waste monitor	LWPP Local	LWPP	Low, High	Local drains
Laundry and hot shower	LWPP Local	LWPP	Low, High	None
Steam generator blowdown spent resin storage	SGBP	SGBP	Low, High	None

MCP = main control panel
BRP = boron recycle panel
LWPP = liquid waste processing panel
SGBP = steam generator blowdown processing panel

TABLE 11.2.2-1

RANGE OF MEASURED DECONTAMINATION
FACTORS FOR SELECTED ISOTOPES

Isotope	Minimum	Maximum
I-131	1.1×10^{1}	1.6×10^{4}
I-133	1.1×10^{1}	1.8×10^{4}
I-135	1.4×10^{1}	2.0×10^{4}
Cs-137	2.4	1.3×10^{3}
F-18	1.73×10^{1}	1.5×10^{3}
Co-58	3.2×10^{1}	8.2×10^{3}
Mn-54	$>2.5 \times 10^{1}$	$>1.3 \times 10^{2}$

These values were observed across mixed bed demineralizers containing cation resin in the lithium-7 form and anion resin in the borated form.

TABLE 11.2.3-1

CALCULATED RELEASES OF RADIOACTIVE MATERIALS IN LIQUID EFFLUENTS (PER UNIT) ASSUMING EXPECTED FUEL LEAKAGE

Nuclide	Half-Life (days)	Coolant Concentrations Primary (µCi/ml)	Secondary (µCi/ml)	Liquid` Effluents Annual Releases to Discharge Canal Boron RS (Ci)	Misc .Wastes (Ci)	Secondary (Ci)	Turb Bldg (Ci)	Total LWS (Ci)	Adjusted Total (Ci/year)	Detergent Wastes (Ci/year)	Total (Ci/year)
Corrosion and Activation Products											
Cr-51	2.78E+01	1.90E-03	4.19E-08	0.00000	0.00000	0.00000	0.00000	0.00000	0.00003	0.00000	0.00003
Mn-54	3.03E+02	3.10E-04	1.00E-08	0.00000	0.00000	0.00000	0.00000	0.00000	0.00001	0.00100	0.00100
Fe-55	9.50E+02	1.60E-03	3.50E-08	0.00000	0.00000	0.00000	0.00000	0.00000	0.00002	0.00000	0.00003
Fe-59	4.50E+01	1.00E-03	2.57E-08	0.00000	0.00000	0.00000	0.00000	0.00000	0.00002	0.00000	0.00002
Co-58	7.13E+01	1.60E-02	3.57E-07	0.00000	0.00000	0.00003	0.00000	0.00003	0.00025	0.00400	0.00420
Co-60	1.92E+03	2.00E-03	4.50E-08	0.00000	0.00000	0.00000	0.00000	0.00000	0.00003	0.00870	0.00870
Zr-95	6.50E+01	0.00E+00	0.00E+00	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000	0.00140	0.00140
Nb-95	3.50E+01	0.00E+00	0.00E+00	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000	0.00200	0.00200
Np-239	2.35E+00	1.20E-03	2.35E-08	0.00000	0.00000	0.00000	0.00000	0.00000	0.00002	0.00000	0.00002
Fission Products											
Br-83	1.00E-01	4.80E-03	6.75E-08	0.00000	0.00000	0.00000	0.00000	0.00001	0.00007	0.00000	0.00007
Rb-86	1.87E+01	8.50E-05	2.15E-09	0.00001	0.00000	0.00000	0.00000	0.00001	0.00009	0.00000	0.00009
Rb-88	1.24E-02	2.00E-01	6.97E-07	0.00000	0.00001	0.00050	0.00000	0.00051	0.00402	0.00000	0.00400
Y-91m	3.47E-02	3.60E-04	2.09E-08	0.00000	0.00000	0.00000	0.00000	0.00000	0.00001	0.00000	0.00001
Mo-99	2.79E+00	8.40E-02	2.22E-06	0.00000	0.00001	0.00016	0.00002	0.00019	0.00148	0.00000	0.00150
Tc-99m	2.50E-01	4.80E-02	7.91E-06	0.00000	0.00001	0.00057	0.00005	0.00062	0.00494	0.00000	0.00490
Ru-103	3.96E+01	4.50E-05	1.03E-09	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000	0.00014	0.00014
Ru-106	3.67E+02	1.00E-05	2.51E-10	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000	0.00240	0.00240
Ag-110m	2.53E+02	0.00E+00	0.00E+00	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000	0.00044	0.00044
Te-127	3.92E-01	8.50E-04	3.96E-08	0.00000	0.00000	0.00000	0.00000	0.00000	0.00003	0.00000	0.00002
Te-129m	3.40E+01	1.40E-03	3.12E-08	0.00000	0.00000	0.00000	0.00000	0.00000	0.00002	0.00000	0.00002
Te-129	4.79E-02	1.60E-03	3.76E-07	0.00000	0.00000	0.00003	0.00000	0.00003	0.00022	0.00000	0.00022
I-130	5.17E-01	2.10E-03	4.17E-08	0.00000	0.00001	0.00000	0.00000	0.00001	0.00009	0.00000	0.00009
Te-131m	1.25E+00	2.50E-03	5.10E-08	0.00000	0.00000	0.00000	0.00000	0.00000	0.00003	0.00000	0.00003
Te-131	1.74E-02	1.10E-03	4.12E-07	0.00000	0.00000	0.00003	0.00000	0.00003	0.00024	0.00000	0.00024
I-131	8.05E+00	2.70E-01	6.43E-06	0.00051	0.00248	0.00046	0.00063	0.00408	0.03241	0.00006	0.03200
Te-132	3.25E+00	2.70E-02	5.65E-07	0.00000	0.00000	0.00004	0.00001	0.00005	0.00038	0.00000	0.00038
I-132	9.58E-02	1.00E-01	5.07E-06	0.00000	0.00021	0.00036	0.00009	0.00066	0.00527	0.00000	0.00530
I-133	8.75E-01	3.80E-01	7.96E-06	0.00007	0.00149	0.00057	0.00065	0.00278	0.02211	0.00000	0.02200
I-134	3.67E-02	4.70E-02	3.91E-07	0.00000	0.00001	0.00003	0.00000	0.00003	0.00027	0.00000	0.00027
Cs-134	7.49E+02	2.50E-02	6.01E-07	0.00310	0.00067	0.00043	0.00001	0.00420	0.03340	0.01300	0.04600
I-135	2.79E-01	1.90E-01	3.44E-06	0.00001	0.00027	0.00025	0.00018	0.00071	0.00562	0.00000	0.00560
Cs-136	1.30E+01	1.30E-02	2.76E-07	0.00093	0.00032	0.00020	0.00000	0.00145	0.01153	0.00000	0.01200
Cs-137	1.10E+04	1.80E-02	4.00E-07	0.00225	0.00048	0.00029	0.00000	0.00302	0.02404	0.02400	0.04800
Ba-137m	1.77E-03	1.60E-02	7.92E-06	0.00211	0.00045	0.00057	0.00000	0.00313	0.02486	0.00000	0.02500
Ce-144	2.84E+02	3.30E-05	1.00E-09	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000	0.00520	0.00520
Pr-144	1.20E-02	3.30E-05	1.75E-08	0.00000	0.00000	0.00000	0.00000	0.00000	0.00001	0.00000	0.00001
All Others		5.01E-03	8.27E-08	0.00000	0.00000	0.00001	0.00000	0.00001	0.00005	0.0	0.00005
Total (Except Tritium)		1.46E+00	4.56E-05	0.00900	0.00640	0.00454	0.00165	0.02158	0.17158	0.06234	0.23000
Tritium Release			720 Curies per year								

TABLE 11.2.3-2 (SHEET 1 OF 2)

COMPARISON OF CALCULATED CONCENTRATIONS
IN EFFLUENT WATER DISCHARGE WITH 10 CFR 20 LIMITS
ASSUMING EXPECTED FUEL LEAKAGE

Isotope	Annual Release to Discharge[(a)] Pipe (μCi)	Concentration in Circulating Water Discharge[(b)] (μCi/ml)	Maximum Permissible Concentration[(c)] (μCi/ml)	Fraction of Maximum Permissible Concentration
Cr-51	3.00E+01	1.26E-12	2.00E-03	6.28E-10
Mn-54	1.00E+03	4.19E-11	1.00E-04	4.19E-07
Fe-55	2.00E+01	8.38E-13	8.00E-04	1.05E-09
Fe-59	2.00E+01	8.38E-13	6.00E-05	1.40E-08
Co-58	4.20E+03	1.76E-10	1.00E-04	1.76E-06
Co-60	8.70E+03	3.64E-10	5.00E-05	7.29E-06
Zr-95	1.40E+03	5.86E-11	6.00E-05	9.77E-07
Nb-95	2.00E+03	8.38E-11	1.00E-04	8.38E-07
Np-239	2.00E+01	8.38E-13	1.00E-04	8.38E-09
Br-83	7.00E+01	2.93E-12	3.00E-06	9.77E-07
Rb-86	9.00E+01	3.77E-12	7.00E-05	5.39E-08
Rb-88	4.00E+03	1.68E-10	3.00E-06	5.58E-05
Y-91m	1.00E+01	4.19E-13	3.00E-05	1.40E-08
Mo-99	1.50E+03	6.28E-11	2.00E-04	3.14E-07
Tc-99m	4.90E+03	2.05E-10	6.00E-03	3.42E-08
Ru-103	1.40E+02	5.86E-12	8.00E-05	7.33E-08
Ru-106	2.40E+03	1.01E-10	1.00E-05	1.01E-05
Ag-110m	4.40E+02	1.84E-11	3.00E-05	6.14E-07
Te-127	2.00E+01	8.38E-13	6.00E-05	1.40E-08
Te-129m	2.00E+01	8.38E-13	3.00E-05	2.79E-08
Te-129	2.20E+02	9.21E-12	8.00E-04	1.15E-08
I-130	9.00E+01	3.77E-12	3.00E-06	1.26E-06
Te-131m	3.00E+01	1.26E-12	6.00E-05	2.09E-08
Te-131	2.40E+02	1.01E-11	3.00E-06	3.35E-06
I-131	3.20E+04	1.34E-09	3.00E-07	4.47E-03
Te-132	3.80E+02	1.59E-11	3.00E-05	5.31E-07
I-132	5.30E+03	2.22E-10	8.00E-06	2.77E-05
I-133	2.20E+04	9.21E-10	1.00E-06	9.21E-04
I-134	2.70E+02	1.13E-11	2.00E-05	5.65E-07
Cs-134	4.60E+04	1.93E-09	9.00E-06	2.14E-04
I-135	5.60E+03	2.35E-10	4.00E-06	5.86E-05
Cs-136	1.20E+04	5.03E-10	9.00E-05	5.58E-06
Cs-137	4.80E+04	2.01E-09	2.00E-05	1.01E-04

TABLE 11.2.3-2 (SHEET 2 OF 2)

Isotope	Annual Release to Discharge[a] Pipe (µCi)	Concentration in Circulating Water Discharge[b] (µCi/ml)	Maximum Permissible Concentration[c] (µCi/ml)	Fraction of Maximum Permissible Concentration
Ba-137m	2.50E+04	1.05E-09	3.00E-06	3.49E-04
Ce-144	5.20E+03	2.18E-10	1.00E-05	2.18E-05
Pr-144	1.00E+01	4.19E-13	3.00E-08	1.40E-05
All Others	5.00E+01	2.09E-12	3.00E-08	6.98E-05
Total	2.33E+05			5.99E-03
H-3	7.10E+08	2.97E-05	3.00E-03	9.91E-03
Total + H-3	7.10E+08			1.59E-02[d]

a. Total $\overline{C}_i$ (µCi/year) from table 11.2.3-1.

b. $C_i = \overline{C}_i / \overline{Q}_a$ where $\overline{Q}_a$ =15,000 gpm or 2.387×10^{13} ml/yr.

c. From 10 CFR 20, Appendix B, Table II, Column 2.

d. $\Sigma_i \, C_i/MPC_i \leq 1$.

TABLE 11.2.3-3 (SHEET 1 OF 2)

RADIOACTIVE LIQUID RELEASES FROM WESTINGHOUSE-DESIGNED PRESSURIZED WATER REACTOR PLANTS (ZIRCALOY FUEL CLADDING)

Plant	Year	Primary Coolant[a] Activity, Fraction of Design Basis	Total Annual Release (Ci)	Average Discharge Concentration (μCi/ml)
R. E. Ginna	1972	0.126	0.375	5.18×10^{-10}
	1973	0.023	0.074	9.74×10^{-11}
	1974	0.025	0.138	2.15×10^{-10}
	1975	0.013	0.420	6.09×10^{-10}
	1976	0.064	0.689	1.09×10^{-9}
	1977	0.033	0.065	9.01×10^{-11}
H. B. Robinson 2	1972	0.006	0.371	3.4×10^{-9}
	1973	0.003	0.306	2.1×10^{-9}
	1974	0.003	2.9	5.2×10^{-9}
	1975	0.002	0.44	6.79×10^{-10}
	1976	<0.001	0. 375	6.07×10^{-10}
	1977	<0.001	0. 329	4.7×10^{-10}
Point Beach 1 and 2	1972	0.062	0.934	1.9×10^{-9}
	1973	0.015	0.746	7.39×10^{-10}
	1974	0.101	0.196	3.29×10^{-10}
	1975	0.049	3.35	6.03×10^{-9}
	1976	0.012	3.24	6.17×10^{-9}
	1977	0.009	1.5	2.6×10^{-9}
Surry 1 and 2	1972	<0.001	0. 083	6.04×10^{-10}
	1973	0.001	0.145	2.65×10^{-10}
	1974	<0.001	29.2	2.32×10^{-8}
	1975	0.002	27.5	1.12×10^{-8}
	1976	0.011	33.7	1.45×10^{-9}
	1977	0.005	65.5	2.58×10^{-8}
Turkey Point 3 and 4	1973	<0.001	0.037	1.97×10^{-10}
	1974	0.010	1.69	3.21×10^{-9}
	1975	0.012	3.07	2.31×10^{-9}
	1976	0.006	8.65	2.92×10^{-10}
	1977	0.005	8.9	5.06×10^{-9}
Zion 1 and 2	1973	<0.001	<0.001	2.77×10^{-13}

TABLE 11.2.3-3 (SHEET 2 OF 2)

Plant	Year	Primary Coolant[a] Activity, Fraction of Design Basis	Total Annual Release (Ci)	Average Discharge Concentration (μCi/ml)
	1974	<0.001	0.005	8.03×10^{-12}
	1975	0.003	0.0087	1.18×10^{-11}
	1976	0.013	0.16	1.58×10^{-11}
	1977	0.009	0.95	8.33×10^{-10}
Prairie Island 1 and 2	1974	0.002	<0.001	2.29×10^{-12}
	1975	0.006	0.45	1.18×10^{-9}
	1976	0.008	<0.012	$<3.59 \times 10^{-11}$
	1977	0.008	0.013	1.41×10^{-11}
Kewaunee	1974	0.012	0.422	3.49×10^{-9}
	1975	0.005	0.447	2.68×10^{-9}
	1976	0.002	2.85	2.32×10^{-8}
	1977	<0.001	1.26	8.69×10^{-9}
Donald C. Cook 1	1975	<0.001	0.26	3.19×10^{-9}
	1976	0.003	1.87	2.69×10^{-8}
	1977	0.005	1.52	2.09×10^{-8}

a. Inferred by radioiodine in reactor coolant - includes diffusion and recoil sources.

TABLE 11.2.3-4

ESTIMATED INDIVIDUAL DOSES FROM LIQUID EFFLUENTS
AS LOW AS REASONABLY ACHIEVABLE[b]

Pathway	Usage (kg/yr h/yr)	Dilution	Time (h)	Shorewidth Factor = 0.2	ADULT DOSES (mrem/year Intake)[a]							
					Skin	Bone	Liver	Total Body	Thyroid	Kidney	Lung	Gi-LLi
Fish	21.0	10.0	24.01			9.51E-01	1.74E+00	1.30E+00	6.90E-02	5.88E-01	1.94E-01	1.30E-01
Drinking[c]	730.0	10.0	12.01			1.69E-02	2.13E -01	2.05E -01	3.42E-01	1.93E-01	1.86E-01	1.88E-01
Shoreline	12.0	10.0	0.01		1.21E-03	1.04E-03	1.04E -03	1.04E -03	1.04E-03	1.04E-03	1.04E-03	1.04E-03
Total					1.21E-03	9.69E-01	1.95E+00	1.51E+00	4.12E-01	7.82E-01	3.80E-01	3.19E-01

Pathway	Usage (kg/yr h/yr)	Dilution	Time (h)	Shorewidth Factor = 0.2	TEENAGE DOSES (mrem/year Intake)[a]							
					Skin	Bone	Liver	Total Body	Thyroid	Kidney	Lung	Gi-LLi
Fish	16.0	10.0	24.01			9.99E-01	1.78E+00	7.49E-01	6.39E-02	5.96E-01	2.26E-01	9.36E-02
Drinking[c]	510.0	10.0	12.01			1.63E-02	1.58E -01	1.41E-01	2.66E-01	1.39E-01	1.32E-01	1.32E-01
Shoreline	67.0	10.0	0.01		6.78E-03	5.80E-03	5.80E -03	5.80E-03	5.80E-03	5.80E-03	5.80E-03	5.80E-03
Total					6.78E-03	1.02E+00	1.95E+00	8.96E-01	3.36E-01	7.41E-01	3.64E-01	2.32E-01

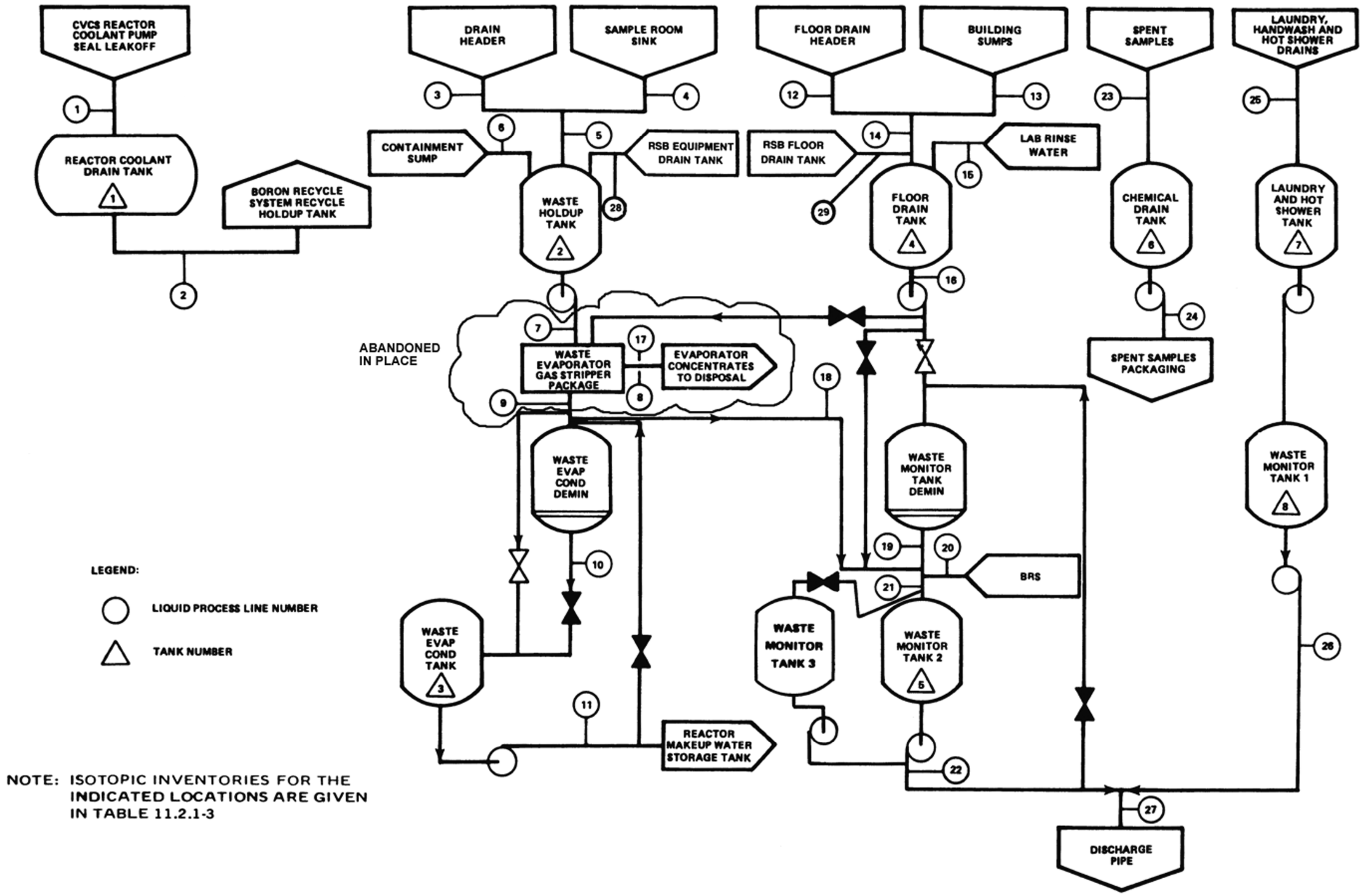
Pathway	Usage (kg/yr h/yr)	Dilution	Time (h)	Shorewidth Factor = 0.2	CHILD DOSES (mrem/year Intake)[a]							
					Skin	Bone	Liver	Total Body	Thyroid	Kidney	Lung	Gi-LLi
Fish	6.9	10.0	24.01			1.24E+00	1.55E+00	3.00E-01	6.60E-02	5.01E-01	1.78E-01	3.57E-02
Drinking[c]	610.0	10.0	12.01			4.68E -02	3.05E-01	2.58E-01	5.81E-01	2.67E-01	2.53E-01	2.50E-01
Shoreline	14.0	10.0	0.01		1.42E-03	1.21E -03	1.21E-03	1.21E-03	1.21E-03	1.21E-03	1.21E-03	1.21E-03
Total					1.42E-03	1.28E+00	1.85E+00	5.59E-01	6.48E-01	7.69E-01	4.32E-01	2.87E-01

Pathway	Usage (kg/yr h/yr)	Dilution	Time (h)	INFANT DOSES (mrem/year Intake[a]							
				Skin	Bone	Liver	Total Body	Thyroid	Kidney	Lung	Gi-LLi
Fish	0.0	10.0	24.01		0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Drinking[c]	330.0	10.0	12.01		4.90E -02	3.14E -01	2.49E -01	7.68E-01	2.63E -01	2.49E-01	2.44E -01
Shoreline				0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Total				0.00E+00	4.90E -02	3.14E -01	2.49E -01	7.68E-01	2.63E -01	2.49E-01	2.44E -01

a. Individual doses calculated using the LADTAP II code. All data is on a per unit basis.

b. Appendix I Design Objectives for Liquid Effluents: total body dose = 3 mrem/year per unit from all pathways; dose to any organ = 10 mrem/year per unit from all pathways. Docket RM-50-2 Annex to Appendix I Design Objectives: total body dose = 5 mrem/year per site from all pathways; dose to any organ = 5 mrem/year per site from all pathways; nontritium releases = 5 Ci/year per unit (table 11.2.3-1).

c. Although the dose due to the drinking water pathway has been included in this evaluation; currently no river water is used for potable water within 100 river miles of the site.



NOTE: ISOTOPIC INVENTORIES FOR THE INDICATED LOCATIONS ARE GIVEN IN TABLE 11.2.1-3

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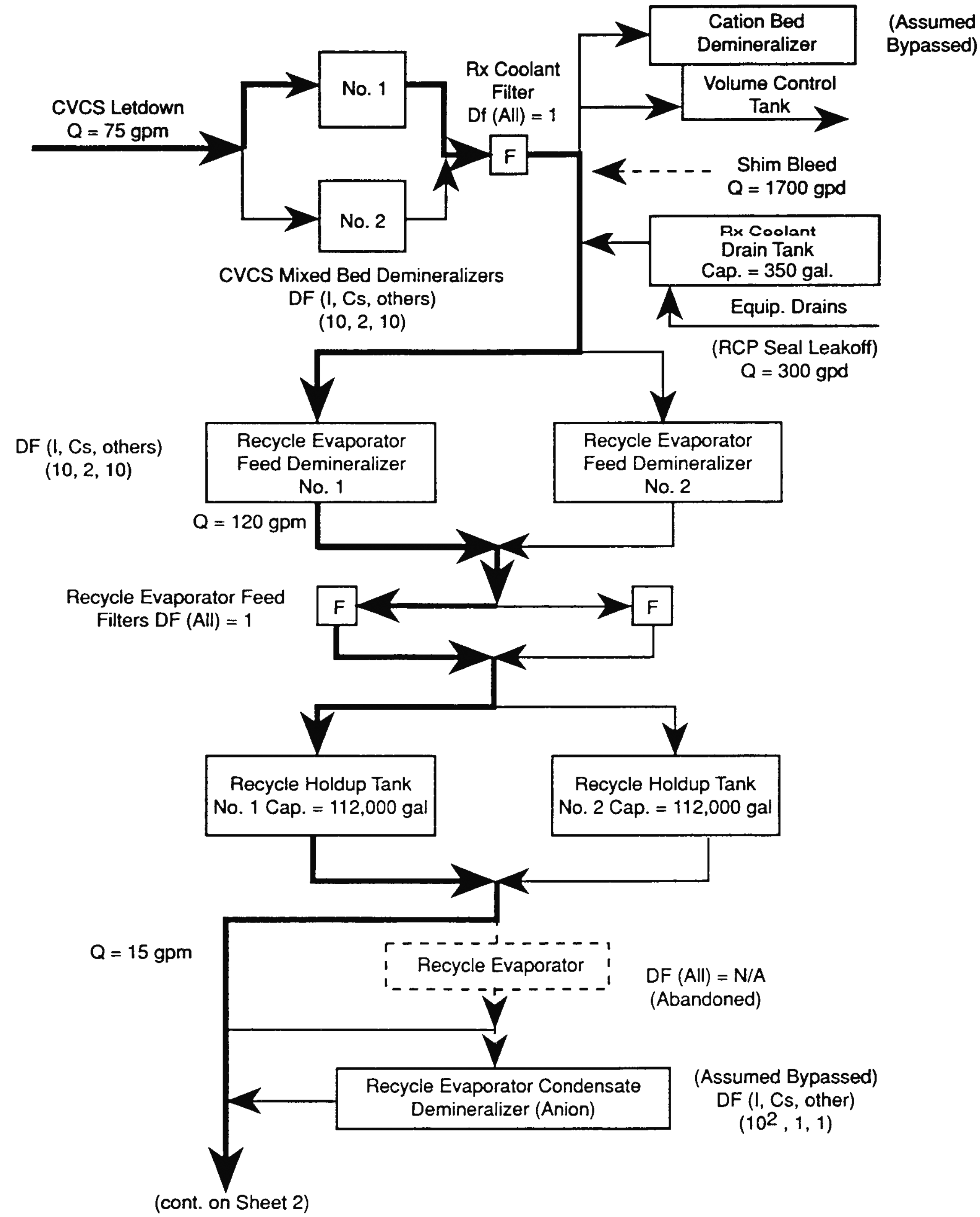
UNIT 1 AND UNIT 2

LWPS PROCESS FLOW DIAGRAM

FIGURE 11.2.2-1



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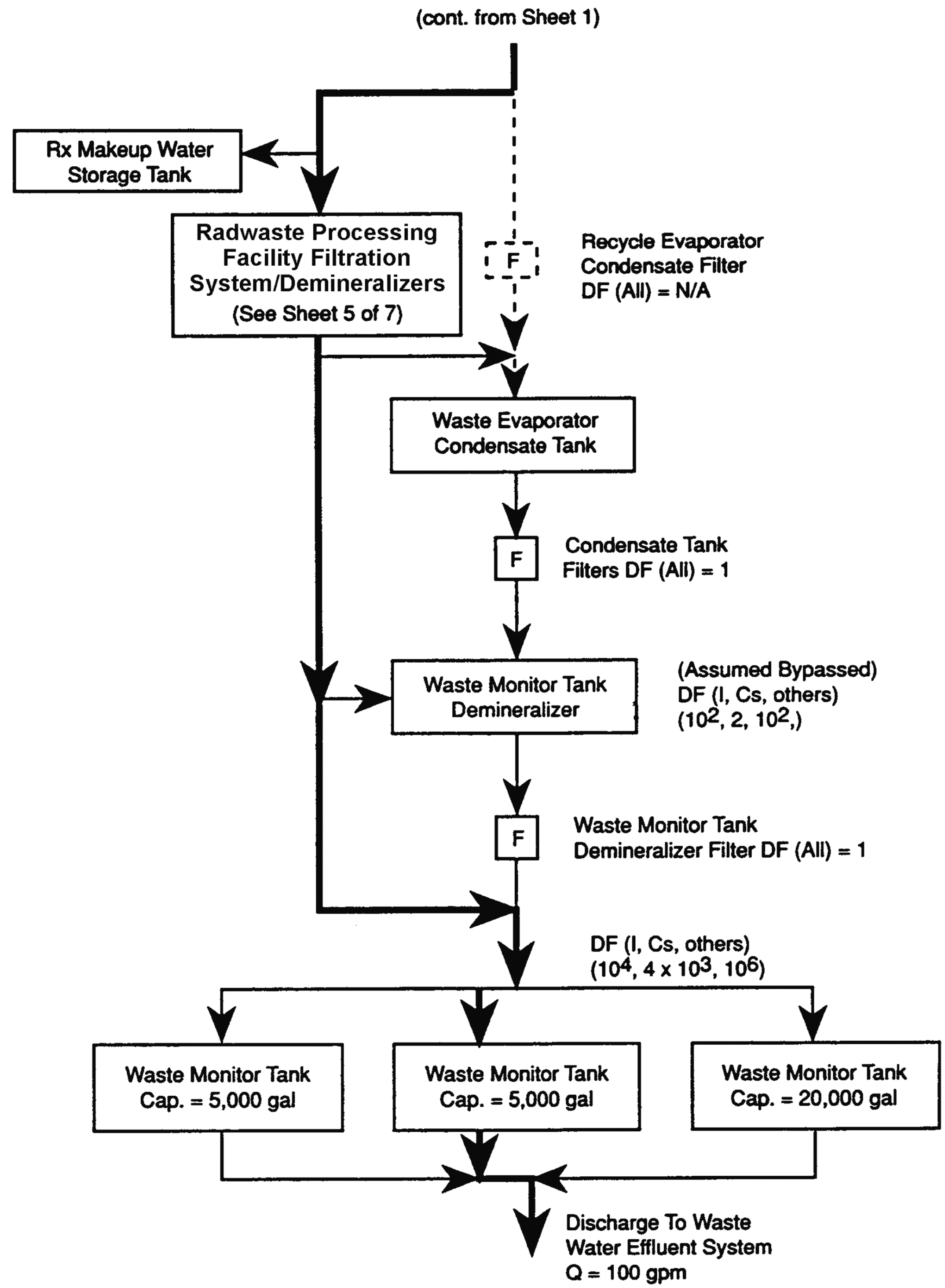
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

PROCESS FLOW MODEL (FOR CVCS LETDOWN & EQUIPMENT DRAINS)

FIGURE 11.2.3–1 (SHEET 1 OF 7)



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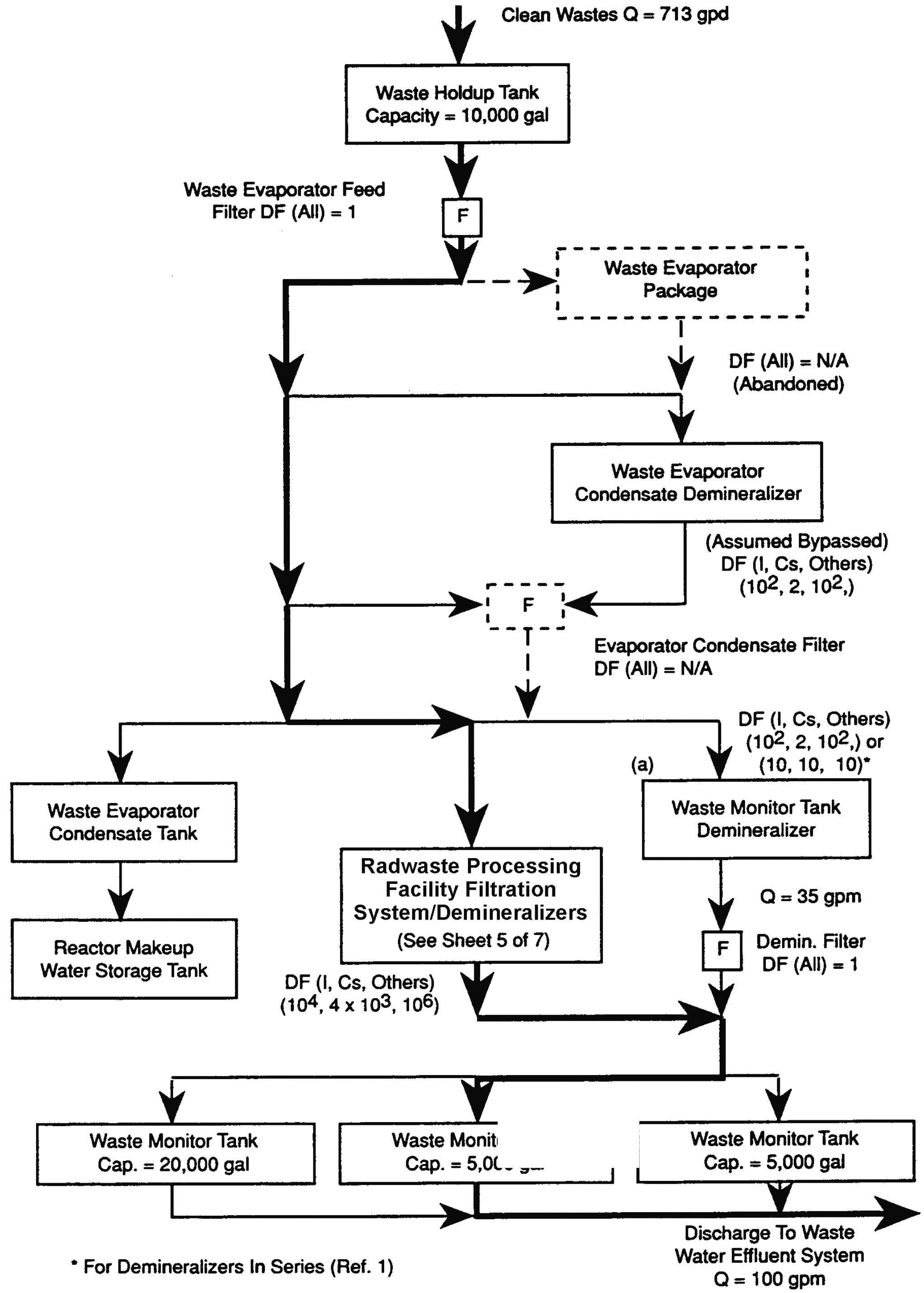
UNIT 1 AND UNIT 2

PROCESS FLOW MODEL (FOR CVCS LETDOWN & EQUIPMENT DRAINS)

FIGURE 11.2.3–1 (SHEET 2 OF 7)



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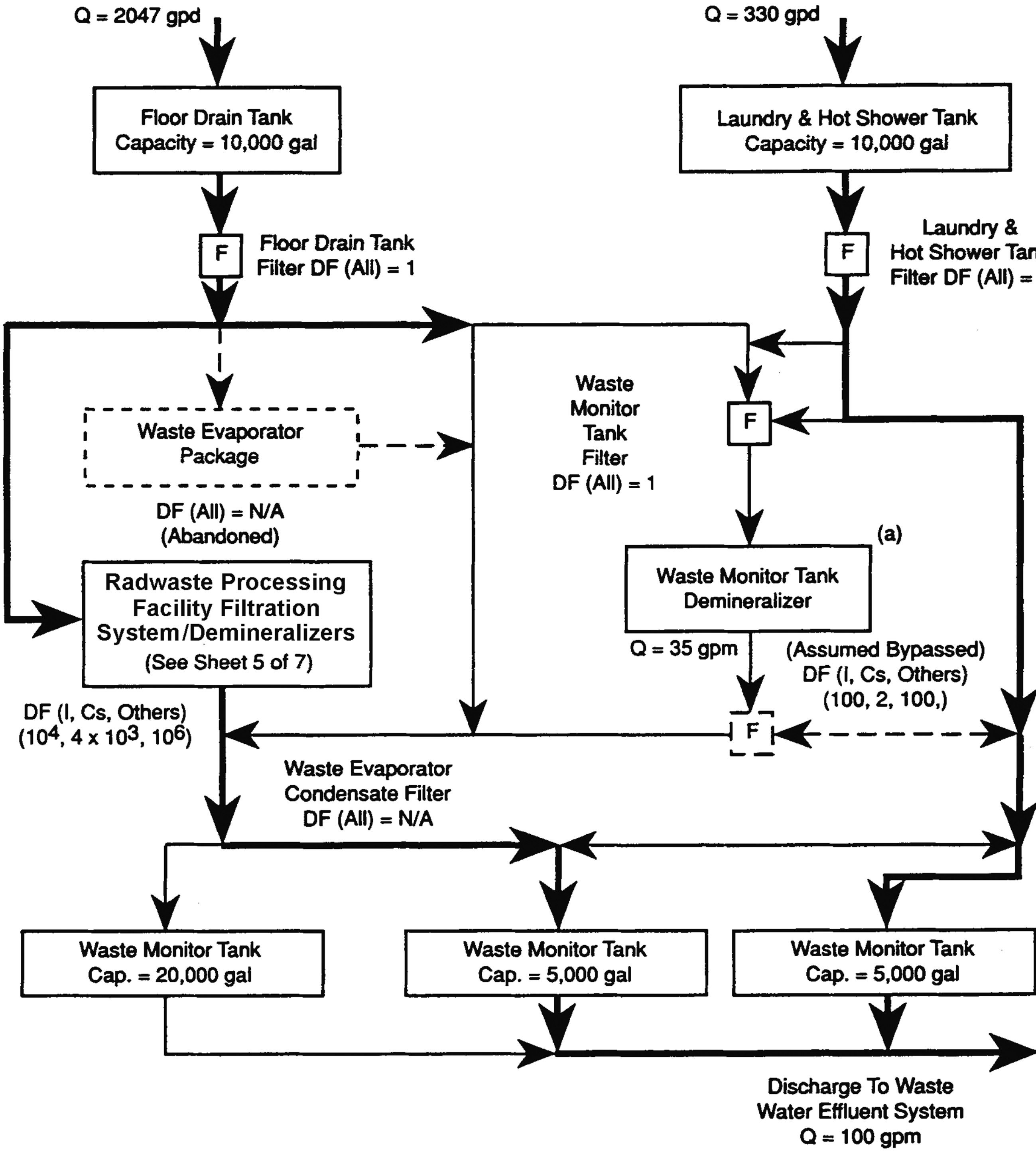
UNIT 1 AND UNIT 2

PROCESS FLOW MODEL (FOR CLEAN WASTE)

FIGURE 11.2.3–1 (SHEET 3 OF 7)



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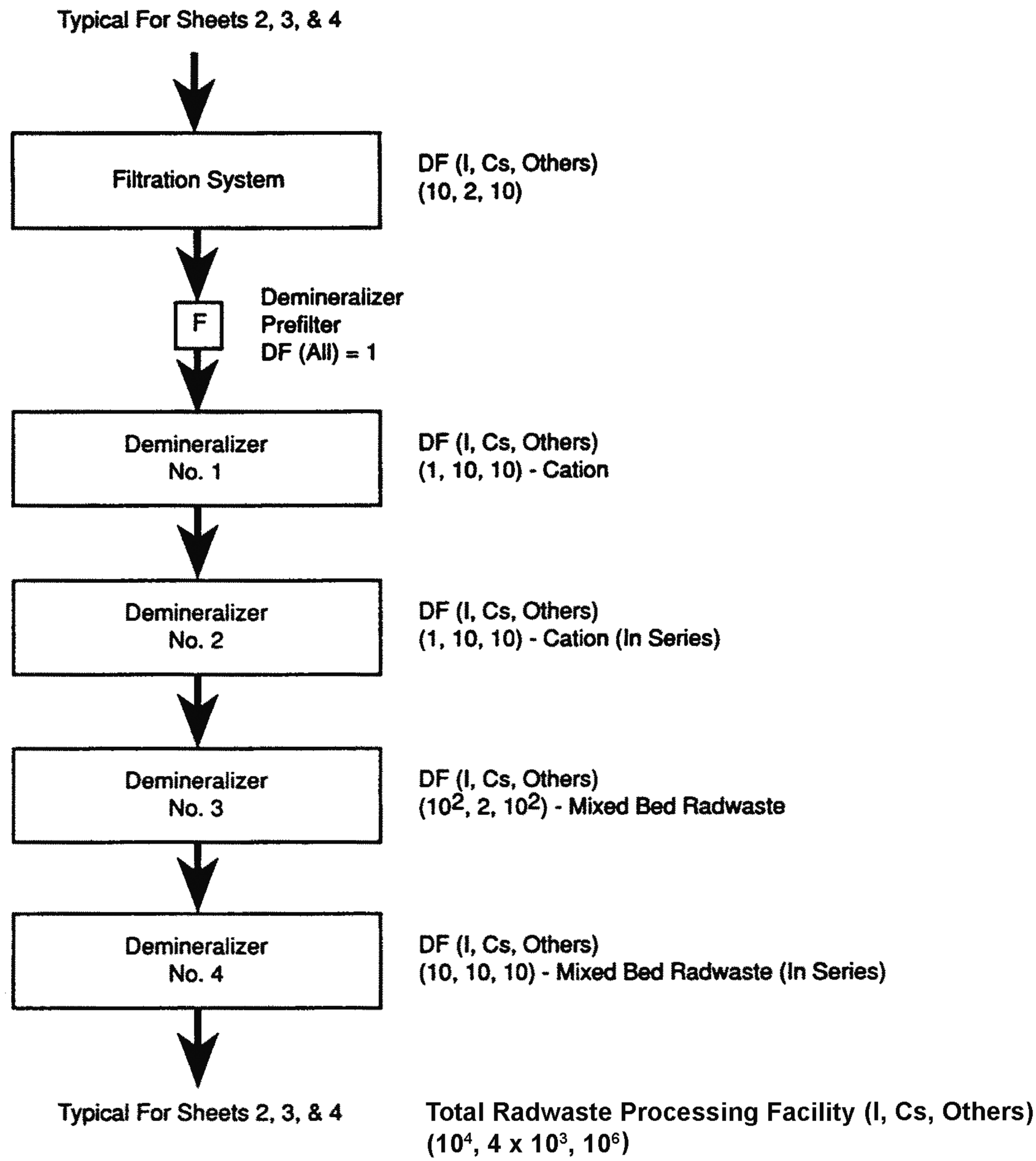
UNIT 1 AND UNIT 2

PROCESS FLOW MODEL
(FOR DIRTY WASTE & LAUNDRY)

FIGURE 11.2.3–1 (SHEET 4 OF 7)



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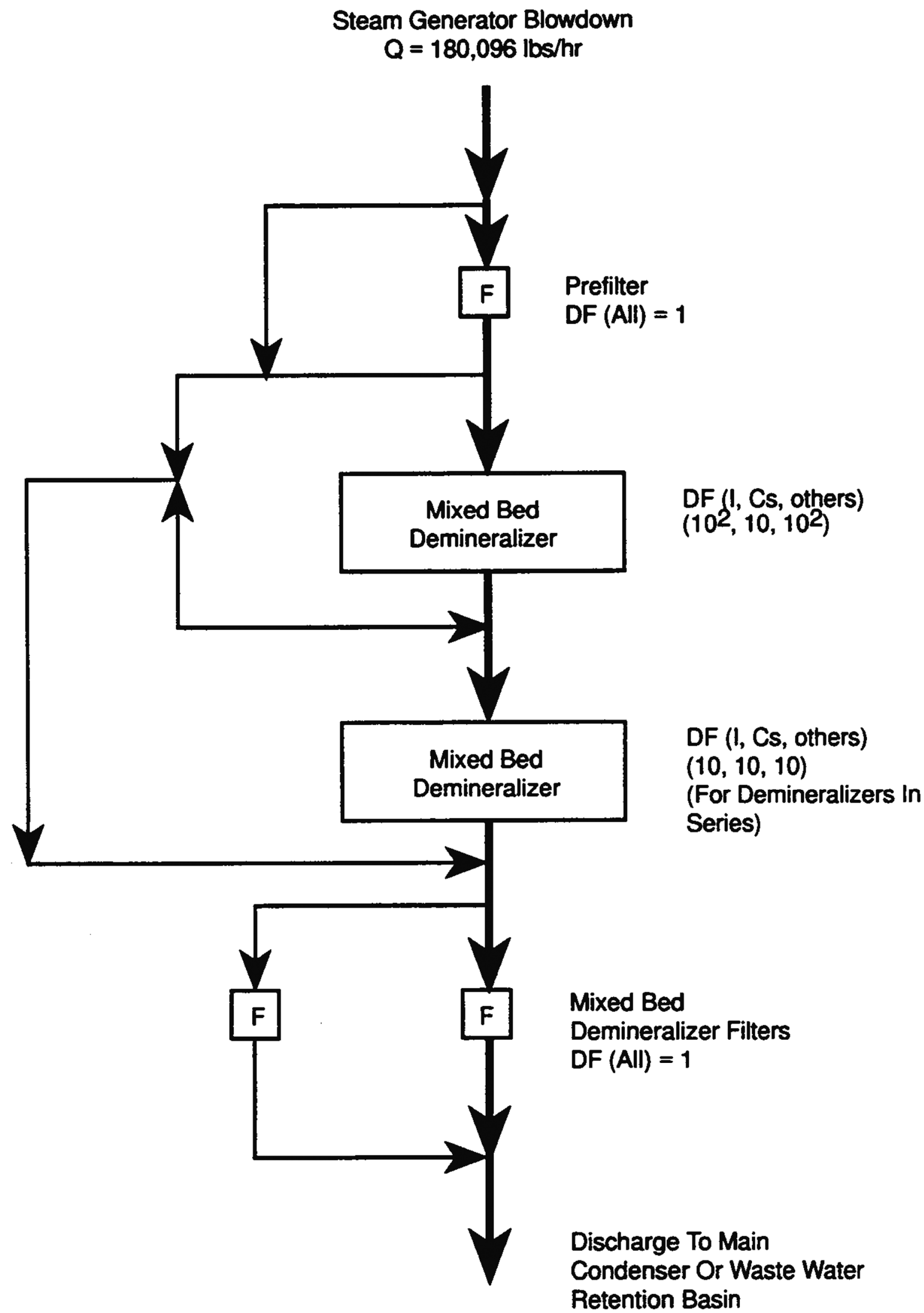
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

PROCESS FLOW MODEL (FOR RADWASTE PROCESSING FACILITY)

FIGURE 11.2.3–1 (SHEET 5 OF 7)



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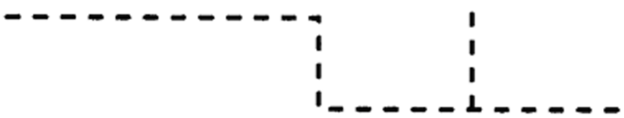
UNIT 1 AND UNIT 2

PROCESS FLOW MODEL (FOR STEAM GENERATOR BLOWDOWN)

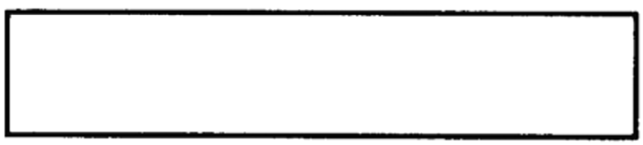
FIGURE 11.2.3–1 (SHEET 6 OF 7)

Normal Operation
Process Flow Path

Optional Process
Flow Path



Unavailable Or Deleted
Flow Path



Available Processing
Equipment

Unavailable Or Deleted
Processing Equipment

Q - Design Flow Rate

DF - Equipment Decontamination Factor For Radioactive Iodine (I), Cesium (Cs) And Other Miscellaneous Isotopes Defined By NUREG-0017 (Ref.1)

(a) - Note: Assumed Normally Bypassed

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SOUTHERN COMPANY *Energy to Serve Your World®*	VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2	PROCESS FLOW MODEL (LEGEND) FIGURE 11.2.3–1 (SHEET 7 OF 7)

11.3 GASEOUS WASTE MANAGEMENT SYSTEMS

The seismic classification and safety class information for the gaseous waste processing system (GWPS) are in section 3.2.

11.3.1 DESIGN BASES

The GWPS is designed to collect, process, and store gaseous wastes generated by plant operations including anticipated operational occurrences. The system is designed to assure that the release of gaseous effluents from the plant and expected offsite doses are as low as reasonably achievable (ALARA) as defined in Appendix I of 10 CFR 50. An evaluation of plant conformance to Appendix I is given in subsection 11.3.3. The GWPS has sufficient capacity and redundancy to keep releases within the discharge concentration limits of 10 CFR 20.1 - 20.601 during periods of design basis fuel leakage, as discussed in subsection 11.3.3.

In addition, the GWPS conforms to the requirements of General Design Criterion (GDC) 60 by providing long-term holdup capacity, thus precluding the necessity of releasing radioactive effluents during unfavorable environmental conditions. All gaseous effluent discharge paths are monitored for radioactivity, in compliance with GDC 64. Monitoring of radioactive effluents is discussed in section 11.5. The system's capability for long-term storage eliminates the need for frequent routine discharge of radioactive gases to the environment; it thereby allows any discharges to be made selectively under favorable environmental conditions.

During normal operation, the annual releases by leakage and planned discharges from the GWPS will be sufficiently low to control site boundary doses from the effluent to a small fraction of regulatory requirements.

The design of the GWPS is based on continuous operation of both Unit 1 and Unit 2 of VEGP, assuming that 1 percent of the rated core power is generated by fuel rods containing cladding defects. This condition is assumed to exist over the life of the plant[a].

While the normal inflow to the GWPS is 0.7 sf^3/min per unit, either of the two main process loops of the GWPS can accommodate surges such as may result from venting a recycle holdup tank or a reactor coolant drain tank. No other anticipated operational occurrence will cause a significant surge in GWPS process flow. However, the operator may elect to increase the volume control tank purge flow above the normal 0.7 sf^3/min per unit.

If waste gases are being released to the environment, the release is automatically terminated when the radioactivity level exceeds a predetermined level. The radiological monitoring and control instrumentation are described in section 11.5.

In order to control the release of radioactive gases resulting from equipment failure or operator error, the GWPS design has all waste gas decay tanks (WGDTs) (seven tanks for Unit 1, seven tanks for Unit 2, and two shared shutdown tanks) isolated from each other with valves. The tanks are in separate compartments. Thus, the maximum uncontrolled release would be limited

[a] The renewed operating licenses authorized a 20-year period of extended operation for both VEGP units, resulting in a total plant operating life of 60 years. Since the inventory in the Waste Gas Decay Tanks (WGDTs) has been routinely released during the first 20 years of operation and is expected to continue to be routinely released during future operation, the inventory of the WGDTs accumulated during the first 20 years of operation will be released prior to entering the period of extended operation. Therefore, the stated design capacity of the GWPS remains sufficient, and the analysis of the maximum fission product inventory in the GWPS over a 40-year plant life remains bounding for a 60-year plant life.

to the contents of one WGDT. The effects of this occurrence would be the same as the postulated WGDT rupture accident which is discussed in section 15.7. To reduce GWPS downtime resulting from equipment failures, the GWPS design includes redundant waste gas compressors (two per unit) and catalytic hydrogen recombiners (one per unit and one shared).

The system is designed to preclude the possibility of an internal explosion. However, the system volume is distributed among seven isolated tanks for each unit located in separate compartments so that the dose in the unlikely event of an explosion is the same as the dose due to a WGDT rupture.

The GWPS component design parameters and a description of process instrumentation and control are provided in subsection 11.3.2.

Design features incorporated in the GWPS to reduce leakage of radioactive gases are described in subsection 11.3.2.

Conformance of the GWPS design with the criteria of Regulatory Guide 1.143 is discussed in section 1.9.

11.3.2 SYSTEM DESCRIPTION

This section describes the design, operating features, and performance of the gaseous waste processing system (GWPS) and other plant gaseous waste management systems with respect to the collection and control of radioactive gases. Detailed descriptions of the plant ventilation systems and condenser vacuum system are presented in sections 9.4 and 10.4, respectively.

The GWPS is provided with two main process loops. Each processes gaseous wastes from one of the two plant units. Interconnections are provided to allow either process loop to process waste gas from either or both units.

All equipment in the GWPS is controlled from the waste processing panel. The GWPS consists mainly of two closed loops comprised of a waste gas compressor, a catalytic hydrogen recombiner, and seven waste gas decay tanks (GDTs) to accumulate the fission product gases. All pipes containing radioactive gases are shielded as necessary, and no piping is run through normally occupied areas.

Each main process loop also includes a GDT drain pump, six gas traps, and a waste gas drain filter. All of the equipment is located in the auxiliary building.

The piping and instrumentation diagram for the system is shown in drawings 1X4DB128 and 1X4DB129. This diagram indicates safety classes for all components and piping.

The GWPS reduces the fission gas concentration in the reactor coolant system (RCS), which in turn reduces the escape of fission gases from the RCS during maintenance or through equipment leakage.

The primary source of radioactive gas to the GWPS is the volume control tank (VCT) purge. Smaller quantities of radioactive gas are received via the vent connections from the reactor coolant drain tank (RCDT), the pressurizer relief tank, and the recycle holdup tanks.

Although the GWPS functions to contain radioactive gases, at no time do the radioactive gases constitute more than a small fraction of the gases stored in the GDTs.

Since hydrogen is continuously removed in the hydrogen recombiner, this gas does not build up in the GWPS. The largest contributor to the nonradioactive gas accumulation is helium generated by a $B^{10}(n,\alpha)Li^{7}$ reaction in the reactor core. The second largest contributors are the impurities in the bulk hydrogen and oxygen supplies.

The expected accumulation rate is 575 sf^3/year per unit (1150 sf^3/year total), assuming the following:

- Two-unit operation.
- A 0.7-sf^3/min hydrogen purge for the VCT of each unit.
- An 80-percent plant load factor for each unit.
- 99.95-percent pure hydrogen.
- 99.5-percent pure oxygen.

At this rate of accumulation and assuming zero leakage from the GWPS, the 14 GDTs have sufficient combined capacity to hold all the gaseous wastes produced during 40 years[a] of plant operations without any releases to the environment. This assumes that the waste gas holdup tanks are operated with an initial charge of 5 psig of nitrogen and the pressure in the tanks is allowed to accumulate to 100 psig.

Operation of the system is such that fission gases from one unit are distributed throughout seven normal operation GDTs. Separation of the gaseous inventory into several tanks ensures that the allowable site boundary dose will not be exceeded in the event any one of the GDTs ruptures. Radiological consequences of such a postulated rupture are discussed in chapter 15.

The GWPS also provides sufficient capacity to hold indefinitely the gases generated during reactor shutdown. Nitrogen gas from previous shutdowns is contained in one of the shutdown GDTs. This is used to strip hydrogen from the RCS during subsequent shutdowns. The second shutdown tank is normally at low pressure and is used to accept relief valve discharges from the normal operation GDTs, the hydrogen recombiners, and the waste gas compressors.

Table 11.3.2-1, based on the RCS activities given in table 11.1-2, shows the maximum fission product inventory in the GWPS over the 40-year plant life[a]. Table 11.3.2-2, based on the RCS activities given in table 11.1-7, shows the expected fission product inventory in the GWPS over the 40-year life[a] assuming no releases from the system.

Figures 11.3.2-1 and 11.3.2-2 are based on the RCS activities given in tables 11.1-2 and 11.1-7, respectively; the figures show that the quantity of fission gas activity accumulated after 40 continuous years of operation[a] is about twice the activity accumulated after 30 days of operation. Most of the accumulated activity arises from short-lived isotopes reaching equilibrium in 1 month or less.

The difference between the 30-day and 40-year[a] accumulations is predominantly krypton-85. This accumulation of krypton-85 is not a hazard to the plant operator because:

A. Krypton-85 is principally a beta emitter, for which the tanks themselves provide adequate shielding.

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[a] The renewed operating licenses authorized a 20-year period of extended operation for both VEGP units, resulting in a total plant operating life of 60 years. Since the inventory in the WGDTs has been routinely released during the first 20 years of operation and is expected to continue to be routinely released during future operation, the inventory of the WGDTs accumulated during the first 20 years of operation will be released prior to entering the period of extended operation. Therefore, the stated design capacity of the GWPS remains sufficient, and the analysis of the maximum fission product inventory in the GWPS over a 40-year plant life remains bounding for a 60-year plant life.

B. The activity of the system's inventory is distributed among seven normal operation GDTs, minimizing inventory in any single tank.

The removal of fission gas from the reactor coolant by the GWPS during normal operation reduces the plant activity levels caused by a leakage of reactor coolant. Operation of the GWPS allows the collection of virtually all the krypton-85 released to the reactor coolant and can reduce the fission product gas inventory in the RCS, as shown in table 11.3.2-3. Table 11.3.2-3 is based on the RCS activities given in table 11.1-2. Provisions are made to collect any gases from the RCDT and gases from the recycle holdup tanks.

Process flow diagrams are shown in figures 11.3.2-3 and 11.3.2-4. Table 11.3.2-4 gives process parameters for key locations in the GWPS for the normal operating mode with the normal operation GDTs at low pressure (less than 25 psig). This operating mode is used early in plant life and is shown in figure 11.3.2-3. Table 11.3.2-6 gives process parameters for key locations in the system for the normal operating mode at the end of plant life[a], when the waste GDTs are at high pressure. This operating mode, shown in figure 11.3.2-4, is used from the time the normal operation GDT pressure reaches 25 psig until the end of plant life[a].

Table 11.3.2-5 gives the process parameters for key locations in the GWPS based on a 90-day holdup of the waste gases.

11.3.2.1 <u>Component Design</u>

The gaseous waste processing equipment design parameters are given in table 11.3.2-7. Since the GWPS performs no function related to the safe shutdown of the plant, some components are classified as nonnuclear safety. Component safety classes and the corresponding code and code class are shown in section 3.2.

11.3.2.1.1 Waste Gas Compressor Packages

Two waste gas compressor packages are provided to circulate gases around each GWPS system loop. One is normally in use and the other on a standby basis.

The compressor packages are water-sealed centrifugal displacement machines which are skid mounted in a self-contained package. The waste gas compressor packages are primarily constructed of carbon steel. Mechanical seals are provided to minimize seal water leakage.

11.3.2.1.2 Catalytic Hydrogen Recombiner Packages

Three catalytic hydrogen recombiners are provided for the two units. One recombiner per unit is used in each main process loop to remove hydrogen from the hydrogen-nitrogen fission gas mixtures by oxidation to water vapor, which is removed by condensation. The third recombiner is available on a standby basis. The units are self-contained and are designed for continuous operation.

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[a] The renewed operating licenses authorized a 20-year period of extended operation for both VEGP units, resulting in a total plant operating life of 60 years. Since the inventory in the WGDTs has been routinely released during the first 20 years of operation and is expected to continue to be routinely released during future operation, the inventory of the WGDTs accumulated during the first 20 years of operation will be released prior to entering the period of extended operation. Therefore, the stated design capacity of the GWPS remains sufficient, and the analysis of the maximum fission product inventory in the GWPS over a 40-year plant life remains bounding for a 60-year plant life.

11.3.2.1.3 Waste Gas Decay Tanks

The tanks are vertical cylindrical type constructed of carbon steel. There are a total of 16 waste decay tanks. Seven are used by each unit during normal operation; the remaining two are shared between the two units and are used for shutdown and startup.

11.3.2.1.4 Gas Decay Tank Drain Pump

The waste GDTs may contain water from condensation from the process gas or from maintenance operations. This pump is used to drain water from the waste GDTs when there is insufficient pressure in the GWPS to transfer the fluid. Two canned motor pumps are provided, one for each set of seven normal operation GDTs.

11.3.2.1.5 Waste Gas Drain Filter

Two waste gas drain filters are provided, one to filter the drain water from each of the two process loops.

11.3.2.1.6 Waste Gas Traps

The gas traps serve to prevent undissolved gases from leaving the GWPS via the liquid drains. For each of the two process loops there are four gas traps - one in the compressors drain line, one in the recombiner drain line, and two mounted in the waste GDT drain line. There are also gas traps on each low point of the gaseous portion of the system to avoid condensate accumulation in the piping that may block the flow path.

11.3.2.1.7 Valves and Piping

Each valve in the hydrogen recombiner packages is designed to meet the temperature, pressure, and code requirements for the specific application in which it is used. The recombiner circuits contain manual valves provided with a metal diaphragm to prevent stem leakage and control valves provided with gaseous leakoffs returned to the GWPS. Other parts of the GWPS use elastomer diaphragm valves and control valves with bellows seals. Relief valves have soft seats and operate at pressures which are normally less than two-thirds of the relief valve set pressure. The relief valves of the major components discharge to one of the shutdown GDTs. This allows the discharge to be monitored before being released. It also provides a means of containing and detecting seal leakage across the relief valves.

All piping from the waste GDTs, up to and including the isolation valves, is designed to Seismic Category 1 requirements to preclude any accidental release of gas to the environment.

11.3.2.2 <u>Instrumentation and Control Design</u>

The GWPS instrumentation is shown on the piping and instrumentation diagram, drawings 1X4DB128 and 1X4DB129.

The instrumentation readout is located mainly on the waste processing system (WPS) panel, while some instruments are read locally.

All alarms are shown separately on the WPS panel and further relayed to one common WPS annunciator on the main control board.

Where suitable, instrument lines are provided with diaphragm seals to prevent fission gas leakage through the instrument.

11.3.2.2.1 Waste Gas Compressor Package Instrumentation and Control

Figure 11.3.2-5 shows the location of the instruments on the waste gas compressor package.

The compressors are interlocked with the seal water inventory in the moisture separators; they trip off on either high or low moisture separator level.

During normal operation the proper seal water inventory is maintained automatically.

11.3.2.2.2 Hydrogen Recombiner Package Instrumentation and Control

The catalytic hydrogen recombiner packages are designed for automatic operation with a minimum of operator attention.

Figure 11.3.2-6 shows the location of the instruments on the hydrogen recombiner. A multipoint temperature recorder monitors temperatures at several locations in the packages. Process gas flowrate is measured by an orifice located upstream of the hydrogen recombiner preheater. Local pressure gauges indicate the hydrogen recombiner process inlet pressure and the oxygen supply pressure.

The oxygen concentration is monitored and controlled to ensure that a flammable hydrogen-oxygen mixture does not occur. The GWPS is provided with two analyzers to monitor oxygen concentrations. One is between the oxygen supply and the hydrogen recombiner package, and one is downstream of the hydrogen recombiner. When the hydrogen concentration is above the lower flammability limit of 4 percent, the minimum concentration of oxygen necessary for deflagration is 5 percent. The control function assigned to these analyzers is to automatically terminate the oxygen supply before reaching GWPS oxygen concentrations favorable for hydrogen flammability. Each hydrogen recombiner package also includes two hydrogen analyzers. One monitors the process stream entering the hydrogen recombiner, and one monitors the discharge stream.

The controls and alarms incorporated to maintain the gas composition outside the range of flammable and explosive mixtures are described below.

A. The maximum concentration of hydrogen that the hydrogen recombiner can process in a single pass is 6-percent hydrogen by volume.

B. If the hydrogen recombiner feed concentration exceeds 6-percent hydrogen by volume, a high-hydrogen alarm sounds to warn that all hydrogen entering the hydrogen recombiner is not reacted. This alarm will be followed by a second alarm indicating high hydrogen in the hydrogen recombiner discharge. These alarms warn of a possible hydrogen accumulation in the GWPS.

C. If the hydrogen concentration in the hydrogen recombiner feed reaches 9 percent by volume, a high-high hydrogen alarm sounds; the oxygen feed is terminated; and the VCT hydrogen purge flow is terminated. These controls limit the possible accumulation of hydrogen in the system to 3 percent by volume.

D. If the oxygen concentration in the hydrogen recombiner feed reaches 3 percent by volume, an alarm sounds and oxygen feed flow is limited so that no further increase in flow is possible. This control maintains the system oxygen concentration at 3 percent or less, which is below the flammable limit for hydrogen-oxygen mixtures.

E. If hydrogen in the hydrogen recombiner discharge exceeds 1.5 percent by volume, an alarm sounds. This alarm warns of high hydrogen feed, possible reactor malfunction, or loss of oxygen feed.

F. If oxygen in the hydrogen recombiner discharge exceeds 15 ppm, a high alarm sounds. If it exceeds 60 ppm, a high-high alarm sounds and oxygen feed is terminated. This control prevents any accumulation of oxygen in the system in case of reactor malfunction.

G. On low flow through the hydrogen recombiner, oxygen feed is terminated. This control prevents an accumulation of oxygen following system malfunction.

H. On high discharge temperature from the cooler-condenser (downstream from the catalytic reactor) oxygen feed is terminated. This protects against loss of cooling water flow in the cooler-condenser.

I. On high temperature indication by any one of six thermocouples in the catalyst bed, the oxygen feed is terminated, and the VCT hydrogen purge flow is terminated.

J. On high temperature indication at the recombiner catalytic reactor discharge, oxygen feed to the recombiner is terminated.

11.3.2.2.3 Volume Control Tank Purge Flow Control

Each VCT purge line into the GWPS is equipped with a trip valve, a pressure-reducing valve and a hand-operated flow control valve. This arrangement will maintain a constant purge flow as selected by the operator.

Pressure fluctuations caused by changes in the VCT water level are absorbed by the pressure-regulating valve, which maintains a constant downstream pressure. This provides a constant head loss across the hand-operated control valve. Design flow is 0.7 sf^3/min hydrogen, with a range of 0.3 sf^3/min to 1.2 sf^3/min. When mixed with the 40 sf^3/min nitrogen stream, the maximum hydrogen content (both units operating) in the recombiner feed is 6.0 volume percent.

The purge line trip valve closes as a result of:

A. A low-pressure signal from the VCT pressure instrumentation. This prevents depressurization of the VCT if the hydrogen supply is lost.

B. A low-pressure signal from the gas compressor suction line. This signal also stops the gas compressor.

C. Closure of the oxygen supply valve to the hydrogen recombiner. The purge line trip valve is closed only if the oxygen supply valves for both the normally operating recombiner and the standby recombiner are closed.

11.3.2.3 System Operation

11.3.2.3.1 Startup Operation

The GWPS is initially purged with nitrogen to remove all air. During startup operation, one waste gas compressor, one hydrogen recombiner, and one shutdown GDT are in service in the process loop serving that plant unit. The reactor is at cold shutdown and the VCT contains nitrogen in the gas space. Reactor coolant contains neither hydrogen nor fission gases, but it may be saturated with air. While one unit is being started up, the VCT purge for the other unit is unaffected since each unit is served by a separate process loop of the GWPS.

When the reactor startup procedure requires that a hydrogen blanket be established in the VCT gas space, fresh hydrogen is charged into the VCT. The hydrogen-nitrogen mixture vented from the VCT enters the GWPS circulating nitrogen stream at the waste gas compressor suction. Nitrogen added to the GWPS accumulates in the shutdown GDT, causing the tank's pressure to rise.

Initially, the VCT vent gas will be very lean in hydrogen, and almost all the gas entering the GWPS will accumulate in the shutdown GDT. As the operation continues, however, the vent gas hydrogen content will gradually increase until it is almost totally hydrogen at the point when all of the nitrogen has been removed from the reactor coolant. At that time, hydrogen gas is passing through the VCT and mixing with the circulating nitrogen stream to give a mixture of hydrogen in nitrogen at the hydrogen recombiner inlet. A sufficient amount of oxygen is added in the hydrogen recombiner to react with the hydrogen to yield a discharge stream with a low residual concentration of hydrogen in nitrogen. After the water vapor is condensed and removed, the gas flow is directed to a shutdown GDT and from there to the waste gas compressor and back to the hydrogen recombiner.

When the reactor coolant hydrogen concentration is within operating specifications, the shutdown GDT is isolated, and flow is routed to one of the normal operation GDTs provided for normal power service. Gas accumulated in the shutdown GDT will be retained for use during operations to strip hydrogen from the reactor coolant when a plant unit is shut down.

11.3.2.3.2 Normal Operations

During normal power operation, nitrogen gas with entrained fission gases is continuously circulated around each GWPS process loop by one of the two waste gas compressors in the loop. Fresh hydrogen gas is charged to the VCT, where it is mixed with fission gases which have been stripped from the reactor coolant into the VCT gas space. The contaminated hydrogen gas is continuously vented from the VCT into the circulating nitrogen stream to transport the fission gases into the GWPS. The resulting mixture of nitrogen-hydrogen fission gas is pumped by the waste gas compressor to the hydrogen recombiner, where enough oxygen is added to reduce the hydrogen to a low residual concentration by oxidation to water vapor on a catalytic surface. After the water vapor is removed, the resulting gas stream is circulated to a normal operation GDT and back to the waste gas compressor suction to complete the circuit.

Each normal operation GDT is capable of being isolated, and only one tank is valved into operation in each process loop at any time. This minimizes the amount of radioactive gases that could be released as a consequence of any single failure, such as the rupture of any single tank or connected piping. A normal operation GDT is valved into the GWPS recirculation loop until the pressure or curie content of the inservice GDT reaches a value determined by approved plant procedure(s), after which it is isolated and another tank is placed in service. This process is illustrated in figure 11.3.2-3 for GDT pressure ≤ 20 psig and as illustrated in figure 11.3.2-4 when GDT pressure will exceed 20 psig. By alternating the use of these tanks, the accumulated activity from one unit is distributed among all seven normal operation GDTs in the process loop.

With continued plant operation, pressure in the normal operation will gradually increase as nonremovable gases accumulate in the system. The initial system equipment lineup, as described above, will be from the waste gas compressor to the hydrogen recombiner and then to the normal operation GDT. As the normal operation GDT pressure builds up, compensation must be made by periodic adjustment of the hydrogen recombiner backpressure control valve. When the normal operation GDT pressure reaches approximately 20 psig, the backpressure control valve will be fully open, so that no more adjustment can be made. At this time, the appropriate bypass lines are opened to line up the equipment for flow from the waste gas compressor to the normal operation GDTs and then to the hydrogen recombiner, as shown in figure 11.3.2-4. The hydrogen recombiner backpressure control valve is reset as required for the new arrangement, and the normal operation GDT pressure indicators are switched to read high range. This arrangement is suitable for operation up to 100 psig. Note that this high-pressure mode of operation will also normally be utilized during shutdown/startup operations.

11.3.2.3.3 Shutdown and Degassing of the Reactor Coolant System

Plant shutdown operations are essentially startup operations in reverse sequence. The VCT hydrogen blanket is maintained until after the reactor is shut down and reactor coolant fission gas concentrations have been reduced to desired level. During this operation hydrogen purge flow may be increased to speed up reactor coolant degassing. At this time, the VCT hydrogen purge is stopped for the plant unit being shut down. The normal operation GDT in service for that unit is then valved out and a nitrogen purge from a shutdown GDT to the VCT is begun for the unit being shut down. This shutdown GDT is placed in the GWPS process loop at the waste gas compressor discharge so that the gas mixture from the VCT vents to the waste gas compressor suction, passes through the shutdown GDT and to the hydrogen recombiner, where hydrogen is removed, and remaining gases are returned to the waste gas compressor suction. The nitrogen purge continues until reactor coolant hydrogen concentration reaches the required level or chemical degassing may be used by the operator to reduce RCS hydrogen concentrations during cold shutdown operations. Degassing is then complete, and the RCS may be opened for maintenance or refueling. At this point the shutdown GDT may be isolated.

During the first plant cold shutdown, fresh nitrogen is charged to the VCT to strip hydrogen from the reactor coolant. The resulting accumulation of nitrogen in the shutdown GDT is accommodated by allowing the tank pressure to increase. During subsequent shutdowns, however, there is no additional accumulation since the gas from the first shutdown will be reused.

11.3.2.3.4 System Drains

During operation, water may accumulate in the waste gas compressor moisture separator, in the hydrogen recombiner phase separator, and in the GDTs. Normally, the waste gas compressor and hydrogen recombiner drains will discharge to the VCT under motive head provided by internal component pressure. During maintenance, the drains are directed either to the recycle holdup tanks (if the drains contain dissolved fission gases) or to the waste holdup tank through the drain header (if the drains contain no fission gases). A gas trap is provided in the drain lines from the waste gas compressors and hydrogen recombiners to prevent undissolved gases from leaving the GWPS.

All drains from the waste GDTs are manually operated. Depending on the internal tank pressure on the drain routing, the waste GDT drain pump is used or bypassed. As necessary, the tank is drained by pumping the accumulated condensate to the VCT. However, during tank maintenance, waste GDT drains are routed to the recycle holdup tank because the volume of water in a waste GDT may be large enough to cause a noticeable dilution of reactor coolant boron concentration. Gas traps are provided in the waste GDT drain line to prevent the discharge of gases into other parts of the plant. All drains from the GWPS are filtered before entering either the VCT or the recycle holdup tank.

11.3.2.3.5 Atmospheric Releases

Although the GWPS is designed to accommodate continuous operation without atmospheric releases, the system design permits controlled discharge of gas from the system. Before a tank is emptied to the atmosphere via the plant vent, a gas sample must be analyzed to determine and record the activity to be removed. While the contents of a tank are being released to the atmosphere, a trip valve in the discharge line will close automatically if a high activity level is detected in the plant vent effluent.

11.3.3 RADIOACTIVE RELEASES

11.3.3.1 Discharge Requirements

Radioactive gaseous effluents and particulates discharged from the plant may not exceed the limits specified in the Offsite Dose Calculation Manual.

11.3.3.2 Westinghouse Pressurized Water Reactor (PWR) Experience Releases

Surveys have been performed of gaseous discharges from several operating Westinghouse PWR plants. The results are presented in table 11.3.3-1.

11.3.3.3 Estimated Releases

Radioactive effluent releases from the plant for normal operation are given in table 11.3.3-2. These release rates were calculated using the PWR-GALE code (reference 1) and plant operating parameters referenced in paragraph 11.1.1.2 and table 11.1-8. The releases are calculated for one unit; to obtain the combined releases for the two units, double the values listed in table 11.3.3-2.

Potential effluent releases from the radwaste solidification building vent are provided in table 11.3.3-2. These releases were conservatively estimated using the design basis reactor coolant activities in table 11.1-2 adjusted to 0.12-percent failed fuel as the source term. These sources were used in conjunction with the waste stream flow and processing methodology outlined in the GALE code to determine the activities of the wastes being input to the volume reduction system. These releases were determined using the decontamination factors for the system outlined in section 11.4.

11.3.3.4 Release Points for Determination of Dilution Factors

Gaseous and particulate radioactive effluents may be normally released through the following three points: the plant vent, turbine building vent, and radwaste processing facility vent. Subsection 2.3.5 outlines the methodology and information used to determine the long-term atmospheric dilution (χ/Q) and deposition (D/Q) factors for these release points. Effluent sources and associated vents are listed in table 2.3.5-2. Vent design information and input assumptions utilized for the long-term diffusion estimates are given in tables 2.3.5-3 and 2.3.5-4.

11.3.3.5 Effluent Concentration and Dilution Factors

A comparison of maximum offsite (at site boundary) airborne gaseous and particulate effluent concentrations with 10 CFR 20.1 - 20.601 limits is given in table 11.3.3-3. These concentrations are based on the sum of concentrations from the plant vent and turbine building vent using the site boundary dilution factors for the wake split model (plant vent) and the ground release model (turbine building vent) as given in subsection 2.3.5 and tables 2.3.5-10 and 2.3.5-11. For this calculation, the GALE code isotopic release rates in table 11.3.3-2 were grouped as follows:

A. The plant vent releases are composed of contributions from gas stripping, reactor building, auxiliary building, and waste gas system.

B. The turbine building vent releases are composed of contributions from turbine building and air ejector exhaust.

11.3.3.6 Estimated Doses from Atmospheric Releases

Estimated annual average doses from radionuclides released to the atmosphere from the plant during normal operation are given in table 11.3.3-4. These doses were calculated using the GASPAR code (reference 2). Various exposure pathways and nearest potential receptors within 5.0 miles (~8000 meters) of the plant were considered. As in paragraph 11.3.3.5, the isotopic release rates in table 11.3.3-2 were grouped according to discharge vent and associated atmospheric diffusion model. Thus, table 11.3.3-4 represents the total dose based on the releases from the plant vent and turbine building vent using the diffusion data referenced in paragraph 11.3.3.4. Note that all of these calculated doses are well within the guidelines of Appendix I to 10 CFR 50, and the Annex to Appendix I, Docket RM-50-2.

11.3.3.7 References

1. U.S. Nuclear Regulatory Commission, "Calculation of Releases from Pressurized Water Reactors," PWR-GALE Computer Code, NUREG-0017, April 1976.

2. U.S. Nuclear Regulatory Commission, "Calculation of Radiation Exposure to Man from Routine Air Releases of Nuclear Reactor Effluents," GASPAR Computer Code, NUREG-0597, June 1980 (based on the methodology outlined in Regulatory Guide 1.109, Revision 1, for atmospheric releases).

TABLE 11.3.2-1

DESIGN BASIS ACCUMULATED RADIOACTIVITY IN THE GWPS
AFTER 40 YEARS' OPERATION OF TWO UNITS[a]

Isotope	Activity (Ci) Following Plant Shutdown[b]		
	Zero Decay	30 Days	50 Days
Kr-85	78,000	77,600	77,300
All other noble gases			
Kr-83m	9	0	0
Kr-85m	116	0	0
Kr-87	10.8	0	0
Kr-88	110	0	0
Xe-131m	1240	230	69
Xe-133	132,000	2550	180
Xe-133m	6600	0	0
Xe-135	960	0	0
Xe-135m	3.8	0	0
Xe-138	0.26	0	0

The table is based on 40 years continuous operation[a] with 1 percent of rated core power generated by fuel rods containing cladding defects; therefore, the values are conservatively high. Power is assumed to be 3565 MWt. The data are based on a VCT purge rate of 0.7 sf^3/min for each of 2 units, the stripping fractions listed in table 11.1-1, and the reactor coolant concentrations given in table 11.1-2.

[a] The renewed operating licenses authorized a 20-year period of extended operation for both VEGP units, resulting in a total plant operating life of 60 years. Since the inventory in the Waste Gas Decay Tanks (WGDTs) has been routinely released during the first 20 years of operation and is expected to continue to be routinely released during future operation, the inventory of the WGDTs accumulated during the first 20 years of operation will be released prior to entering the period of extended operation. Therefore, the stated design capacity of the GWPS remains sufficient, and the analysis of the maximum fission product inventory in the GWPS over a 40-year plant life remains bounding for a 60-year plant life.
[b] A review of the table values for the effects of MUR power uprate determined those values need not be updated.

TABLE 11.3.2-2

EXPECTED ACCUMULATED RADIOACTIVITY IN THE GWPS
AFTER 40 YEARS' OPERATION OF TWO UNITS[a]

	Activity (Ci) Following Plant Shutdown[b]		
Isotope	Zero Decay	30 Days	50 Days
Kr-85	9400	9350	9320
All other noble gases			
Kr-83m	0.32	0	0
Kr-85m	5.6	0	0
Kr-87	0.52	0	0
Kr-88	5.6	0	0
Xe-131m	70	12	4
Xe-133	8600	170	12
Xe-135	74	0	0
Xe-135m	0.006	0	0
Xe-138	0.018	0	0

Inventories are based on reactor coolant concentrations given in table 11.1-7. The table is based on 40 years continuous operation[a]. Power is assumed to be 3565 MWt. The data are based on a VCT purge rate of 0.7 sf^3/min for each of 2 units and on the stripping fractions listed in table 11.1-7.

[a] The renewed operating licenses authorized a 20-year period of extended operation for both VEGP units, resulting in a total plant operating life of 60 years. Since the inventory in the Waste Gas Decay Tanks (WGDTs) has been routinely released during the first 20 years of operation and is expected to continue to be routinely released during future operation, the inventory of the WGDTs accumulated during the first 20 years of operation will be released prior to entering the period of extended operation. Therefore, the stated design capacity of the GWPS remains sufficient, and the analysis of the maximum fission product inventory in the GWPS over a 40-year plant life remains bounding for a 60-year plant life.

[b] A review of the table values for the effects of MUR power uprate determined those values need not be updated.

TABLE 11.3.2-3

REDUCTION IN RCS GASEOUS FISSION PRODUCTS
RESULTING FROM NORMAL OPERATION OF THE GWPS[a] [c]

	Reactor Coolant Gaseous Fission Product Activities (μCi/g)	
Isotope	GWPS Operating[b]	GWPS Not Operating
Kr-83m	0.45	0.46
Kr-85	0.059	7.3
Kr-85m	1.9	2.0
Kr-87	1.3	1.3
Kr-88	3.6	3.6
Xe-131m	0.25	2.2
Xe-133	62	270
Xe-133m	7.4	17
Xe-135	6.1	7.2
Xe-135m	0.48	0.48
Xe-138	0.64	0.64

a. Based on operating with cladding defects in fuel generating 1 percent of the rated core thermal power (3565 MWt) and a purification letdown rate of 75 gal/min.

b. The VCT purge rate is 0.7 sf^3/min.

c. A review of the table values for the effects of MUR power uprate determined those values need not be updated.

TABLE 11.3.2-4 (SHEET 1 OF 2)
PROCESS PARAMETERS FOR GWPS, INDEFINITE HOLDUP, 1-MONTH ACCUMULATION [a][b][j]
(FIRST HALF)

Item Description Gas Streams	Temp (°F)	Press (psig)	Flow (sf^3/min)	N_2 + He (%)	H_2 (%)	Isotopic Concentration (µCi/cm³) Kr-83m	Kr-85	Kr-85m	Kr-87
1. VCT purge[e]	130	18	0.7	0	100	4.36(-2)	2.56(-2)	3.10(-1)	9.89(-2)
2. GDT discharge	Amb[f]	5	40	99.9	0.1	6.74(-3)	1.69(1-)	1.15(-1)	1.10(-2)
3. Compressor suction	Amb	0.5	40.7	98.3	1.7	7.38(-3)	1.66(1-)	1.18(-1)	1.25(-2)
4. Compressor discharge	140	30	40.7	98.3	1.7	7.38(-3)	1.66(1-)	1.18(-1)	1.25(-2)
5. Recombiner discharge	140	5	40	99.9	0.1	7.50(-3)	1.69(-1)	1.20(-1)	1.27(-2)
6. Misc. vents: evap. RCDT recycle holdup tank eductor	140	0.5	Neg[g]	0	100	0	0	0	0
7. Recombiner oxygen supply	Amb	50	0.35	0	0	0	0	0	0

Item Description Liquid Streams	Temp (°F)	Press (psig)	Flow (gal/d)	Isotopic Concentration (mCi/cm³)[d][h] Kr-83m	Kr-85	Kr-85m	Kr-87
1. Waste gas compressor drain	140	40	0	1.80(-3)	4.06(-2)	2.89(-2)	3.04(-3)
2. Recombiner drain	140	30	12	1.50(-3)	3.38(-2)	2.40(-2)	2.53(-3)
3. GDT drain	Amb	5	36	5.98(-4)	1.50(-2)	1.02(-2)	9.71(-4)
4. System drains to VCT	140	5-40	48	7.26(-4)	1.77(-2)	1.22(-2)	1.19(-3)

Item Component	Temp (°F)	Press (psig)	Vol (ft^3)	N_2 + He (%)	H_2 (%)	Component Inventory (Ci/)[i] Kr-83m	Kr-85	Kr-85m	Kr-87
1. Compressor	140	40	4	98.3	1.7	2.84(-3)	7.12(-2)	4.85(-2)	4.62(-3)
2. Recombiners	140	30	4	99.3	0.1	2.32(-3)	5.82(-2)	3.96(-2)	3.77(-3)
3. GDT	Amb	5	600	99.9	0.1	1.55(-1)	3.88	2.64	2.52(-1)
Total System						1.6(-1)	27.3	2.8	2.6(-1)

TABLE 11.3.2-4 (SHEET 1 OF 2)
PROCESS PARAMETERS FOR GWPS, INDEFINITE HOLDUP, 1-MONTH ACCUMULATION[(a)(b)(j)]
(SECOND HALF)

Kr-88	Kr-89	Xe-131m	Xe-133	Xe-133m	Xe-135	Xe-135m	Xe-137	Xe-138
4.94(-1)	5.39(-4)	6.29(-2)	16.6	3.46(-1)	9.0(-1)	5.83(-3)	1.17(-3)	1.93(-2)
1.18(1-)	2.40(-6)	2.50(-1)	37.1	4.70(-1)	5.99(-1)	1.31(-4)	6.32(-6)	3.83(-4)
1.24(-1)	1.16(-5)	2.46(-1)	36.7	4.68(-1)	6.04(-1)	2.29(-4)	2.63(-5)	7.09(-4)
1.24(-1)	1.16(-5)	2.46(-1)	36.7	4.68(-1)	6.04(-1)	2.29(-4)	2.63(-5)	7.09(-4)
1.26(-1)	1.18(-5)	2.50(-1)	37.4	4.76(-1)	6.15(-1)	2.33(-4)	2.68(-5)	7.22(-4)
0	0	0	0	0	0	0	0	0
0	0	0	0	0	0	0	0	0

Kr-88	Kr-89	Xe-131m	Xe-133	Xe-133m	Xe-135	Xe-135m	Xe-137	Xe-138
3.03(-2)	2.84(-6)	4.96(-2)	7.40	9.44(-2)	1.22(-1)	4.62(-5)	5.31(-6)	1.43(-4)
2.52(-2)	2.36(-6)	4.12(-2)	6.16	7.85(-2)	1.01(-1)	3.84(-5)	4.41(-6)	1.19(-4)
1.04(-2)	2.13(-7)	1.82(-2)	2.71	3.44(-2)	4.39(-2)	9.57(-6)	4.63(-7)	2.81(-5)
1.25(-2)	5.20(-7)	2.15(-2)	3.21	4.07(-2)	5.21(-2)	1.37(-5)	1.03(-6)	4.11(-5)

Kr-88	Kr-89	Xe-131m	Xe-133	Xe-133m	Xe-135	Xe-135m	Xe-137	Xe-138
4.96(-2)	1.01(-6)	1.05(-1)	15.6	1.98(-1)	2.53(-1)	5.51(-5)	2.66(-6)	1.62(-4)
4.05(-2)	8.27(-7)	8.57(-2)	12.8	1.62(-1)	2.06(-1)	4.50(-5)	2.18(-6)	1.32(-4)
2.70	5.52(-5)	5.72	851	10.8	13.8	3.0(-3)	1.45(-4)	8.81(-3)
2.8	5.7(-5)	35	4300	37	17	3.1(-3)	1.5(-4)	9.1(-3)

TABLE 11.3.2-4 (SHEET 2 OF 2)

NOTES:

a. Basis: Type of operation = indefinite holdup

Power level = 3656 MWt

Number of units = 1

Normal operation GDTs in rotational use = 7

GDT operating interval = 1 day

Accumulation period = 1 month

b. Concentrations based on stripping fractions and reactor coolant activities from table 11.1-2.

c. Concentrations in $\mu Ci/cm^3$ of gas at atmospheric pressure and 140°F.

d. Isotopic concentrations are maximum values for all isotopes except krypton-85. Krypton-85 is at its maximum concentration after 14 ½ years of operation.

e. Parameters reflect the gas streams from one of the two operating reactors.

f. Amb = ambient.

g. Neg = negligible.

h. Concentrations in $\mu Ci/cm^3$ at room temperature.

i. Total system inventories are maximum values except for krypton-85. The maximum inventory of krypton-85 occurs at the end of plant life[1]. For all other isotopes, the maximum inventory is reached soon after initiating plant operation and is maintained until the end of plant life.

j. A review of the table values for the effects of MUR power uprate determined those values need not be updated.

[1] The renewed operating licenses authorized a 20-year period of extended operation for both VEGP units, resulting in a total plant operating life of 60 years. Since the inventory in the Waste Gas Decay Tanks (WGDTs) has been routinely released during the first 20 years of operation and is expected to continue to be routinely released during future operation, the inventory of the WGDTs accumulated during the first 20 years of operation will be released prior to entering the period of extended operation. Therefore, the stated design capacity of the GWPS remains sufficient, and the analysis of the maximum fission product inventory in the GWPS over a 40-year plant life remains bounding for a 60-year plant life.

TABLE 11.3.2-5 (SHEET 1 OF 2)
PROCESS PARAMETERS FOR GWPS, INDEFINITE HOLDUP, 90-DAY ACCUMULATION[(a)(b)(g)]
(FIRST HALF)

Item Description Gas Streams	Temp (°F)	Press (psig)	Flow (sf^3/min)	N_2 + He (%)	H_2 (%)	Isotopic Concentration (mCi/cm^3)[(c)] Kr-83m	Kr-85	Kr-85m	Kr-87
1. VCT purge[(d)]	130	18	0.7	0	100	4.36(-2)	2.56(-2)	3.10(-2)	9.89(-2)
2. GDT discharge	Amb	20	40	99.9	0.1	1.37(-3)	5.79	2.34(-2)	2.23(-3)
3. Compressor suction	Amb	0.5	40.7	98.3	1.7	2.10(-3)	5.69	2.36(-2)	3.89(-3)
4. Compressor discharge	140	30	40.7	98.3	1.7	2.10(-3)	5.69	2.36(-2)	3.89(-3)
5. Recombiner discharge	140	20	40	99.9	0.1	2.14(-3)	5.79	2.40(-2)	3.96(-3)
6. Misc. vents: evap. RCDT recycle holdup tank eductor[(d)]	140	0.5	Neg	0	100	0	0	0	0
7. Recombiner oxygen supply	Amb	50	0.35	0	0	0	0	0	0

Item Description Liquid Streams	Temp (°F)	Press (psig)	Flow (gal/d)	Isotopic Concentration (mCi/cm^3)[(e)] Kr-83m	Kr-85	Kr-85m	Kr-87
1. Waste gas compressor drain	140	40	0	5.12(-4)	1.39	5.75(-4)	9.50(-4)
2. Recombiner drain	140	30	12	4.26(-4)	1.16	4.78(-3)	7.90(-4)
3. GDT drain	Amb	20	36	6.14(-4)	2.59	1.05(-2)	9.97(-4)
4. System drains to VCT	140	30-45	48	5.87(-4)	2.38	9.66(-3)	9.68(-4)

Item Component	Temp (°F)	Press (psig)	Vol (ft^3)	N_2 + He (%)	H_2 (%)	Component Inventory (Ci/) Kr-83m	Kr-85	Kr-85m	Kr-87
1. Compressor	140	40	4	98.3	1.7	5.78(-4)	2.44	9.88(-3)	9.40(-4)
2. Recombiners	140	30	4	99.9	0.1	4.73(-4)	1.99	8.07(-3)	7.68(-4)
3. GDT	Amb	20	600	99.9	0.1	1.59(-1)	671	2.71	2.58(-1)
Total System						1.6(-1)	4700	2.8	2.6(-1)

TABLE 11.3.2-5 (SHEET 1 OF 2)
PROCESS PARAMETERS FOR GWPS, INDEFINITE HOLDUP, 90-DAY ACCUMULATION[(a)(b)(g)]
(SECOND HALF)

Kr-88	Kr-89	Xe-131m	Xe-133	Xe-133m	Xe-135	Xe-135m	Xe-137	Xe-138
4.94(-1)	5.39(-4)	6.29(-2)	16.6	3.46(-1)	8.99(-1)	5.84(-3)	1.17(-3)	1.93(-2)
2.40(2-)	4.89(-7)	5.07(-2)	7.5	9.57(-2)	1.22(-1)	2.66(-5)	1.29(-6)	7.81(-5)
3.20(-2)	9.76(-6)	5.09(-2)	7.7	1.0(-1)	1.35(-1)	1.27(-4)	2.14(-5)	4.09(-4)
3.20(-2)	9.76(-6)	5.09(-2)	7.7	1.0(-1)	1.35(-1)	1.27(-4)	2.14(-5)	4.09(-4)
3.26(-2)	9.93(-6)	5.18(-2)	7.8	1.02(-1)	1.38(-1)	1.29(-4)	2.17(-5)	4.16(-4)
0	0	0	0	0	0	0	0	0
0	0	0	0	0	0	0	0	0

Kr-88	Kr-89	Xe-131m	Xe-133	Xe-133m	Xe-135	Xe-135m	Xe-137	Xe-138
7.82(-3)	2.38(-6)	1.03(-2)	1.55	2.02(-2)	2.73(-2)	2.55(-5)	4.31(-6)	8.25(-5)
6.50(-3)	1.98(-6)	8.53(-3)	1.29	1.68(-2)	2.27(-2)	2.12(-5)	3.58(-6)	6.86(-5)
1.07(-2)	2.19(-7)	1.87(-2)	2.79	3.53(-2)	4.51(-2)	9.82(-6)	4.75(-7)	2.88(-5)
1.01(-2)	4.70(-7)	1.73(-2)	2.57	3.27(-2)	4.19(-2)	1.15(-5)	9.19(-7)	3.45(-5)

Kr-88	Kr-89	Xe-131m	Xe-133	Xe-133m	Xe-135	Xe-135m	Xe-137	Xe-138
1.01(-2)	2.06(-7)	2.14(-2)	3.18	4.03(-2)	5.14(-2)	1.12(-5)	5.42(-7)	3.29(-5)
8.25(-3)	1.68(-7)	1.75(-2)	2.60	3.30(-2)	4.20(-2)	9.16(-6)	4.43(-7)	2.69(-5)
2.77	5.66(-5)	5.87	874	11.1	14.1	3.08(-3)	1.49(-4)	9.04(-3)
2.8	5.7(-5)	35	4300	37	17	3.1(-3)	1.5(-4)	9.1(-3)

TABLE 11.3.2-5 (SHEET 2 OF 2)

NOTES:

a. Basis: Type of operation = periodic release of gases

Power level = 3565 MWt

Number of units = 1

Normal operation GDTs in rotational use = 7

GDT operating interval = 1 day

Accumulation period = 40 years[1]

b. Concentrations based on stripping fractions and reactor coolant activities from table 11.1-2.

c. Concentrations in $\mu Ci/cm^3$ of gas at atmospheric pressure and 140°F.

d. Parameters reflect the gas streams from one of the two operating reactors.

e. Concentrations in $\mu Ci/cm^3$ at room temperature and atmospheric pressure.

f. Total system inventories are maximum values.

g. A review of the table values for the effects of MUR power uprate determined those values need not be updated.

[1] The renewed operating licenses authorized a 20-year period of extended operation for both VEGP units, resulting in a total plant operating life of 60 years. Since the inventory in the Waste Gas Decay Tanks (WGDTs) has been routinely released during the first 20 years of operation and is expected to continue to be routinely released during future operation, the inventory of the WGDTs accumulated during the first 20 years of operation will be released prior to entering the period of extended operation. Therefore, the stated design capacity of the GWPS remains sufficient, and the analysis of the maximum fission product inventory in the GWPS over a 40-year plant life remains bounding for a 60-year plant life.

TABLE 11.3.2-6 (SHEET 1 OF 2)
PROCESS PARAMETERS FOR GWPS, INDEFINITE HOLDUP, 90-DAY HOLDUP AND RELEASE[(a)(b)(f)]
(FIRST HALF)

Item Description Gas Streams	Temp (°F)	Press (psig)	Flow (sf^3/min)	N_2 + He (%)	H_2 (%)	Isotopic Concentration (mCi/cm^3)[(c)] Kr-83m	Kr-85	Kr-85m	Kr-87
1. VCT purge[(d)]	130	18	0.7	0	100	4.36(-02)	2.56(-2)	3.10(-1)	9.89(-2)
2. GDT discharge	Amb	100	40	99.9	0.1	6.43(-3)	1.11	1.10(-1)	1.04(-2)
3. Compressor suction	Amb	0.5	40.7	98.3	1.7	7.07(-3)	1.09	1.08(-1)	1.20(-2)
4. Compressor discharge	140	100	40.7	98.3	1.7	7.07(-3)	1.09	1.08(-1)	1.20(-2)
5. Recombiner discharge	140	20	40	99.9	0.1	7.19(-3)	1.11	1.10(-1)	1.22(-2)
6. Misc. vents: evap. RCDT recycle holdup tank eductor[(d)]	140	0.5	Neg	0	100	0	0	0	0
7. Recombiner oxygen supply	Amb	50	0.35	0	0	0	0	0	0

Item Description Liquid Streams	Temp (°F)	Press (psig)	Flow (gal/d)	Isotopic Concentration (mCi/cm^3)[(e)] Kr-83m	Kr-85	Kr-85m	Kr-87
1. Waste gas compressor drain	140	40	0	1.72(-3)	2.67(-1)	2.65(-2)	2.92(-3)
2. Recombiner drain	140	30	12	1.43(-3)	2.22(-1)	2.20(-2)	2.43(-3)
3. GDT drain	Amb	100	36	5.99(-4)	1.04(-1)	1.02(-2)	9.73(-4)
4. System drains to VCT	140	30-100	48	7.18(-4)	1.21(-1)	1.19(-2)	1.18(-3)

Item Component	Temp (°F)	Press (psig)	Vol (ft^3)	N_2 + He (%)	H_2 (%)	Component Inventory (Ci/)[(f)] Kr-83m	Kr-85	Kr-85m	Kr-87
1. Compressor	140	40	4	98.3	1.7	2.71(-3)	4.69(-1)	4.63(-2)	4.40(-3)
2. Recombiners	140	30	4	99.9	0.1	2.21(-3)	3.84(-1)	3.78(-2)	3.60(-3)
3. GDT	Amb	100	600	99.9	0.1	1.55(-1)	26.9	2.65	2.52(-1)
Total System						1.6(-1)	81.4	2.8	2.6(-1)

TABLE 11.3.2-6 (SHEET 1 OF 2)
PROCESS PARAMETERS FOR GWPS, INDEFINITE HOLDUP, 90-DAY HOLDUP AND RELEASE[(a)(b)(f)]
(SECOND HALF)

Kr-88	Kr-89	Xe-131m	Xe-133	Xe-133m	Xe-135	Xe-135m	Xe-137	Xe-138
4.94(-1)	5.39(-4)	6.29(-2)	16.6	3.46(-1)	8.99(-1)	5.84(-3)	1.17(-3)	1.93(-2)
1.12(1-)	2.29(-6)	4.96(-1)	65.3	6.55(-1)	5.74(-1)	1.25(-4)	6.03(-6)	3.66(-4)
1.19(-1)	1.15(-5)	4.89(-1)	64.4	6.50(-1)	5.79(-1)	2.23(-4)	2.60(-5)	6.92(-4)
1.19(-1)	1.15(-5)	4.89(-1)	64.4	6.50(-1)	5.79(-1)	2.23(-4)	2.60(-5)	6.92(-4)
1.21(-1)	1.17(-5)	4.97(-1)	65.6	6.61(-1)	5.89(-1)	2.27(-4)	2.65(-5)	7.04(-4)
0	0	0	0	0	0	0	0	
0	0	0	0	0	0	0	0	0

Kr-88	Kr-89	Xe-131m	Xe-133	Xe-133m	Xe-135	Xe-135m	Xe-137	Xe-138
2.90(-2)	2.81(-6)	9.85(-2)	13	1.31(-1)	1.17(-1)	4.50(-5)	5.25(-6)	1.40(-4)
2.41(-2)	2.34(-6)	8.19(-2)	10.8	1.09(-1)	9.71(-2)	3.74(-5)	4.36(-6)	1.16(-4)
1.04(-2)	2.13(-7)	3.82(-2)	5	5.04(-2)	4.41(-2)	9.58(-6)	4.64(-7)	2.81(-5)
1.24(-2)	5.17(-7)	4.44(-2)	5.8	5.88(-2)	5.17(-2)	1.36(-5)	1.02(-6)	4.07(-5)

Kr-88	Kr-89	Xe-131m	Xe-133	Xe-133m	Xe-135	Xe-135m	Xe-137	Xe-138
4.73(-2)	9.65(-7)	2.09(-1)	27.5	2.76(-1)	2.42(-1)	5.25(-5)	2.54(-6)	1.54(-4)
3.86(-2)	7.89(-7)	1.71(-1)	22.5	2.26(-1)	1.98(-1)	4.29(-5)	2.08(-6)	1.26(-4)
2.71	5.52(-5)	12	1575	15.8	13.8	3.0(-3)	1.45(-4)	8.82(-3)
2.8	5.7(-5)	35	4300	37	17	3.1(-3)	1.5(-4)	9.1(-3)

TABLE 11.3.2-6 (SHEET 2 OF 2)

NOTES:

a. Basis: Type of operation = indefinite holdup

Power level = 3565 MWt

Number of units = 1

Normal operation GDTs in rotational use = 3

GDT operating interval = 1 day

Accumulation period = 90 days

b. Concentrations based on stripping fractions and reactor coolant activities from table 11.1-2.

c. Concentrations in $\mu Ci/cm^3$ of gas at atmospheric pressure and 140°F.

d. Parameters reflect the gas streams from one of the two operating reactors.

e. Concentrations in $\mu Ci/cm^3$ at room temperature and atmospheric pressure.

f. A review of the table values for the effects of MUR power uprate determined those values need not be updated.

TABLE 11.3.2-7 (SHEET 1 OF 2)

GWPS COMPONENT DESCRIPTION

<u>Waste Gas Compressor Packages</u>	
Number (two units)	4
Design pressure (psig)	150
Design temperature (°F)	180
Normal operating temperature (°F)	70-130
Normal operating pressure (psig)	
Suction	0.5-2.0
Discharge	25-100
Design flowrate (N_2 at 140°F, 110 psig discharge) (sf^3/min)	40
Material of construction	Carbon steel
<u>Waste GDTs</u>	
Number (7 per unit plus 2 shared)	16
Design pressure (psig)	150
Design temperature (°F)	180
Volume (each) (ft^3)	600
Normal operating pressure (psig)	5-100
Normal operating temperature (°F)	50-140
Material of construction	Carbon steel
<u>Catalytic Hydrogen Recombiner Packages</u>	
Number (2 units)	3
Design pressure (psig)	150
Design temperature (°F)	(a)
Inlet pressure (psig)	25-100
Inlet temperature (°F)	70-140
Design flowrate (sf^3/min)	50
Design hydrogen recombiner rate (sf^3/min)	3.0
Discharge pressure (psig)	0-20
Discharge temperature (°F)	70-140
Material of construction	Stainless steel

TABLE 11.3.2-7 (SHEET 2 OF 2)

GDT Drain Pump

Quantity (2 units)	2
Design temperature (°F)	180
Design pressure (psig)	150
Design flow (gal/min)	10
Design head (ft)	90
NPSH required (ft)	10
Maximum operating temperature (°F)	140
Maximum operating suction pressure (psig)	30[b]
Fluid pumped	Reactor makeup water
Material	Stainless steel

Waste Gas Drain Filter

Quantity (2 units)	2
Type	Disposable cartridge
Design temperature (°F)	200
Design pressure (psig)	150
Design flow (gal/min)	35
Retention of 25-μm particles (%)	98
Materials	Stainless steel

Gas Traps

Quantity (2 units)	2	10
Design temperature (°F)	180	180
Design pressure (psig)	150	150
Operating inlet pressure (psig)		
Maximum	40	100
Minimum	5	5
Design flow (gal/min)	25	5
Material	Carbon steel	Carbon steel

a. Varies by component but exceeds component operating temperature by 100°F.

b. Pump not required when waste GDT pressure exceeds 30 psig.

TABLE 11.3.3-1

GASEOUS RELEASES FROM WESTINGHOUSE-DESIGNED OPERATING REACTORS[a]

	Noble Gases (Ci)		
	1975	1976	1977
R. E. Ginna	1.04(4)	5.52(3)	3.20(3)
San Onofre 1	1.11(3)	4.16(2)	1.54(2)
Surry 1 and 2	8.04(3)	1.91(4)	1.90(4)
H. B. Robinson	1.17(3)	6.40(2)	4.76(2)
Point Beach 1 and 2	4.45(4)	1.91(3)	1.13(3)
Connecticut Yankee	4.80(2)	4.52(2)	3.12(3)

	Iodine (Ci) I-131 and Particulate with half-lives > 8 days		
	1975	1976	1977
R. E. Ginna	2.0(-2)	3.17(-2)	2.55(-2)
San Onofre 1	4.0(-2)	<1.0(-2)	1.86(-4)
Surry 1 and 2	5.0(-2)	3.46(-1)	1.20(-1)
H. B. Robinson	2.0(-2)	9.96(-2)	3.88(-3)
Point Beach 1 and 2	7.0(-2)	1.85(-2)	5.02(-3)
Connecticut Yankee	<1.00(-2)	<1.00(-2)	1.74(-3)

a. "Radioactive Materials Released from Nuclear Power Plants," NUREG-0521, Annual Report, 1977.

TABLE 11.3.3-2

EXPECTED ANNUAL AVERAGE RELEASE OF AIRBORNE RADIONUCLIDES[(a)(b)]

Gaseous Release Rate (Ci/year)[(c)]

Nuclide	Primary Coolant (μCi/g)	Secondary Coolant (μCi/g)	Gas Stripping: Shutdown	Gas Stripping: Continuous	Building Ventilation: Containment	Building Ventilation: Auxiliary	Building Ventilation: Turbine	Blowdown Vent Offgas	Air Ejector Exhaust	Total
Kr-83m	2.119E-02	5.387E-09	0.0	0.0	4.0E+00	0.0	0.0	0.0	0.0	4.0E+00
Kr-85m	1.049E-01	2.721E-08	0.0	0.0	3.2E+01	2.0E+00	0.0	0.0	1.0E+00	3.5E+01
Kr-85	8.231E-03	2.121E-09	4.0E+00	2.6E+02	6.0E+00	0.0	0.0	0.0	0.0	2.7E+02
Kr-87	6.139E-02	1.507E-08	0.0	0.0	8.0E+00	1.0E+00	0.0	0.0	0.0	9.0E+00
Kr-88	1.976E-01	5.001E-08	0.0	0.0	4.5E+01	4.0E+00	0.0	0.0	3.0E+00	5.2E+01
Kr-89	5.265E-03	1.357E-09	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Xe-131m	1.941E-02	5.034E-09	0.0	3.0E+00	1.4E+01	0.0	0.0	0.0	0.0	1.7E+01
Xe-133m	1.044E-01	2.708E-08	0.0	0.0	6.9E+01	2.0E+00	0.0	0.0	1.0E+00	7.2E+01
Xe-133	5.246E+00	1.341E-06	0.0	1.0E+00	3.7E+03	1.1E+02	0.0	0.0	7.0E+01	3.9E+03
Xe-135m	1.362E-02	3.471E-09	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Xe-135	3.026E-01	7.719E-08	0.0	0.0	1.3E+02	6.0E+00	0.0	0.0	4.0E+00	1.4E+02
Xe-137	9.475E-03	2.422E-09	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Xe-138	4.608E-02	1.157E-08	0.0	0.0	1.0E+00	1.0E+00	0.0	0.0	0.0	1.0E+00
Total Noble Gases										4.5E+03
I-131	2.845E-01	6.773E-06	0.0	0.0	2.0E-02	4.5E-03	3.7E-04	0.0	2.8E-02	5.3E-02
I-133	4.005E-01	8.392E-06	0.0	0.0	2.2E-02	6.4E-03	4.5E-04	0.0	4.0E-02	6.9E-02

Tritium gaseous release = 730 Ci/year

Airborne Particulate Release Rate (Ci/year)

Nuclide	Waste Gas System	Building Ventilation: Containment	Building Ventilation: Auxiliary	Total
Mn-54	4.5E-05	2.2E-04	1.8E-04	4.4E-04
Fe-59	1.5E-05	7.4E-05	6.0E-05	1.5E-04
Co-58	1.5E-04	7.4E-04	6.0E-04	1.5E-03
Co-60	7.0E-05	3.4E-04	2.7E-04	6.8E-04
Sr-89	3.3E-06	1.7E-05	1.3E-05	3.3E-05
Sr-90	6.0E-07	3.0E-06	2.4E-06	6.0E-06
Cs-134	4.5E-05	2.2E-04	1.8E-04	4.4E-04
Cs-137	7.5E-05	3.8E-04	3.0E-04	7.5E-04

a. For one unit.

b. Twenty-five Ci/year of argon-41 are released from the containment, and 8 Ci/year of carbon-14 are released from the waste gas processing system (from PWR-GALE code, section 11.1, reference 1). 710 Ci/year of tritium are released via vapor pathways.

c. The appearance of 0.0 in the table indicates release is less than 1.0 Ci/year for noble gas and 0.0001 Ci/year for I.

TABLE 11.3.3-3 (SHEET 1 OF 2)

COMPARISON OF CALCULATED MAXIMUM OFFSITE AIRBORNE CONCENTRATION WITH 10 CFR 20 ASSUMING EXPECTED FUEL LEAKAGE

Isotope	Total Annual Release from One Unit[a] (Ci/year)	Maximum Site Boundary Concentration[b] (µCi/ml)	Maximum Permissable Concentration (MPC)[c] (Ci/ml)	Fraction of MPC
Cr-51	2.66E-07	3.11E-19	4.0E-07	7.78E-13
Mn-54	4.40E-04	5.15E-16	1.0E-08	5.15E-08
Fe-55	3.62E-07	4.23E-19	3.0E-08	1.41E-11
Fe-59	1.50E-04	1.76E-16	5.0E-09	3.52E-08
Co-58	1.50E-03	1.76E-15	3.0E-08	5.87E-08
Co-60	6.80E-04	7.96E-16	1.0E-08	7.96E-08
Br-83	6.89E-10	8.06E-22	1.0E-10	8.06E-12
Br-84	1.98E-11	2.32E-23	3.0E-08	7.73E-16
Br-85m	2.02E-14	2.36E-26	3.0E-08	7.87E-19
Rb-86	1.02E-08	1.19E-20	1.0E-08	1.19E-12
Sr-89	3.41E-05	3.99E-17	3.0E-10	1.33E-07
Sr-90	6.00E-06	7.02E-18	3.0E-11	2.34E-07
Sr-91	1.26E-09	1.47E-21	2.0E-08	7.35E-14
Y-90	2.50E-11	2.93E-23	4.0E-09	7.33E-15
Y-91	5.22E-08	6.11E-20	1.0E-09	6.11E-11
Y-91m	1.23E-10	1.44E-22	8.0E-07	1.80E-16
Y-93	7.42E-11	8.68E-23	6.0E-09	1.45E-14
Zr-95	1.13E-08	1.32E-20	4.0E-09	3.30E-12
Nb-95	9.52E-09	1.11E-20	2.0E-08	5.55E-13
Mo-99	1.77E-06	2.07E-18	3.0E-08	6.90E-11
Tc-99m	7.61E-07	8.90E-19	1.0E-06	8.90E-13
Ru-103	7.34E-09	8.59E-21	2.0E-08	4.30E-13
Ru-106	2.14E-08	2.50E-20	3.0E-09	8.33E-12
Rh-103m	8.56E-12	1.00E-23	3.0E-06	3.33E-18
Rh-105m	1.53E-10	1.79E-22	1.0E-10	1.79E-12
Te-129	3.55E-10	4.15E-22	2.0E-07	2.08E-15
Te-129m	2.13E-07	2.49E-19	3.0E-09	8.30E-11
Te-131	3.22E-10	3.77E-22	3.0E-06	1.26E-16
Te-131m	2.26E-08	2.64E-20	1.0E-08	2.64E-12
Te-132	7.02E-07	8.21E-19	7.0E-09	1.17E-10
I-130	6.06E-07	7.09E-19	1.0E-10	7.09E-10
I-131	5.35E-02	6.26E-14	1.0E-10	6.26E-04
I-132	3.42E-06	4.00E-18	3.0E-09	1.33E-09
I-133	7.24E-02	8.47E-14	4.0E-10	2.12E-04
I-134	9.36E-08	1.10E-19	6.0E-09	1.83E-11
I-135	2.03E-05	2.38E-17	1.0E-09	2.38E-08
Cs-134	4.46E-04	5.22E-16	1.0E-09	5.22E-07
Cs-136	1.30E-06	1.52E-18	1.0E-08	1.52E-10
Cs-137	7.57E-08	8.86E-20	2.0E-09	4.43E-11
Ba-137m	1.84E-12	2.15E-24	3.0E-06	7.17E-19
Ba-140	1.94E-08	2.27E-20	4.0E-09	5.68E-12
La-140	5.26E-06	6.15E-18	5.0E-09	1.23E-09
Ce-141	1.06E-08	1.24E-20	2.0E-08	6.20E-13
Ce-143	4.14E-10	4.84E-22	9.0E-09	5.38E-14
Ce-144	7.22E-09	8.45E-21	3.0E-10	2.82E-11

TABLE 11.3.3-3 (SHEET 2 OF 2)

Isotope	Total Annual Release from One Unit[(a)] (Ci/year)	Maximum Site Boundary Concentration[(b)] (μCi/ml)	Maximum Permissable Concentration (MPC)[(c)] (Ci/ml)	Fraction of MPC
Np-239	2.15E-08	2.52E-20	3.0E-08	8.40E-13
Rb-88	5.42E-10	6.34E-22	3.0E-06	2.11E-16
Pu-239	7.61E-09	8.90E-21	6.0E-14	1.48E-07
Xe-131m	1.60E+01	1.87E-11	4.0E-07	4.68E-05
Xe-133m	6.70E+01	7.84E-11	3.0E-07	2.61E-04
Xe-133	3.58E+03	4.19E-09	3.0E-07	1.40E-02
Xe-135	1.41E+02	1.65E-10	1.0E-07	1.65E-03
Xe-138	2.00E+00	2.34E-12	3.0E-06	7.80E-07
Kr-83m	3.00E+00	3.51E-12	3.0E-06	1.17E-06
Kr-85m	3.40E+01	3.98E-11	1.0E-07	3.98E-04
Kr-85	2.60E+02	3.04E-10	3.0E-07	1.01E-03
Kr-87	8.00E+00	9.36E-12	2.0E-08	4.68E-04
Kr-88	5.10E+01	5.97E-11	2.0E-08	2.99E-03
Ar-41	2.50E+01	2.93E-11	4.0E-08	7.33E-04
C-14	8.00E+00	9.36E-12	1.0E-07	9.36E-05
H-3	9.25E+02	1.08E-09	2.0E-07	5.40E-03

a. Includes total Ci/year from table 11.3.3-2.

b. Based on the sum of contributions from the plant vent, radwaste solidification vent, and turbine building vent using the total maximum dilution factors (χ/Q) at the nearest site boundary in 16 compass directions:

χ/Q (plant vent) = 1.92E-06 s/m^3 (wake split model total)
χ/Q (turbine building) = 2.75E-05 s/m^3 (ground release model total)

c. From Appendix B, Table II, column 1 of 10 CFR 20.1 - 20.601.

TABLE 11.3.3-4 (SHEET 1 OF 6)

ESTIMATED ANNUAL DOSES TO AN INDIVIDUAL FROM GASEOUS AND PARTICULATE EFFLUENTS[(a)]

Gaseous Dose Rate[(b)]

Location	Pathway	Gamma Dose Rate in Air (mrad/year)	Beta Dose Rate in Air (mrad/year)	Total Body Dose Rate (mrem/year)	Skin Dose Rate (mrem/year)
Nearest site boundary (0.68 mile NE)	Plume	2.24E-02	3.94E-02	1.41E-02	3.34E-02
Nearest residence (1.2 mile WSW)	Plume	1.20E-02	2.21E-02	7.63E-03	1.82E-02
Nearest vegetable garden (1.4 mile WSW)	Plume	1.01E-02	1.83E-02	6.35E-03	1.53E-02
Nearest meat animal (3.10 mile SW)	Plume	5.35E-03	9.91E-03	3.36E-03	8.21E-03
Nearest milk cow and goat (4.60 mile SE)	Plume	2.31E-03	4.27E-03	1.46E-03	3.54E-03

Radioiodines and Particulates Dose Rate (mrem/year)[(c)]

Location	Pathway	Total Body	GI Tract	Bone	Liver	Kidney	Thyroid	Lung	Skin
Nearest site boundary (0.68 mile NE)	Ground deposition	2.42E-01	2.42E-01	2.42E-01	2.42E-01	2.42E-01	2.42E-01	2.42E-01	2.83E-01
	Inhalation								
	Adult	2.70E-02	2.51E-02	2.09E-03	2.78E-02	3.22E-02	5.78E-02	2.77E-02	2.78E-02
	Teen	2.65E-02	2.54E-02	2.93E-03	2.90E-02	2.68E-02	6.62E-02	2.81E-02	2.79E-02
	Child	2.30E-02	2.24E-02	3.96E-03	2.59E-02	2.38E-02	6.97E-02	2.51E-02	2.50E-02
	Infant	1.31E-02	1.29E-02	2.42E-03	1.56E-02	1.38E-02	5.64E-02	1.54E-02	1.55E-02
	Total Dose to Receptor[(d)]								
	Adult	2.69E-01	2.67E-01	2.44E-01	2.70E-01	2.74E-01	3.00E-01	2.70E-01	3.11E-01
	Teen	2.69E-01	2.67E-01	2.45E-01	2.71E-01	2.69E-01	3.08E-01	2.70E-01	3.11E-01
	Child	2.65E-01	2.64E-01	2.46E-01	2.68E-01	2.66E-01	3.12E-01	2.67E-01	3.08E-01
	Infant	2.55E-01	2.55E-01	2.44E-01	2.58E-01	2.56E-01	2.98E-01	2.57E-01	2.99E-01

TABLE 11.3.3-4 (SHEET 2 OF 6)

Dose Rate (mrem/year)

Location	Pathway	Total Body	GI Tract	Bone	Liver	Kidney	Thyroid	Lung	Skin
Nearest site boundary (0.68 Mile NE)	Ground deposition	2.42E-01	2.42E-01	2.42E-01	2.42E-01	2.42E-01	2.42E-01	2.42E-01	2.83E-01
	Vegetables								
	Adult	1.43E-01	3.94E-02	1.62E-01	1.98E-01	9.18E-02	1.53E-02	1.98E-02	3.59E-02
	Teen	1.36E-01	5.04E-02	2.66E-01	3.04E-01	1.35E-01	1.45E-01	2.53E-02	4.65E-02
	Child	1.51E-01	8.80E-02	6.36E-01	5.25E-01	2.30E-01	2.37E-01	5.07E-02	8.50E-02
	Meat								
	Adult	1.72E-02	7.95E-03	2.78E-02	2.24E-02	1.27E-02	2.51E-02	9.50E-03	7.57E-03
	Teen	9.54E-03	5.59E-03	2.33E-02	1.73E-02	9.48E-03	1.81E-02	6.93E-03	5.36E-03
	Child	1.08E-02	8.54E-03	4.35E-01	2.42E-02	1.36E-02	2.76E-02	1.03E-02	8.41E-03
	Cow Milk								
	Adult	9.59E-02	1.64E-02	1.11E-01	1.40E-01	5.84E-02	5.04E-01	2.75E-02	1.35E-02
	Teen	9.75E-02	2.33E-02	2.01E-01	2.41E-01	9.89E-02	7.96E-01	4.86E-02	1.96E-02
	Child	9.59E-02	3.99E-02	4.86E-01	4.22E-01	1.68E-01	1.57E00	8.16E-02	3.70E-02
	Infant	1.23E-01	6.86E-02	8.09E-01	8.20E-01	2.79E-01	3.80E00	1.47E-01	6.58E-02
	Inhalation								
	Adult	2.70E-02	2.51E-02	2.09E-03	2.78E-02	3.22E-02	5.78E-02	2.77E-02	2.78E-02
	Teen	2.65E-02	2.54E-02	2.93E-03	2.90E-02	2.68E-02	6.62E-02	2.81E-02	2.79E-02
	Child	2.30E-02	2.24E-02	3.96E-03	2.59E-02	2.38E-02	6.97E-02	2.51E-02	2.50E-02
	Infant	1.31E-02	1.29E-02	2.42E-03	1.56E-02	1.38E-02	5.64E-02	1.54E-02	1.55E-02
	Total Dose to Receptor[e]								
	Adult	5.25E-01	3.31E-01	5.45E-01	6.30E-01	4.37E-01	9.82E-01	3.27E-01	3.68E-01
	Teen	5.12E-01	3.47E-01	7.35E-01	8.33E-01	5.12E-01	1.27E00	3.51E-01	3.82E-01
	Child	5.23E-01	4.01E-01	1.80E00	1.24E00	6.77E-01	2.15E00	4.10E-01	4.38E-01
	Infant	3.78E-01	3.24E-01	1.05E00	1.08E00	5.35E-01	4.10E00	4.04E-01	3.64E-01

TABLE 11.3.3-4 (SHEET 3 OF 6)

Dose Rate (mrem/year)

Location	Pathway	Total Body	GI Tract	Bone	Liver	Kidney	Thyroid	Lung	Skin
Nearest Residence (1.2 mile WSW)	Ground deposition	8.01E-02	8.01E-02	8.01E-02	8.01E-02	8.01E-02	8.01E-02	8.01E-02	9.35E-02
	Inhalation								
	Adult	1.12E-02	1.06E-02	7.35E-04	1.15E-02	1.10E-02	2.28E-02	1.07E-02	1.06E-02
	Teen	1.22E-02	1.07E-02	1.03E-03	1.19E-02	1.12E-02	2.59E-02	1.08E-02	1.06E-02
	Child	1.07E-02	9.41E-03	1.39E-03	1.07E-02	9.91E-03	2.71E-02	9.57E-03	9.39E-03
	Infant	6.57E-03	5.41E-03	8.53E-04	6.36E-03	5.72E-03	2.17E-02	5.52E-03	5.40E-03
	Total Dose to Receptor								
	Adult	9.13E-02	9.07E-02	8.08E-02	9.16E-02	9.11E-02	1.03E-01	9.08E-02	1.04E-01
	Teen	9.23E-02	9.08E-02	8.11E-02	9.20E-02	9.13E-02	1.06E-01	9.09E-02	1.04E-01
	Child	9.08E-02	8.95E-02	8.15E-02	9.08E-02	9.00E-02	1.07E-01	8.97E-02	1.03E-02
	Infant	8.67E-02	8.55E-02	8.10E-02	8.65E-02	8.58E-02	1.02E-01	8.56E-02	9.89E-02
Nearest Vegetable garden (1.4 mile WSW)	Ground deposition	6.08E-02	6.08E-02	6.08E-02	6.08E-02	6.08E-02	6.08E-02	6.08E-02	7.10E-02
	Vegetables								
	Adult	4.05E-02	1.48E-02	5.15E-02	5.45E-02	2.78E-02	1.53E-02	4.90E-02	1.38E-02
	Teen	4.10E-02	1.94E-02	8.46E-02	8.32E-02	4.05E-02	1.96E-02	5.27E-02	1.84E-02
	Child	5.15E-02	3.57E-02	2.04E-01	1.46E-01	7.13E-02	3.68E-02	8.77E-02	3.50E-02
	Inhalation								
	Adult	9.27E-03	8.78E-03	1.14E-03	9.50E-03	9.07E-03	9.31E-03	1.82E-02	8.75E-03
	Teen	9.19E-03	8.83E-03	1.60E-03	9.83E-03	9.25E-03	9.52E-03	2.07E-02	8.80E-03
	Child	7.95E-03	7.80E-03	2.17E-03	8.79E-03	8.20E-03	8.61E-03	2.16E-02	7.79E-03
	Infant	4.56E-03	4.48E-03	1.33E-03	5.23E-03	4.73E-03	5.23E-03	1.70E-02	4.48E-03
	Total Dose to Receptor								
	Adult	1.11E-01	8.43E-02	1.13E-01	1.25E-01	9.77E-02	8.54E-02	1.28E-01	9.36E-02
	Teen	1.11E-01	8.90E-02	1.47E-01	1.54E-01	1.11E-01	8.99E-02	1.34E-01	9.82E-02
	Child	1.20E-01	1.04E-01	2.67E-01	2.16E-01	1.40E-01	1.06E-01	1.70E-01	1.14E-01
	Infant	6.54E-02	6.53E-02	6.21E-02	6.60E-02	6.55E-02	6.61E-02	7.78E-02	7.55E-02

TABLE 11.3.3-4 (SHEET 4 OF 6)

Dose Rate (mrem/year)

Location	Pathway	Total Body	GI Tract	Bone	Liver	Kidney	Thyroid	Lung	Skin
Nearest meat animal (3.1 mile SW)	Ground deposition	1.80E-02	1.80E-02	1.80E-02	1.80E-02	1.80E-02	1.80E-02	1.80E-02	2.10E-02
	Meat								
	Adult	2.32E-03	1.63E-03	5.57E-03	2.69E-03	1.98E-03	3.05E-03	1.72E-03	1.60E-03
	Teen	1.50E-03	1.21E-03	4.70E-03	2.08E-03	1.50E-03	2.25E-03	1.31E-03	1.19E-03
	Child	2.16E-03	1.99E-03	8.80E-03	3.45E-03	2.37E-03	3.57E-03	2.12E-03	1.98E-03
	Inhalation								
	Adult	4.08E-03	3.89E-03	2.12E-04	4.16E-03	4.01E-03	7.80E-03	3.92E-03	3.88E-03
	Teen	4.06E-03	3.91E-03	2.96E-04	4.29E-03	4.08E-03	8.81E-03	3.98E-03	3.90E-03
	Child	3.52E-03	3.46E-03	4.00E-04	3.83E-03	3.61E-03	9.12E-03	3.51E-03	3.46E-03
	Infant	2.01E-03	1.99E-03	2.45E-04	2.25E-03	2.09E-03	7.18E-03	2.02E-03	1.99E-03
	Total Dose to Receptor								
	Adult	2.44E-02	2.35E-02	2.38E-02	2.48E-02	2.39E-02	2.88E-02	2.36E-02	2.65E-02
	Teen	2.36E-02	2.31E-02	2.29E-02	2.44E-02	2.36E-02	2.90E-02	2.33E-02	2.61E-02
	Child	2.37E-02	2.34E-02	2.72E-02	2.53E-02	2.40E-02	3.07E-02	2.36E-02	2.65E-02
	Infant	2.00E-02	2.00E-02	1.82E-02	2.02E-02	2.01E-02	2.52E-02	2.00E-02	2.30E-02

TABLE 11.3.3-4 (SHEET 5 OF 6)

Dose Rate (mrem/year)

Location	Pathway	Total Body	GI Tract	Bone	Liver	Kidney	Thyroid	Lung	Skin
Nearest milk cow (4.60 mile SE)	Ground deposition	7.20E-03	7.20E-03	7.20E-03	7.20E-03	7.20E-03	7.20E-03	7.20E-03	8.40E-03
	Cow Milk								
	Adult	3.57E-03	1.21E-03	4.96E-03	4.86E-03	2.47E-03	1.73E-03	1.54E-03	1.12E-03
	Teen	4.03E-03	1.82E-03	9.07E-03	8.31E-03	4.08E-03	2.74E-02	2.57E-03	1.71E-03
	Child	5.17E-03	3.51E-03	2.20E-02	1.48E-02	7.36E-03	5.43E-02	4.74E-03	3.42E-03
	Infant	8.08E-03	6.44E-03	3.89E-02	2.87E-02	1.27E-02	1.29E-01	8.74E-03	6.35E-03
	Inhalation								
	Adult	1.83E-03	1.75E-03	9.22E-05	1.86E-03	1.79E-03	3.49E-03	1.76E-03	1.74E-03
	Teen	1.81E-03	1.76E-03	1.29E-04	1.92E-03	1.82E-03	3.95E-03	1.78E-03	1.75E-03
	Child	1.58E-03	1.55E-03	1.74E-04	1.71E-03	1.62E-03	4.08E-03	1.57E-03	1.55E-03
	Infant	9.03E-04	8.91E-04	1.07E-04	1.01E-03	9.32E-04	3.22E-03	9.07E-04	8.90E-04
	Total Dose to Receptor								
	Adult	1.26E-02	1.02E-02	1.23E-02	1.39E-02	1.15E-02	2.80E-02	1.05E-02	1.13E-02
	Teen	1.31E-02	1.08E-02	1.64E-02	1.74E-02	1.31E-02	3.86E-02	1.16E-02	1.18E-02
	Child	1.39E-02	1.23E-02	2.93E-02	2.38E-02	1.62E-02	6.55E-02	1.35E-02	1.34E-02
	Infant	1.62E-02	1.45E-02	4.62E-02	3.69E-02	2.08E-02	1.40E-01	1.69E-02	1.56E-02

TABLE 11.3.3-4 (SHEET 6 OF 6)

a. All data is on a per unit basis. Doses were calculated using the GASPAR code.

b. Evaluated at a location that could be occupied during the term of plant operation.

Appendix I design objectives – gaseous effluents (noble gases only):

Gamma dose in air - 10 mrad/year per unit
Beta dose in air - 20 mrad/year per unit
Dose to total body of individual - 5 mrem/year per unit
Dose to skin of individual - 15 mrem/year per unit

Annex to Appendix I, Docket RM-50-2. Design objectives are the same as Appendix I except on a per-site basis; therefore calculated doses should be multiplied by 2.

c. Evaluated at a location where an exposure pathway and dose receptor actually exist at the time of licensing.

Appendix I design objectives - radioiodines and particulates:

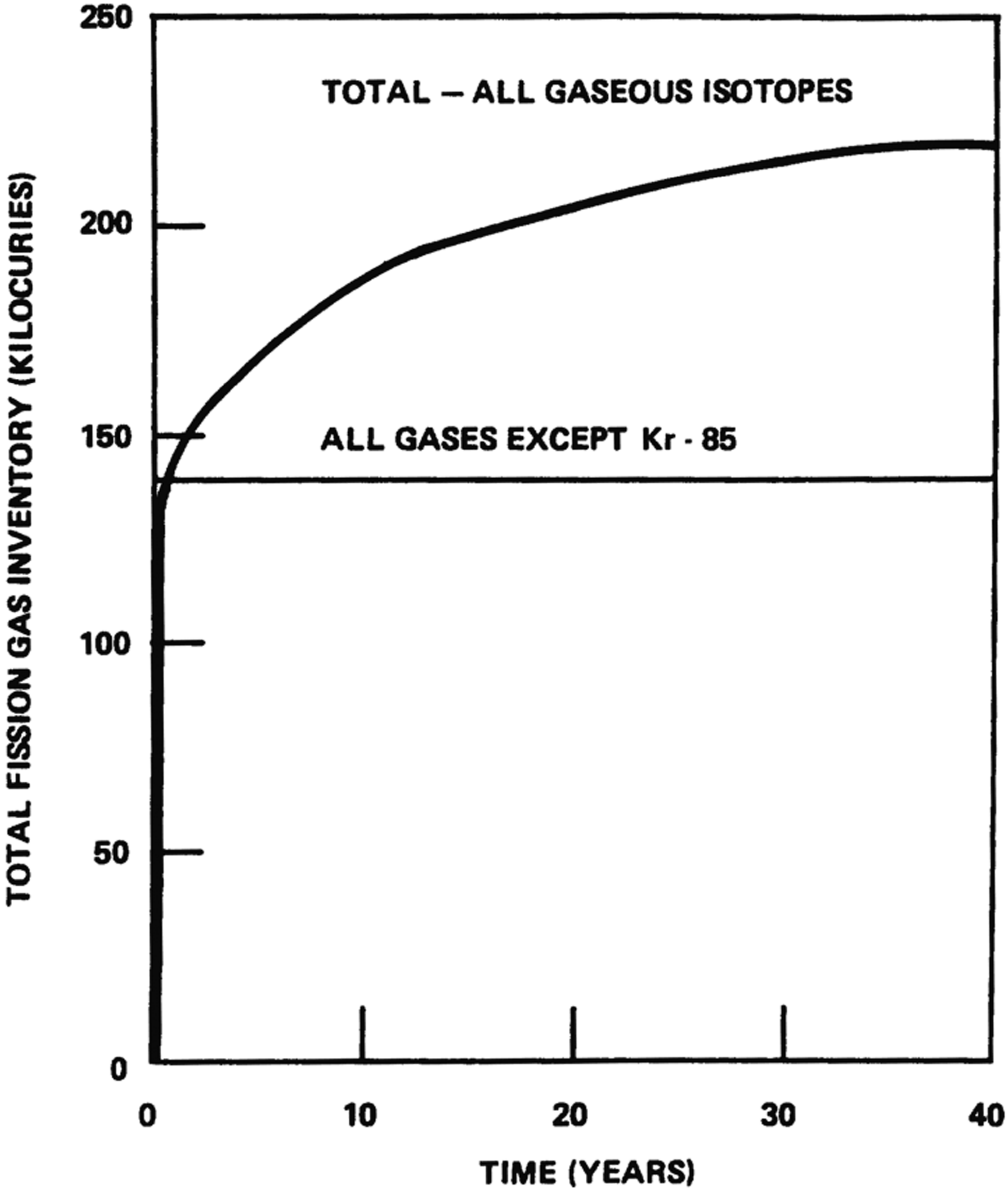
Dose to any organ from all pathways – 15 mrem/year per unit

Annex to Appendix I Docket RM-50-2 design objectives:

Dose to any organ from all pathways – 15 mrem/year per site 1-131 releases – 1 Ci/year per unit (reference table 11.3.3-3)

d. Total dose due to realistic pathways.

e. Provided for information only; a receptor is assumed present at the location of a potential pathway. This evaluation is based on the worst case χ /Q at the site boundary.



A review of the figure values for the effects of MUR power uprate determined the figure need not be updated.

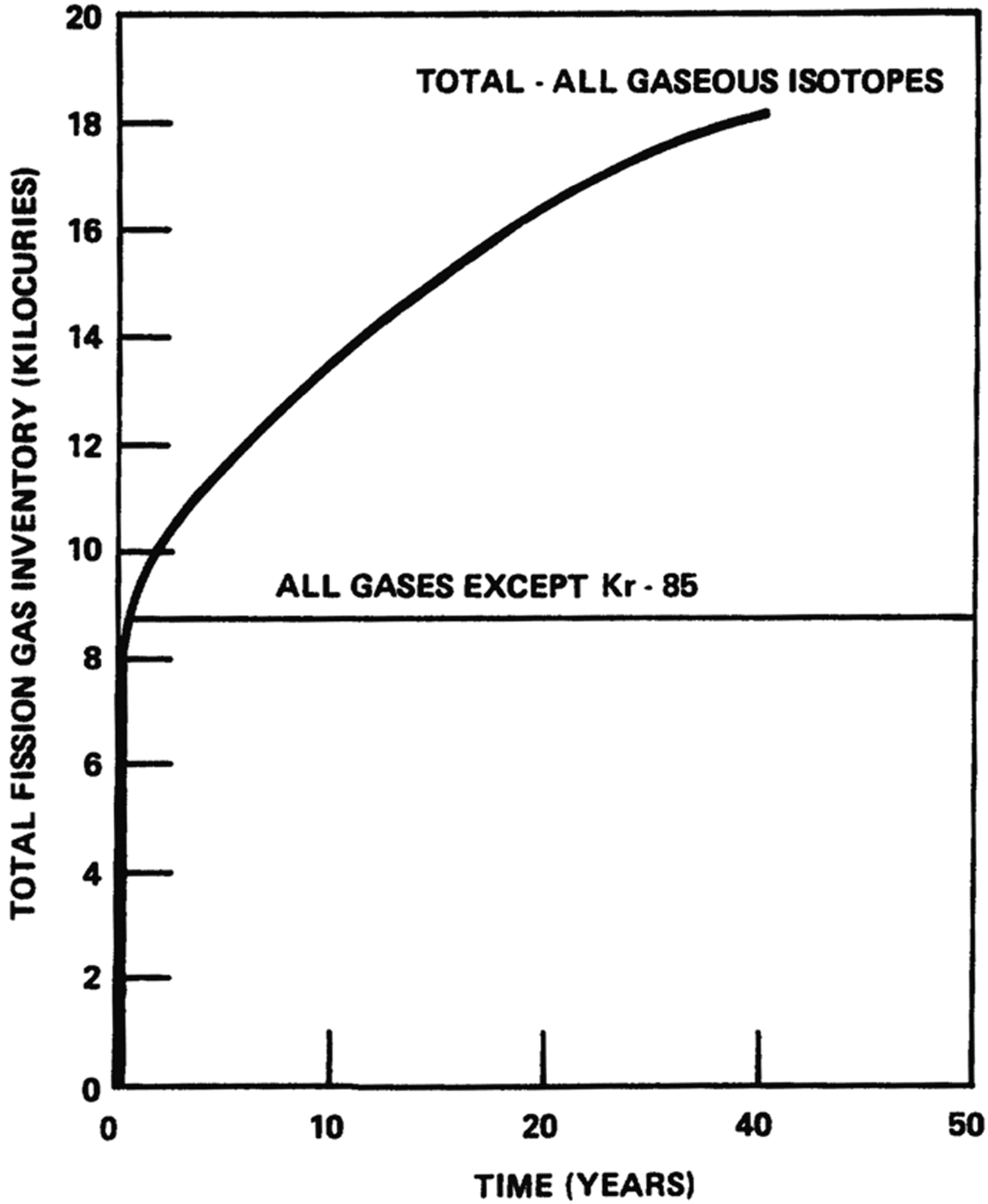
REV 16 10/10

SOUTHERN COMPANY

VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

GWPS FISSION GAS ACCUMULATION BASED ON CONTINUOUS CORE OPERATION OF 2 UNITS AT 3565 MWt WITH 1% FUEL DEFECTS

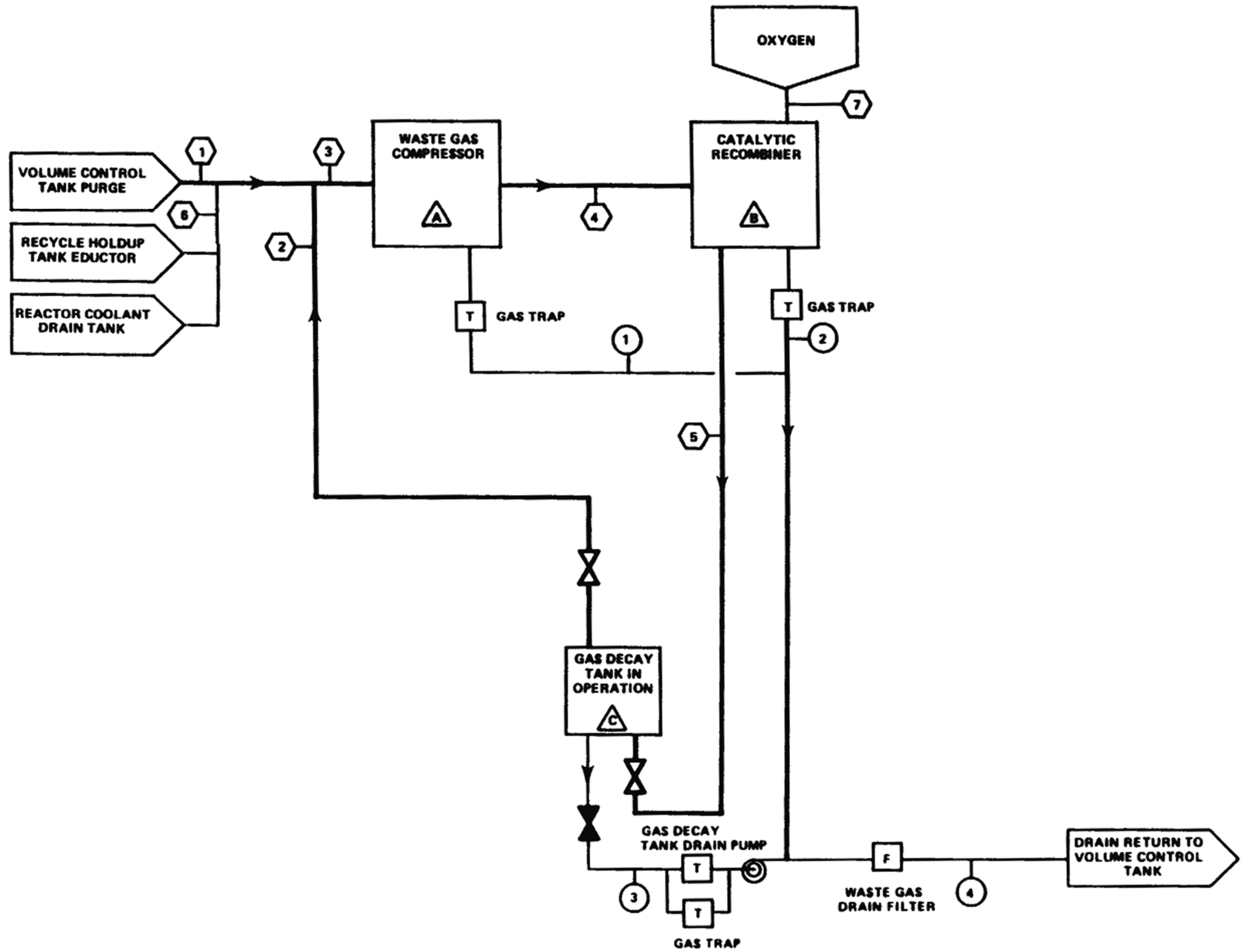
FIGURE 11.3.2–1



A review of the figure values for the effects of MUR power uprate determined the figure need not be updated.

REV 16 10/10

SOUTHERN COMPANY VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2	EXPECTED GWPS FISSION GAS ACCUMULATION BASED ON TABLE AND FULL–POWER OPERATION OF 2 UNITS AT 3565 MWt
	FIGURE 11.3.2–2



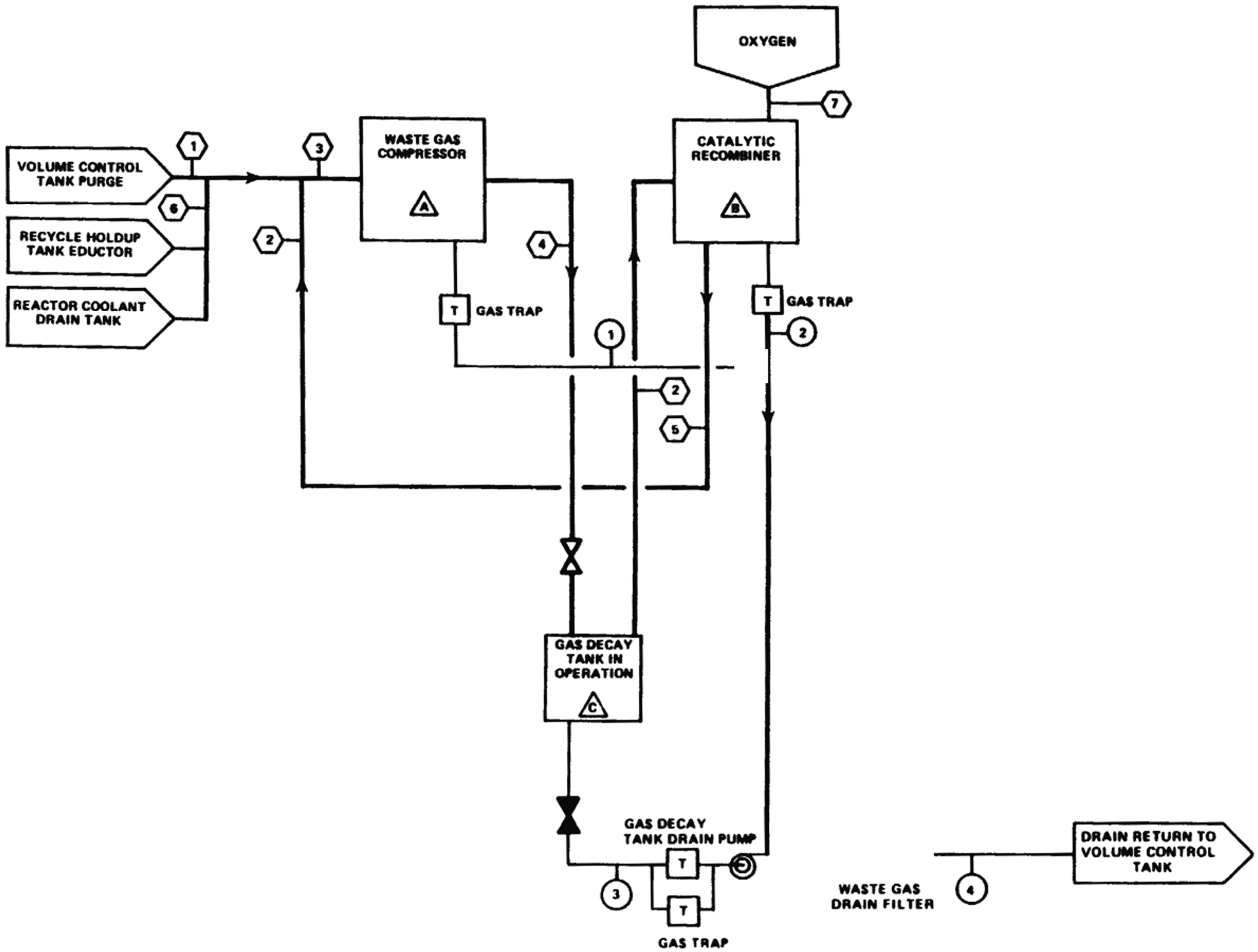
REV 14 10/07

SOUTHERN COMPANY
Energy to Serve Your World®

VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

PROCESS DIAGRAM DURING OPERATION WITH
GDT PRESSURE ≤ 20 psig

FIGURE 11.3.2–3



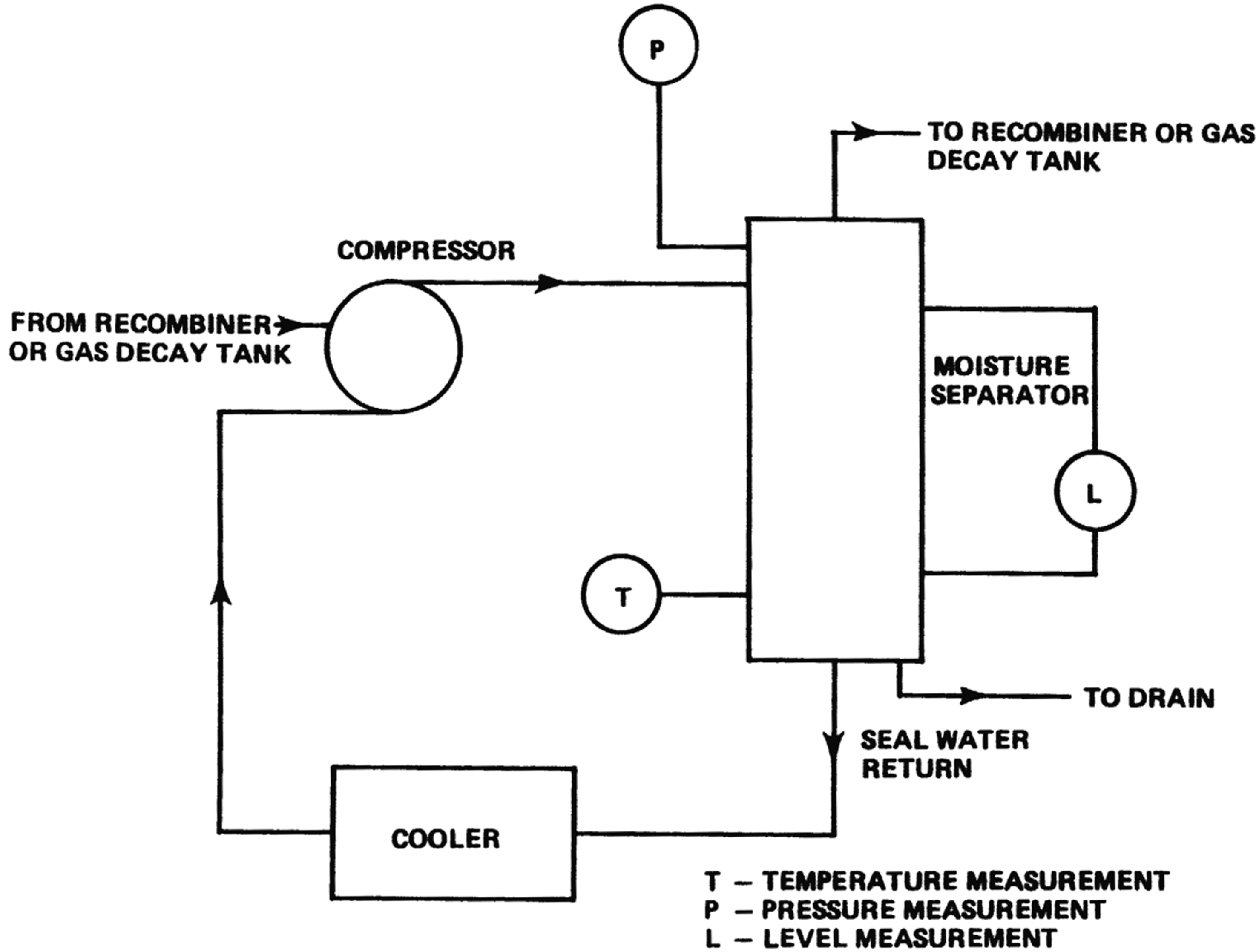
REV 14 10/07

SOUTHERN COMPANY
Energy to Serve Your World®

VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

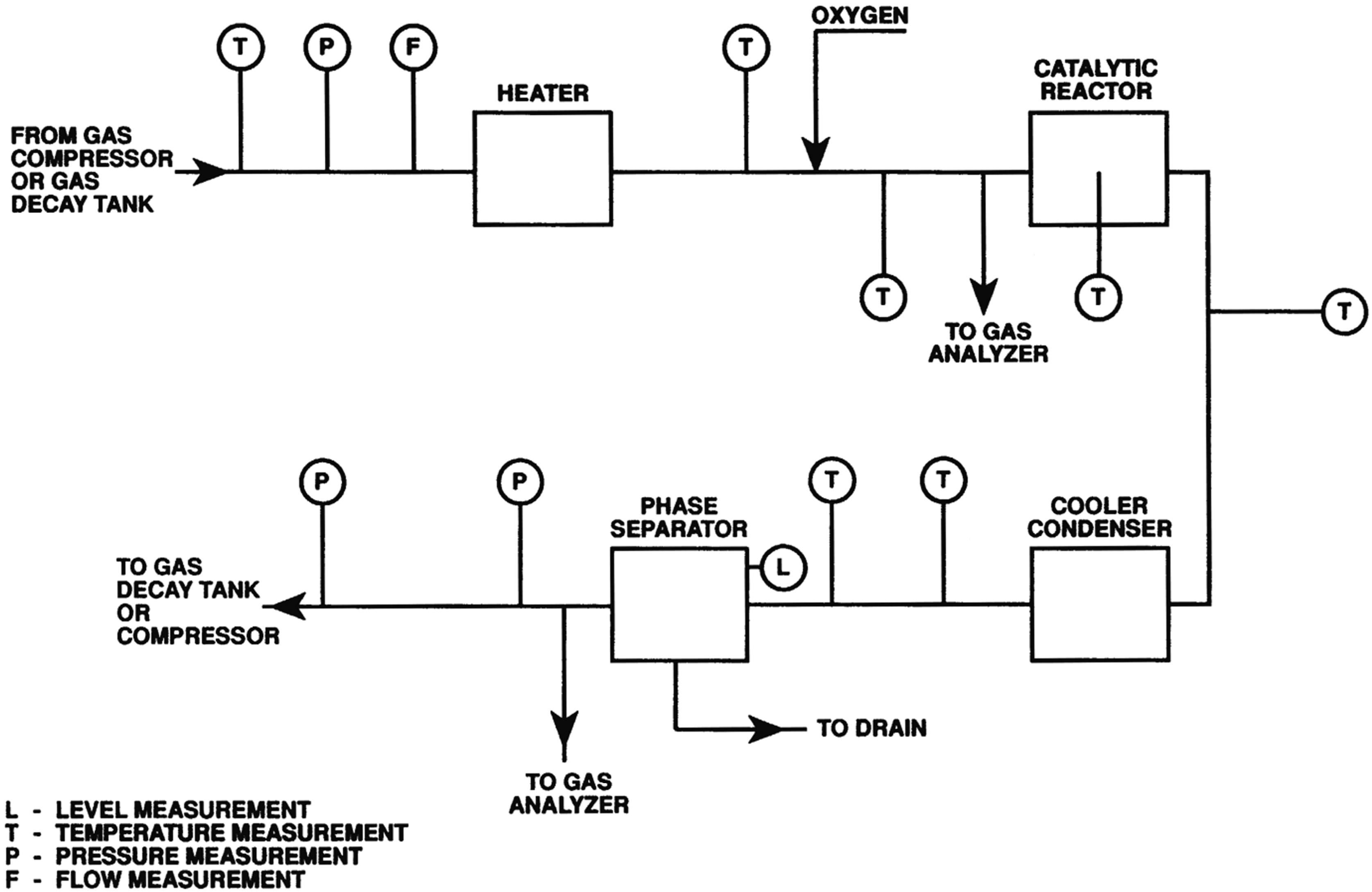
PROCESS DIAGRAM DURING OPERATION WITH GDT PRESSURE BETWEEN 20 AND 100 psig

FIGURE 11.3.2–4



REV 14 10/07

SOUTHERN COMPANY *Energy to Serve Your World®*	VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2	WASTE GAS COMPRESSOR PACKAGE FIGURE 11.3.2–5



REV 14 10/07

SOUTHERN COMPANY *Energy to Serve Your World®* VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2	CATALYTIC HYDROGEN RECOMBINER PACKAGE FIGURE 11.3.2–6

11.4 SOLID WASTE MANAGEMENT SYSTEM

The solid waste management system is designed to collect, process, and package all solid radioactive wastes generated as a result of normal plant operation, including anticipated operational occurrences. The packaged waste is stored until it is shipped to a waste processing facility or burial site. This system does not normally handle large waste materials such as activated core components.

11.4.1 DESIGN BASES

The solid waste management system is designed to meet the following objectives:

A. Provide packaging capability for spent radioactive resins from the liquid waste processing and steam generator blowdown systems.

B. Provide a means to package crud from backflushable filters.

C. Provide a means to package combustible dry solid wastes, such as paper, rags, and contaminated clothing.

D. Provide a means to transfer the spent filter cartridges from filter vessels to shielded disposal containers in a manner which minimizes radiation exposure to operating personnel and the spread of contamination.

E. Provide a means for compacting and packaging miscellaneous dry radioactive materials, such as paper, rags, and contaminated clothing.

F. Provide temporary onsite storage of packaged wastes in the event of delay or disruption of offsite shipping schedules.

G. Package radioactive solid wastes for offsite shipment and disposal in accordance with applicable Department of Transportation (DOT) and Nuclear Regulatory Commission (NRC) regulations, including 49 Code of Federal Regulations (CFR) 170-178 and 10 CFR 71.

The solid waste management system is designed and constructed in accordance with Regulatory Guide 1.143, as described in section 1.9, to meet the requirements of General Design Criterion (GDC) 60 of 10 CFR 50, Appendix A. The seismic design classification of the alternate radwaste, radwaste processing facility, and radwaste transfer buildings is provided in section 3.2. The radwaste processing facility is provided to allow use of portable radwaste systems (demineralizers, filtration system, dewatering/drying, solidification) supplied by contractors for handling solid wastes from plant operations during all plant conditions (paragraph 11.4.2.4). Dry active waste (DAW) locations are provided to process and store DAW (paragraph 11.4.2.5).

The solid waste management system design parameters are based on the radionuclide concentrations and volumes consistent with operating experience for similar reactor designs and with the source terms of section 11.1.

The solid waste management system airborne process effluents are released through the radwaste processing facility vent to the atmosphere as part of the 10 CFR 50, Appendix I analysis presented in subsection 11.3.3.

The solid waste management system is designed to collect, solidify, package, and store radioactive wastes so as to maintain radiation exposure to plant operation or maintenance personnel as low as is reasonably achievable (ALARA) in accordance with GDC 60 of 10 CFR 50, Appendix A, and Regulatory Guide 8.8 in order to maintain personnel exposures below 10 CFR 20 requirements. Access to the process equipment and solid waste storage areas is controlled to minimize personnel exposure by suitable barriers such as locked doors or gates or control cards. (See paragraph 12.3.1.2 and section 12.5.) The solid waste management system has been designed to conform with the design criteria of NRC Branch Technical Position ETSB 11-3.

11.4.1.1 Safety Design Bases

The solid waste management system performs no function related to the safe shutdown of the plant, and its failure does not adversely affect any safety-related system or component; therefore, the solid waste management system has no safety design basis.

11.4.1.2 Power Generation Design Bases

The solid waste management system is designed to minimize the volume of solidified wastes and provide temporary onsite storage for the packaged wastes. The system is designed for a 40-year service life[a], maximum reliability, minimum maintenance, and minimum exposure to operating and maintenance personnel. The system is designed to have sufficient capacity based on normal waste generation rates to ensure that maintenance or repair of the solid waste management system equipment does not impact power generation.

11.4.2 SYSTEM DESCRIPTION

11.4.2.1 General Description

The solid waste management system consists of the following which are shown in drawings 1X4DB148, 1X4DB148-1, 1X4DB148-2, 1X4DB148-3, 1X4DB148-4, AX4DB148-5, AX4DB148-6, AX4DB148-7, 1X4DB148-8, 1X4DB148-9, 1X4DB148-10, 1X4DB148-11, 1X4DB148-12, and 1X4DB148-13.

- Resin transfer.
- Backflushable filter system.
- Crud transfer.

The activity of the influents to the solid waste management system is dependent on the activities of various fluid systems, such as the boron recycle system, liquid waste processing system, chemical and volume control system, spent fuel pool cooling and cleanup system, steam generator blowdown system, and condensate polishing demineralizer system. Reactor

[a] The operating licenses for both VEGP units have been renewed and the original licensed operating terms have been extended by 20 years. In accordance with 10 CFR Part 54, appropriate aging management programs and activities have been initiated to manage the detrimental effects of aging to maintain functionality during the period of extended operation (see chapter 19).

coolant system activities and the decontamination factors for the systems given above also determine the influent activities to the solid waste management system.

Table 11.4.2-1 lists the parameters used in calculation of the estimated activity fed to the solid waste management system. Table 11.4.2-2 lists the estimated expected activities of wastes to be processed on an annual basis, their physical form and source, and their isotopic makeup and curie content. Table 11.4.2-3 provides the same information based on the maximum expected activities of the input wastes. The estimated annual quantities of radwaste input to the solid waste management system and the estimated quantities of radwaste to be shipped offsite are presented in table 11.4.2-4. The estimated expected curie and isotopic content of wastes to be shipped offsite for each waste category is presented in table 11.4.2-5. Table 11.4.2-6 presents the same information based on expected maximum activities.

Section 11.1 provides the bases for determination of liquid source terms which are used to calculate the solid waste source terms. The data presented in tables 11.4.2-2, 11.4.2-3, 11.4.2-5, and 11.4.2-6 are conservatively based on section 11.1 and the following additional information:

A. As a basis for the shipped-from-site activities given in tables 11.4.2-5 and 11.4.2-6, 30 days of decay prior to shipping is assumed. The major contributors to the activity levels in the system are radionuclides having long half-lives; therefore, further onsite decay of radionuclides provides limited benefit.

B. The miscellaneous dry and compacted waste volume is based on reference 1.

C. Shipping volumes from the cartridge filter handling system are based on packaging (per 55-gal drum):

   1. One spent filter cartridge.

   2. The drum is prelined with cement with an integral cavity to accept a filter.

Section 11.1 provides the bases for the quantity of primary and secondary resins. Section 11.2 provides the basis for the quantity of chemical drain wastes, while reference 1 provides the basis for the remaining quantities presented in table 11.4.2-4.

11.4.2.2 <u>Component Description</u>

Codes and standards applicable to the solid waste management system are listed in table 3.2.2-1. The solid waste management system is housed within buildings designed to meet the seismic requirements of Regulatory Guide (RG) 1.143. Conformance with RG 1.143 is to the extent specified in section 1.9.

11.4.2.3 <u>System Operation</u>

11.4.2.3.1 Resin Transfer System

The resin transfer system provides the capability for remote transfer of spent radioactive resin from the auxiliary building to the radwaste processing facility.

11.4.2.3.2 Backflushable Filter System

The backflushable filter system provides the means to remove and deliver radioactive crud from certain process streams to the crud transfer system. The system consists of the filters summarized in table 11.4.2-7, a backflushable filter crud tank, and a centrifugal pump. Each backflushable filter is operated in a similar manner. A sufficient buildup of crud on the filter signals an annunciator and the operator then initiates the automatic filter backflush sequence (Unit 1 only). The operator can manually carry out the backflush sequence if it is desired. The backflush sequence isolates the filter, opens a line to the backflushable crud tank, and introduces approximately 8 ft^3 of high-pressure (350 psig) nitrogen from a nitrogen accumulator through the filter. The nitrogen accumulator is supplied by the auxiliary gas system. The crud removed from the filters collects in the backflushable crud tank. After each backflush operation the lines and tank are water flushed. The contents of the backflushable crud tank may be slurried to the radwaste processing facility at operator discretion or upon high tank level annunciation.

11.4.2.3.3 Filter Handling System

The filter handling system is a semiremote system which provides the capability to remove spent radioactive cartridge filters from their filter housings and place them in drums with external shielding or use of a transfer cask for transport to the radwaste processing facility. The techniques used in the filter handling system included the consideration of the following:

- Operator exposure.
- Time and manpower requirements.
- Potential for the spread of contamination.
- Potential for mechanical difficulties.
- Logistics for filter handling.

The semiremote system, as the name implies, requires the operator to be in the proximity of the filters; however, operator exposure is minimized by distance and shielding.

The filter handling system consists of a working plug, a bell-shaped shielded transfer cask, a mobile cask-to-drum transfer station, and drum capper. The working plug is designed to fit in a hatch above the filter housing and has provisions for viewing the top of the filter housing and penetrations to allow long handled tools to be used for the removal of the filter housing top. The bell-shaped shielded transfer cask is used to retract the spent filter cartridge from its housing and transport it to the filter transfer station where the filter is placed in a shielded drum and capped. The filters in the shielded drums are transported to the radwaste processing facility for packaging. The filters may also be transported using the transfer cask directly to the radwaste processing facility for packaging.

11.4.2.4 <u>Portable Radwaste System</u>

VEGP will utilize vendor-supplied portable radwaste equipment to provide for disposal of spent resins, radioactive cartridge filters, backflushable filter crud, and chemical wastes via dewatering or solidification. Details of this equipment are shown on drawings AX4DB124-2, AX4DB124-3, and AX4DB124-4. In addition, a portable demineralizer system and a filtration system are available as alternate means of processing the contents of the waste holdup tank, chemical drain tank, floor drain tank, and boron recycle holdup tank. Details of this equipment are shown on drawings AX4DB124-2, AX4DB124-3, AX4DB124-4, and AX4DB124-5. These systems are housed in the radwaste processing facility which is shown on drawing AX4DE357.

Isolation valves are provided to allow processing of waste streams at the radwaste processing facility. The valves are manually and remotely operated to achieve the desired configuration. Delivery of waste streams to the radwaste processing facility is controlled from local panels near the waste stream source. Flanged connections are provided at the radwaste processing facility to interface with the vendor-supplied systems. Major components for portable radwaste systems typically include process liners, process skids, and control panels. A separate radwaste processing facility control room and dressout area is provided to facilitate system operation.

Radioactive condensate polishing demineralizer resins, backflushable filter crud, and spent resins from the liquid waste processing system and the steam generator blowdown system will be dewatered. The dewatering system supplied by vendor allows the water to be removed from the spent resins in the shipping containers. A vendor-supplied container vent is provided for the shipping containers thereby minimizing leakage into the building. A vent line to a monitored HVAC exhaust duct in the radwaste processing facility is provided. In addition, demineralizer resins from the portable demineralizers (as discussed in section 11.2) can be sluiced to the container fill skid for dewatering and disposal.

An NRC approved process control program (PCP) will be required of the vendor and appropriately referenced in the VEGP PCP prior to any actual operation. If the burial site does not accept dewatered resins or the waste form criteria cannot be met, VEGP will have the ability to solidify resins utilizing a portable solidification system. Cartridge filters will be loaded into liners for shipment offsite.

Solidification/dewatered liners will normally be shipped after filling and proper cure time, provided the proper shielding is available, without exceeding DOT radiation limits. If 49 CFR 173 dose limitations cannot be met with the available shielding, the liners are stored and allowed to decay until the appropriate shielding is available. Onsite storage for decay of short-lived radionuclides is accomplished both prior to solidification in holdup tanks and in appropriate onsite storage areas.

Short-term storage of solid radwaste for the purpose of activity reduction by radioactive decay is provided by shielded environmental containers located on a concrete pad adjacent to the southeast corner of the abandoned radwaste solidification building. High-integrity containers, containing the solid radwaste, are inserted into the shielded environmental containers which are right circular cylinders of reinforced concrete of sufficient thickness to provide appropriate shielding and normal environmental protection.

11.4.2.5 Dry Active Waste

Low level dry wastes are collected in drums, plastic bags, and other methods as approved by the health physics department, at appropriate locations throughout the plant, as dictated by the volume of these wastes generated during operation or maintenance.

Large components and equipment which have been activated during reactor operation and which are not amenable to solidification, compaction, or volume reduction are handled either by qualified plant personnel or by outside contractors specializing in radioactive materials handling and are packaged in boxes or shipping casks of appropriate size.

11.4.2.5.1 Dry Active Waste Processing

DAW, outage equipment, and other radioactively contaminated material are transported to a central location in containers or contained in some manner which will prevent any leakage of radioactive material during conditions incident to normal transportation. The radioactive material is packaged such that contamination on the outer container/containment surface is below administrative limits. The transport path remains within the owner-controlled area.

The central location determined by the Health Physics department has separate areas for incoming radwaste containers, handling of nonradioactive dry waste, and storage of empty containers. Radioactive waste is packaged such that it will comply with the criteria of 10 CFR part 71 to minimize the need for repackaging for shipment.

Outage equipment, and other radioactively contaminated material that are not shipped offsite for processing are packaged for onsite storage at several locations within the owner-controlled area.

11.4.2.5.2 Dry Active Waste Storage Building

DAW, outage equipment, and other potentially contaminated radioactive material are transported to multiple storage locations within the owner-controlled area in packaging that will comply with the criteria of 10 CFR Part 71 or in containers that will prevent leakage of any radioactive material incident to normal transportation. Lockable doors and/or a fence with lockable gates are provided. DAW may also be shipped offsite for long term processing and storage at an NRC approved facility.

11.4.2.5.3 Low-Level Waste Storage Facility

In lieu of immediate disposal of low-level waste due to the closing of the Barnwell facility to non-Atlantic Compact generators, a low-level waste storage facility is provided near the Unit 2 cooling tower for interim storage of low-level waste pending permanent disposal in an NRC-licensed facility. The low-level waste storage facility is a radiation control area and radioactive material storage area that consists of a storage pad, dense grade aggregate, and fencing with locked gates. Low-level waste stored in the facility will consist primarily of radioactive resins and filters but may also include irradiated components. Consistent with the classifications of 10 CFR Part 61, low-level waste that may be stored in the facility includes Class A, B, C, and greater-than-Class C waste. NRC-licensed disposal facilities are used when available to remove low-level waste from the facility.

Resins stored in the low-level waste storage facility for extended periods are stabilized resins that have been processed for long term storage and subsequent disposal. Dewatered resins may be stored in the low-level storage facility pending shipment of a sufficient quantity to fill a high-integrity container following offsite processing and volume reduction for long-term storage and subsequent disposal. Liquids or flammable materials are not stored in the low-level waste storage facility.

The low-level waste is stored in high-integrity containers and inserted into shielded environmental containers. The shielded environmental containers are right circular cylinders of reinforced concrete of sufficient thickness to provide appropriate shielding and normal environmental protection for the high-integrity container. Supplemental shielding is provided, as necessary, to maintain the doses outside the low-level waste facility fence to Zone I radiation area levels described in table 12.3.1-1. The shielded environmental containers incorporate a vent and drain pipe that is routed through the container wall that provides access for performance of periodic inspection of the volume surrounding the high-integrity container to verify container integrity. The vent and drain pipe is designed to prevent rain water from entering the shielded environmental container as well as preventing any material from escaping from between the high-integrity container and the shielded environmental container. To minimize the potential for spread of contamination, high-integrity containers are not opened at the low-level waste facility.

The high-integrity containers are loaded inside the radwaste processing facility and transported to the low-level waste storage facility inside a shipping cask or transfer bell. The high-integrity containers are transferred to shielded environmental containers at the low-level storage facility using a mobile crane. Transfer of the high-integrity container to the shielded environmental container is not performed during inclement weather to minimize the potential for a tornado missile to impact the high-integrity container when missile protection is not provided by the shielded environmental container.

Other suitable containers designed for a 300-year life equipped with a special vent design that allows depressurization without the migration of radioactive material may be used as described above in lieu of high-integrity containers as appropriate based on the content of each container.

11.4.2.6 References

1. "A Waste Inventory Report for Reactor and Fuel-Fabrication Facility Wastes," ONWI-20, NUS-3314, March 1979.

11.4.3 SAFETY EVALUATION

The solid radwaste system has no safety design basis.

TABLE 11.4.2-1

PARAMETERS USED IN THE CALCULATION OF ESTIMATED
ACTIVITY TO BE SOLIDIFIED

Source	Input Rate (ft^3/year)	Activity and Basis
Condensate polishing spent resin	2850	(a)
Waste processing solidification spent resin	861	(a)
Decant	861	(a)
Evaporator concentrates tanks	2810	(b)
Contaminated oil	200	(c)
Steam generator blowdown spent resin	768	(a)
Waste processing system liquid spent resin	516	(a)
Backflushable filter crud	345	(d)

a. Spent resin design bases and activities are identified in section 12.2.1, and tables 12.2.1-10 and 12.2.1-29.

b. Evaporator concentrates activities and bases are identified in sections 11.2 and 12.2.1.

c. Contaminated oil does not contain significant quantities of radioactivity.

d. For the purposes of determining activities collected by filters, it was assumed that 1 percent of the upstream demineralizer activity was deposited on the respective filter.

TABLE 11.4.2-2 (SHEET 1 OF 2)

ESTIMATED EXPECTED ANNUAL ACTIVITIES OF THE INFLUENTS TO THE SOLID RADWASTE SYSTEM (Ci/year/unit)

Isotope	Spent Resins (Primary)	Spent Resins (Secondary)	Chemical Waste	Radwaste Processing Facility Resins	Backflushable Filter	Cartridge Filters	Dry and Compacted Waste[(a)]
Br-83	6.07(-1)	1.34(-3)	8.59(-5)	5.57(3)	2.92(1)	-	-
Br-84	-	-	1.05(-5)	-	-	-	-
I-130	1.37(0)	8.05(-3)	1.90(-4)	1.23(-2)	1.63(-1)	3.78(-1)	-
I-131	2.47(3)	2.35(1)	2.48(-1)	8.94(0)	1.20(2)	5.78(-1)	-
I-132	1.19(1)	3.73(-1)	1.53(-2)	7.56(-1)	5.72(-1)	9.63(-3)	-
I-133	3.78(2)	2.81(0)	5.51(-2)	3.54(0)	1.85(1)	1.05(-1)	-
I-134	2.15(0)	2.90(-3)	3.09(-4)	2.61(-2)	1.06(-1)	-	-
I-135	6.38(1)	3.11(-1)	9.28(-3)	6.02(-1)	2.00(0)	1.87(-2)	-
Rb-86	-	1.72(-2)	1.03(-4)	3.22(-3)	-	1.55(-1)	-
Rb-88	3.03(0)	-	4.52(-4)	2.94(-2)	1.48(0)	-	-
Cs-134	2.61(2)	5.90(1)	3.79(-2)	1.04(0)	1.06(1)	1.66(0)	-
Cs-136	1.30(1)	1.94(0)	1.47(-2)	4.87(-1)	4.39(-1)	7.25(-2)	-
Cs-137	2.14(2)	5.03(1)	2.79(-2)	7.64(-1)	9.71(0)	1.40(0)	-
Ba-137m	2.04(2)	4.81(1)	2.61(-2)	7.14(-1)	9.31(0)	1.35(0)	-
Cr-51	4.67(1)	5.66(-1)	2.37(-3)	7.12(-2)	2.14(0)	1.29(-2)	-
Mn-54	7.87(-1)	1.30(-2)	4.50(-4)	1.24(-2)	3.80(-2)	2.81(-4)	-
Fe-55	2.32(2)	3.83(0)	2.34(-3)	6.43(-2)	1.15(1)	1.06(-1)	-
Fe-59	5.20(1)	4.81(0)	1.33(-3)	3.86(-2)	1.77(0)	1.10(-2)	-
Co-58	8.16(2)	1.19(1)	2.20(-2)	6.26(-1)	3.98(1)	2.68(-1)	-
Co-60	2.73(2)	5.11(0)	2.93(-3)	8.03(-2)	1.36(1)	1.13(-1)	-
Sr-89	1.99(1)	1.98(-1)	4.85(-4)	1.39(-2)	9.91(-1)	9.68(-3)	-
Sr-90	1.69(0)	2.72(-2)	1.47(-5)	-	8.44(-2)	6.01(-4)	-
Sr-91	2.98(-1)	1.31(-3)	4.57(-5)	2.80(-3)	1.47(-2)	-	-
Ba-140	2.98(0)	2.98(-2)	2.30(-4)	7.62(-3)	1.49(-1)	1.96(-3)	-
Y-90	6.62(-1)	1.09(-2)	1.11(-5)	-	3.31(-2)	2.19(-4)	-
Y-91m	1.86(-1)	-	2.89(-5)	1.87(-3)	-	1.56(-4)	-
Y-91	2.56(-1)	4.46(-2)	9.14(-5)	2.60(-3)	1.32(-2)	8.73(-4)	-
La-140	1.45(1)	6.88(-2)	2.40(-4)	7.32(-3)	7.25(-1)	1.37(-3)	-
Zr-95	1.46(-1)	4.21(-2)	8.37(-5)	2.38(-3)	-	8.41(-4)	-
Nb-95	-	5.43(-2)	7.63(-5)	2.06(-3)	-	1.08(-3)	-
Mo-99	-	2.34(0)	3.72(-2)	1.89(0)	2.36(-2)	4.68(-2)	-
Tc-99m	-	1.44(0)	3.92(-2)	1.96(1)	1.46(-2)	2.88(-2)	-
Ru-103	1.26(-1)	1.95(-2)	6.03(-5)	1.76(-3)	-	3.89(-4)	-
Ru-106	-	2.00(-2)	1.45(-5)	3.99(-3)	-	4.00(-4)	-
Rh-103m	-	2.96(-2)	6.04(-5)	1.76(-3)	-	4.31(-4)	-
Rh-106	-	5.13(-1)	-	-	-	1.03(-2)	-
Te-132	-	9.18(-1)	1.36(-2)	6.56(-1)	-	1.84(-2)	-
Te-125m	-	1.78(-2)	3.95(-5)	1.12(-3)	-	3.04(-3)	-
Te-127m	-	2.99(-1)	3.94(-4)	1.09(-2)	-	5.98(-3)	-
Te-127	-	3.83(-1)	4.32(-4)	1.36(-2)	-	4.80(-3)	-
Te-129m	-	4.93(-1)	1.80(-3)	5.31(-2)	-	9.88(-3)	-
Te-129	-	4.19(-1)	1.16(-3)	3.46(-2)	-	8.59(-3)	-
Te-131m	-	2.86(-2)	5.19(-4)	3.20(-2)	-	5.73(-3)	-

TABLE 11.4.2-2 (SHEET 2 OF 2)

Isotope	Spent Resins (Primary)	Spent Resins (Secondary)	Chemical Waste	Radwaste Processing Facility Resins	Backflushable Filter	Cartridge Filters	Dry and Compacted Waste[(a)]
Te-131	-	5.23(-3)	-	-	-	1.05(-3)	-
Ce-141	-	2.47(-2)	9.09(-3)	2.68(-3)	-	4.91(-4)	-
Ce-144	-	5.92(-2)	4.78(-5)	1.31(-3)	-	1.91(-3)	-
Pr-143	-	7.94(-3)	5.84(-5)	1.88(-3)	-	1.59(-4)	-
Pr-144	-	6.85(-2)	4.78(-5)	1.31(-3)	-	1.37(-3)	-
TOTAL	5.07(3)	2.20(2)	5.63(-1)	4.02(1)	2.44(2)	6.02(0)	7.6(-1)

a. Based on data obtained from reference 1 (a total activity of 7.6(-1) Ci/year from dry and compacted wastes). Major isotopic contributions are from Cr-51, Co-58, Co-60, Mn-54, Cs-134, and Cs-137.

TABLE 11.4.2-3 (SHEET 1 OF 2)

ESTIMATED MAXIMUM ANNUAL ACTIVITIES OF THE INFLUENTS TO THE SOLID RADWASTE SYSTEM (Ci/year/unit)

Isotope	Spent Resins (Primary)	Spent Resins (Secondary)	Chemical Waste	Radwaste Processing Facility Resins	Backflushable Filter	Cartridge Filters	Dry and Compacted Waste[(a)]
Br-83	1.07(1)	-	1.51(-3)	9.80(-1)	5.14(-1)	1.73(-3)	-
Br-84	1.15(0)	-	1.65(-4)	1.07(-2)	-	-	-
I-129	-	2.45(0)	-	-	-	4.90(-2)	-
I-130	1.18(1)	-	1.74(-3)	1.12(-1)	5.76(-1)	3.45(-3)	-
I-131	2.49(4)	2.33(2)	2.48(0)	1.32(2)	1.21(3)	5.78(0)	-
I-132	3.02(2)	1.04(1)	3.89(-1)	1.58(1)	1.47(1)	2.45(-1)	-
I-133	3.97(3)	2.94(1)	5.79(-1)	1.34(2)	1.93(2)	1.10(0)	-
I-134	2.38(1)	3.00(-2)	3.32(-3)	2.81(-1)	1.14(0)	-	-
I-135	6.99(2)	3.36(0)	1.02(-1)	2.76(-1)	3.34(1)	2.05(-1)	-
Rb-86	4.14(1)	4.31(0)	2.56(-2)	8.05(-1)	5.26(-1)	1.55(-1)	-
Rb-88	8.43(1)	-	9.43(-2)	6.14(-1)	3.29(0)	-	-
Rb-89	3.50(0)	-	-	-	-	-	-
Cs-134	2.60(4)	5.21(3)	1.09(1)	1.13(2)	1.02(3)	1.47(2)	-
Cs-136	2.72(3)	3.97(2)	3.06(0)	1.42(2)	1.06(2)	1.51(1)	-
Cs-137	2.13(4)	3.96(3)	2.20(0)	7.30(1)	8.27(2)	1.10(2)	-
Ba-137m	2.02(4)	3.74(3)	6.91(0)	6.91(1)	7.84(2)	1.05(2)	-
Cs-138	3.26(1)	-	-	-	1.19(0)	2.06(-3)	-
Cr-51	1.35(2)	1.64(0)	6.86(-3)	2.06(-1)	6.09(0)	3.72(-2)	-
Mn-54	5.08(1)	8.13(-1)	2.90(-2)	8.00(-1)	2.42(0)	1.81(-2)	-
Mn-56	1.39(1)	-	-	-	1.93(-1)	-	-
Fe-55	2.90(2)	4.78(0)	2.93(-3)	8.04(-2)	1.45(-1)	1.06(-1)	-
Fe-59	1.81(1)	2.51(-1)	6.91(-4)	2.01(-2)	8.37(-1)	5.70(-3)	-
Co-58	7.64(2)	1.11(1)	2.06(-2)	5.87(-1)	3.70(1)	2.51(-1)	-
Co-60	2.59(2)	4.85(0)	1.88(-2)	7.63(-2)	1.28(1)	1.07(-1)	-
Sr-89	2.38(2)	2.37(0)	5.81(-3)	1.64(-1)	1.17(1)	1.16(-1)	-
Sr-90	2.03(1)	3.26(-1)	1.76(-4)	-	1.01(0)	7.21(-3)	-
Sr-91	2.65(0)	-	4.05(-4)	2.48(-2)	2.30(-1)	-	-
Ba-140	5.71(1)	5.69(-1)	4.40(-3)	1.46(-1)	2.86(0)	1.33(-2)	-
Y-90	1.99(1)	3.28(-1)	5.14(-4)	-	9.94(-1)	6.56(-3)	-
Y-91m	-	-	2.33(-4)	1.51(-2)	-	-	-
Y-91	2.24(0)	3.83(-1)	8.02(-4)	2.28(-2)	1.16(-1)	7.66(-3)	-
La-140	5.71(1)	6.01(-1)	2.10(-3)	6.40(-2)	2.86(-2)	1.20(-2)	-
Zr-95	1.56(0)	4.48(-1)	8.93(-4)	2.54(-2)	-	8.96(-3)	-
Nb-95	-	6.92(-1)	9.72(-4)	2.63(-2)	-	1.38(-2)	-
Mo-99	-	2.02(1)	3.21(-1)	3.16(1)	2.01(-1)	4.03(-1)	-
Tc-99m	-	1.91(1)	5.20(-1)	2.96(1)	1.91(-1)	3.82(-1)	-
Ru-103	1.56(0)	2.41(-1)	7.47(-4)	2.18(-2)	-	4.82(-3)	-
Ru-106	-	2.80(-1)	1.85(-4)	-	-	5.60(-3)	-

TABLE 11.4.2-3 (SHEET 2 OF 2)

Isotope	Spent Resins (Primary)	Spent Resins (Secondary)	Chemical Waste	Radwaste Processing Facility Resins	Backflushable Filter	Cartridge Filters	Dry and Compacted Waste[a]
Rh-103m	-	2.41(-1)	6.76(-4)	6.68(-2)	-	4.83(-3)	-
Rh-106	-	6.54(0)	-	-	-	1.31(-1)	-
Te-132	-	9.50(0)	1.41(-1)	1.28(1)	-	1.90(-1)	-
Te-125m	-	1.72(-1)	3.81(-4)	1.08(-2)	-	3.44(-3)	-
Te-127m	-	3.10(0)	4.08(-3)	1.13(-1)	-	6.20(-2)	-
Te-127	-	3.13(0)	5.64(-3)	1.78(-1)	-	6.26(-2)	-
Te-129m	-	6.69(0)	2.44(-2)	7.20(-1)	-	1.34(-1)	-
Te-129	-	4.30(0)	1.16(-2)	3.46(-1)	-	8.59(-2)	-
Te-131m	-	2.87(-1)	5.19(-3)	9.40(-1)	-	5.73(-3)	-
Te-131	-	-	-	-	-	1.05(-3)	-
Te-134	-	-	-	-	-	-	-
Ce-141	-	2.18(-1)	8.60(-4)	2.38(-2)	-	4.36(-3)	-
Ce-143	-	-	1.09(-4)	-	-	-	-
Ce-144	-	6.98(-1)	5.65(-4)	1.55(-2)	-	1.40(-2)	-
Pr-143	-	-	7.22(-4)	2.32(-2)	-	1.96(-3)	-
Pr-144	-	7.03(-1)	4.89(-4)	1.35(-2)	-	1.41(-2)	-
TOTAL	1.02(5)	1.37(4)	2.79(1)	7.61(2)	5.04(3)	3.86(2)	7.6(-1)

a. Based on data obtained from reference 1 (a total activity of 7.6(-1) Ci/year from dry and compacted wastes). Major isotopic contributions are from Cr-51, Co-58, Co-60, Mn-54, Cs 134, and Cs-137.

TABLE 11.4.2-4

ESTIMATED ANNUAL QUANTITIES
OF RADWASTE

Source	Influent to Solid Radwaste (ft^3)	Quantity of Drums Produced[a]
Spent resins (primary)	516	99.0
Spent resins (secondary)	768	147.4
Radwaste processing facility resins	2810	1077
Backflushable filter crud	345	65.3
Dry waste	18,480	994
Filter cartridges	200 elements	200
Condensate demineralizer powdex resins	2,850	484.7

a. Containers other than 55-gal drums are used in conjunction with the radwaste processing facility systems.

TABLE 11.4.2-5

ESTIMATED EXPECTED ANNUAL ACTIVITIES IN SHIPPED RADWASTE (Ci/year/unit)

Isotope	Spent Resins (Primary)	Spent Resins (Secondary)	Chemical Waste	Radwaste Processing Facility Resins	Backflushable Filter	Cartridge Filters	Dry and Compacted Waste[a]
Sr-89	1.31(1)	1.31(-1)	3.21(-4)	9.21(-3)	6.57(-1)	6.41(-3)	-
Sr-90	1.69(0)	2.72(-2)	1.47(-5)	3.97(-4)	8.42(-2)	5.60(-4)	-
Y-90	1.69(0)	2.72(-2)	1.47(-5)	3.97(-4)	8.42(-2)	5.60(-4)	-
Y-91	1.83(-1)	3.14(-2)	6.47(-5)	1.86(-3)	9.42(-3)	6.14(-4)	-
Zr-95	1.06(-1)	3.07(-2)	6.09(-5)	1.73(-3)	2.23(-3)	6.12(-4)	-
Nb-95m	-	-	1.50(-6)	-	-	-	-
Nb-95	5.51(-2)	4.60(-2)	7.39(-5)	2.04(-3)	1.50(-3)	9.15(-4)	-
Mo-99	-	1.24(-3)	1.98(-5)	1.01(-3)	-	-	-
Tc-99m	-	3.27(-3)	5.21(-5)	2.65(-3)	-	-	-
Ru-103	7.45(-2)	1.15(-2)	3.57(-5)	1.04(-3)	1.61(-3)	2.30(-4)	-
Ru-106	-	1.89(-2)	1.37(-5)	3.77(-4)	-	2.78(-4)	-
Te-129m	-	2.65(-1)	9.68(-4)	2.86(-2)	2.65(-3)	5.31(-3)	-
Te-129	-	2.12(-1)	7.75(-4)	2.29(-2)	2.12(-3)	4.26(-3)	-
Te-132	-	1.53(-3)	2.26(-5)	1.09(-3)	-	-	-
I-131	1.87(2)	1.78(0)	1.86(-2)	6.78(-1)	9.09(0)	4.38(-2)	-
I-132	-	1.57(-3)	2.33(-5)	1.31(-3)	-	-	-
Cs-134	2.54(2)	5.74(1)	3.69(-2)	1.01(0)	1.03(1)	1.62(0)	-
Cs-136	2.63(0)	3.92(-1)	2.97(-3)	9.84(-2)	8.87(-2)	1.46(-2)	-
Cs-137	2.14(2)	5.02(1)	2.79(-2)	7.63(-1)	9.69(0)	1.40(0)	-
Ba-137m	2.02(2)	4.75(1)	2.64(-2)	7.22(-1)	9.17(0)	1.32(0)	-
Ba-140	5.87(-1)	5.87(-3)	4.53(-5)	1.50(-3)	2.94(-2)	3.86(-4)	-
La-140	6.76(-1)	6.76(-3)	5.22(-5)	1.73(-3)	3.38(-2)	4.44(-4)	-
Ce-141	-	1.30(-2)	4.78(-5)	1.41(-3)	-	2.58(-4)	-
Ce-143	-	-	-	-	-	-	-
Ce-144	-	5.50(-2)	4.44(-5)	1.22(-3)	-	1.11(-3)	-
Pr-143	-	1.73(-3)	1.29(-5)	4.21(-4)	-	-	-
Pr-144	-	5.50(-2)	4.44(-5)	1.22(-3)	-	1.11(-3)	-
Cr-51	2.21(1)	2.67(-1)	1.12(-3)	3.36(-2)	1.01(0)	6.09(-3)	-
Mn-54	7.36(-1)	1.22(-2)	4.21(-4)	1.16(-2)	3.56(-2)	2.63(-4)	-
Fe-59	3.26(1)	3.02(0)	8.34(-4)	2.42(-2)	1.11(0)	6.90(-3)	-
Co-58	6.10(2)	8.89(0)	1.64(-2)	4.68(-1)	2.97(1)	2.00(-1)	-
Co-60	2.70(2)	5.06(0)	2.90(-3)	7.94(-2)	1.35(1)	1.12(-1)	-
Rh-106	-	1.89(-2)	1.37(-5)	3.77(-4)	-	3.78(-4)	-
Te-127m	-	2.47(-1)	3.26(-4)	9.01(-3)	2.47(-3)	4.94(-3)	-
Te-127	-	2.48(-1)	3.27(-4)	9.04(-3)	2.48(-3)	4.96(-3)	-
Rh-103m	7.45(-2)	1.15(-2)	3.57(-5)	1.04(-3)	1.61(-3)	2.30(-4)	-
TOTAL	1.81(3)	1.76(2)	1.38(-1)	3.99(0)	8.46(1)	4.76(0)	7.6(-1)

a. Based on data obtained from reference 1 (a total activity of 7.6(-1) Ci/year from dry and compacted wastes). Major isotopic contributions are from Cr-51, Co-58, Co-60, Mn-54, Cs-134, and Cs-137.

TABLE 11.4.2-6

ESTIMATED MAXIMUM ANNUAL ACTIVITIES IN SHIPPED RADWASTE (Ci/year/unit)

Isotope	Spent Resins (Primary)	Spent Resins (Secondary)	Chemical Waste	Radwaste Processing Facility Resins	Backflushable Filter	Cartridge Filters	Dry and Compacted Waste[(a)]
Sr-89	1.58(2)	1.57(0)	3.85(-3)	1.09(-1)	7.75(0)	7.68(-2)	-
Sr-90	2.03(1)	3.25(-1)	1.76(-4)	-	1.01(0)	7.20(-3)	-
Y-90	2.03(1)	3.25(-1)	1.76(-4)	-	1.01(0)	7.20(-3)	-
Y-91	1.60(0)	2.69(-1)	5.67(-4)	1.63(-2)	8.38(-2)	5.39(-3)	-
Zr-95	1.14(0)	3.26(-1)	6.50(-4)	1.85(-2)	2.60(-2)	6.52(-3)	-
Nb-95	-	5.52(-1)	8.75(-4)	2.42(-2)	1.73(-2)	1.10(-2)	-
Mo-99	-	-	1.71(-4)	1.68(-2)	-	-	-
Tc-99m	-	-	4.49(-4)	4.42(-2)	-	-	-
Ru-103	-	1.43(-1)	4.42(-1)	1.29(-2)	1.99(-2)	2.85(-3)	-
Ru-106	-	2.65(-1)	1.75(-4)	-	-	5.29(-3)	-
Te-129m	-	3.60(0)	1.31(-2)	3.87(-1)	3.59(-2)	7.21(-2)	-
Te-129	-	2.88(0)	1.05(-2)	3.10(-1)	2.88(-2)	5.77(-2)	-
Te-132	-	-	2.35(-4)	2.13(-2)	-	-	-
I-129	-	2.45(0)	-	-	2.45(-2)	4.90(-2)	-
I-131	1.89(3)	2.52(1)	1.88(-1)	1.00(1)	9.17(1)	4.38(-1)	-
Cs-134	2.53(4)	5.07(3)	1.06(1)	1.10(2)	9.92(2)	1.43(2)	-
Cs-136	5.49(2)	8.02(1)	6.18(-1)	2.87(1)	2.14(1)	3.05(0)	-
Cs-137	2.13(4)	3.95(3)	2.20(0)	7.29(1)	8.25(2)	1.10(2)	-
Ba-137m	2.01(4)	3.74(3)	2.08(0)	6.89(1)	7.81(2)	1.04(2)	-
Ba-140	1.13(1)	1.12(-1)	8.67(-4)	2.88(-2)	5.63(-1)	2.62(-3)	-
La-140	1.30(1)	1.29(-1)	9.97(-4)	3.31(-2)	6.48(-1)	3.01(-3)	-
Ce-141	-	1.15(-1)	4.53(-4)	1.25(-2)	-	2.29(-3)	-
Ce-144	-	6.49(-1)	5.25(-4)	1.44(-2)	-	1.30(-2)	-
Pr-143	-	-	1.59(-4)	-	-	-	-
Pr-144	-	6.49(-1)	5.25(-5)	1.44(-2)	-	1.30(-2)	-
Cr-51	6.38(1)	7.75(-1)	3.24(-3)	9.73(-2)	2.88(0)	1.76(-2)	-
Mn-54	4.75(1)	7.61(-1)	2.71(-2)	7.49(-1)	2.26(0)	1.69(-2)	-
Fe-59	1.14(1)	1.58(-1)	4.34(-4)	1.26(-2)	5.25(-1)	3.58(-3)	-
Co-58	5.71(2)	8.29(0)	1.54(-2)	4.39(-1)	2.76(1)	1.88(-1)	-
Co-60	2.56(2)	4.80(0)	1.86(-2)	7.55(-2)	1.27(1)	1.06(-1)	-
Rh-106	-	2.65(-1)	1.75(-4)	-	-	5.29(-3)	-
Te-127m	-	2.56(0)	3.37(-3)	9.34(-2)	2.56(-2)	5.12(-2)	-
Te-127	-	2.57(0)	3.38(-3)	9.37(-2)	2.57(-2)	5.14(-2)	-
Rh-103m	-	1.43(-1)	4.42(-4)	1.29(-2)	1.99(-2)	2.85(-3)	-
TOTAL	7.03(4)	1.29(4)	1.58(1)	2.93(2)	2.77(3)	3.61(2)	7.6(-1)

a. Based on data obtained from reference 1 (a total activity of 7.6(-1) Ci/year from dry and compacted wastes). Major isotopic contributions are from Cr-51, Co-58, Co-60, Mn-54, Cs-134, and Cs-137.

TABLE 11.4.2-7

SUMMARY OF BACKFLUSHABLE FILTERS

Filter	System Process Filtered / Stream	Number of Filters Per Unit	Location	FSAR Section in Which Filtered System is Described
Seal water return	CVCS / Reactor coolant pump seals	1	Auxiliary building level B	Paragraph 9.3.4.1
Recycle evaporator feed	BRS / Recycle evaporator feed demineralizer	2 (common)	Auxiliary building level B	Paragraph 9.3.4.2
Waste evaporator feed	WPSL / Waste holdup tank (downstream of pumps)	1	Auxiliary building level B	Subsection 11.2.2
Resin sluice	WPSL / Spent resin storage tank (downstream of pumps)	1	Auxiliary building level B	Subsection 11.2.2
Floor drain tank	WPSL / Floor drain tank (downstream of pumps)	1	Auxiliary building level B	Subsection 11.2.2
Waste monitor tank	WPSL / Waste monitor tank demineralizer	1	Auxiliary building level D	Subsection 11.2.2
Steam generator blowdown	SGBS / Steam generator blow down trim heat exchanger	1	Auxiliary building level B	Paragraph 10.4.8.2

11.5 PROCESS AND EFFLUENT RADIOLOGICAL MONITORING AND SAMPLING SYSTEMS

The process and effluent radiation monitoring systems (PERMS) monitor radiation levels in selected plant process systems and in plant effluents. A description of the area radiation monitoring system is found in subsection 12.3.4.

The PERMS operates in conjunction with regular and special radiation surveys and with a chemical and radiological sampling program performed by the plant staff to provide timely, sufficient information for continued safe plant operation.

11.5.1 DESIGN BASES

The PERMS is designed to:

A. Provide equipment and criteria to ensure plant performance which meets applicable regulatory requirements for normal operation and for transient situations that might be expected to occur.

B. Provide support data to health physics personnel to aid them in limiting release of activity to the environment and limiting exposure to operation and maintenance personnel to as low as reasonably achievable.

C. Provide early warning of a system or equipment malfunction that could lead to a health hazard or plant damage.

D. Provide a reliable and usable tool to monitor plant-related radioactivity.

E. Provide support to the plant for compliance with applicable Nuclear Regulatory Commission guidelines including:

   1. 10 CFR 50, Appendix I.
   2. 10 CFR 20.
   3. Regulatory Guide 1.21.
   4. Regulatory Guide 8.8.
   5. Regulatory Guide 1.45.
   6. General Design Criteria 60, 63, and 64.
   7. Regulatory Guide 1.97.
   8. ANSI N13.1-1969.

The scope of PERMS regarding radiological monitoring following accidents as required by General Design Criterion 64 and Regulatory Guide 1.97 is limited to releases associated with the postulated design basis accidents considered in chapter 15. Post accident radiation monitoring is discussed in subsection 11.5.5. Although PERMS is generally a surveillance system, there are certain monitors that perform a safety function. Such monitors meet the normal requirements for safety grade equipment. For safety-related monitors, redundancy is relied upon to maintain surveillance in case of a single failure.

11.5.2 SYSTEM DESCRIPTION

11.5.2.1 Digital Radiation Monitoring System

The process and effluent radiation monitoring system (PERMS) is based on a distributed digital microprocessor approach, where each radiation monitor is self-contained and consists of a detector and a data processing module (DPM) that contains the microprocessor. The DPM is the basic control center for each PERMS channel. Each complete channel consists of a monitor with power supply and preamplifier; a local dedicated data processing module containing a digital buffer and microprocessor with alarm outputs; and access to readout modules (cathode ray tube (CRT) and printer).

The PERMS collects and displays all the information available from the field-mounted detectors on a CRT and hardcopy printer on demand. This is accomplished by either the communications display computer (CDC) or the safety-related display console (SRDC) located in the control room. The CDC receives data from the DPMs (both standard and safety-related) and interfaces with the PERMS display computer (PDC) and, via a boundary isolation device (BID), the integrated plant computer (IPC) (figure 11.5.2-1). The SRDC displays data received from the safety-related channels which are composed of safety-related Class 1E components (figure 11.5.2-2). The safety-related DPMs also interface with the CDC with one-way data transmission. The CDC is shown in figure 11.5.2-3.

11.5.2.2 Monitor Description

The radiation monitors fall into five different functional classifications:

A. Process monitors, which determine concentrations of radioactive material in plant fluid systems. The primary-to-secondary leak detection monitors (N16 and noble gas leak rate detectors) are also included in this category.

B. Effluent monitors, which measure radioactivity discharged to the environs.

C. Airborne monitors, which provide operator information on airborne concentrations of radioactive gases and particulate radioactivity at various points in the ventilation system ducts. (See figure 11.5.2-4.)

D. Area monitors, which provide operator information on external gamma radiation levels at fixed points throughout the plant.

E. Post-accident (or high range) monitors, designed to assess and follow potential pathways for release of radioactive materials during accident conditions.

Table 7.5.4-2 provides cross-references of radiation monitors' conformance with NUREG-0737 guidance. Table 11.5.2-1 gives the design parameters for the PERMS. Table 11.5.2-2 lists the detectors used in the PERMS. Table 11.5.2-3 lists the conditions of service for PERMS. Figures 11.5.2-5 and 11.5.2-6 give the monitoring locations for liquid and gaseous release pathways, and table 11.5.2-5 gives the design parameters for flowrate monitoring. Drawings AX6DD200, AX6DD204, AX6DD205, AX6DD206, AX6DD207, AX6DD209, AX6DD211, AX6DD215, 1X6DD201, 1X6DD202, 2X6DD201, 2X6DD202, AX6DD218, AX6DD221, AX6DD223, AX6DD225 and AX6DD226 show the detector locations.

11.5.2.3 Monitor Functions

Most of the PERMS radiation monitors are centrally (by unit) monitored as part of the digital radiation monitoring system (DRMS). The 112 PERMS channels installed at the VEGP site consist of:

A. Twenty-four area monitors, described in subsection 12.3.4, 17 DRMS (8 per unit and 1 shared), and 7 shared non-DRMS monitors;

B. Sixty process and effluent channels, described in subsection 11.5.2 and table 11.5.2-1 are allocated as follows:

   a. Active Channels: Unit 1-25, Unit 2-26, and 6 shared.

   b. Passive Channels: Unit 1-1, Unit 2-1, and 0 shared.

   c. Inactive Channels: 1 shared non-DRMS.

C. Twenty-eight post-accident monitor DRMS channels, described in subsection 11.5.5, (10 active and 4 passive activity collection per unit).

Some of these monitors are located in areas of the site and serve functions that are common to both units. These shared monitors are identified by an "A" in the front of the monitor's tag number (i.e., "ARE - --------").

The following is a description of each monitor:

A. Airborne Process and Effluent Monitors

   1. RE-12442A - Plant Vent Effluent Air Particulate Monitor (Low Range)

      The purpose of this monitor is to continuously monitor air particulate activity as it is discharged to the environment through the main plant vent and provide data necessary to ensure that particulate activity releases from the plant vent are below specified limits.

      The monitor also serves as a backup to the containment vent particulate monitor RE-2565A. Since the main plant vent discharges directly to the atmosphere, the data from this monitor are most representative of particulate activity releases to the plant environs. Should the activity exceed a specified level, a high radiation level alarm will be annunciated to indicate that an increase in particulate release has occurred. The specific source of release within the plant may be determined through analysis of the monitors upstream of the vent monitor and/or with portable monitoring devices.

   2. RE-12442B - Plant Vent Effluent Iodine Monitor (Low Range)

      The purpose of this monitor is to continuously monitor iodine activity as it is discharged to the environment through the main plant vent and provide data necessary to ensure that gaseous iodine activity releases from the plant vent are below specified limits.

      The monitor also serves as a backup to the containment vent radioiodine monitor RE-2565B. Since the main plant vent discharges directly to the atmosphere, the data from this monitor are most representative of the iodine activity releases to the plant environs. Should the activity exceed a specified level, a high radiation level alarm will be annunciated to indicate that an increase in gaseous iodine release has occurred. The specific source of release within the plant may be determined through analysis of the monitors upstream of the vent monitor and/or with portable monitoring devices.

3. RE-12442C - Plant Vent Effluent Radiogas Monitor (Low Range)

   The purpose of this monitor is to continuously monitor the gaseous activity as it is discharged to the environment through the main plant vent and provide data necessary to ensure that the gaseous activity releases from the plant vent are below specified limits.

   The monitor also serves as a backup to other monitored streams that are routed to the main plant vent, including the waste gas processing system effluent ARE-0014, fuel building ARE-2532A and B and ARE-2533A and B, and containment RE-2565C gas monitors. Since the main plant vent discharges directly to the atmosphere, the data from this monitor are most representative of gaseous activity releases to the plant environs. Should the activity exceed a specified level, a high radiation level alarm will be annunciated to indicate that an unexpected increase in gaseous release has occurred. The specific source of release within the plant may be determined through analysis of the monitors upstream of the vent monitor and/or with portable monitoring devices.

4. RE-2565A - Containment Vent Effluent Air Particulate Monitor

   The purpose of this monitor is to measure air particulate radioactivity in the containment purge vent and provide data necessary to ensure that the release rate through the containment vent during purging is below specified limits.

   The containment purge flow is filtered and routed through the main plant vent, where activity is again monitored by the plant vent air particulate monitor RE-12442A and then discharged to the atmosphere.

   This monitor initiates automatic closure of the containment purge supply and exhaust duct valves for high radiation levels.

5. RE-2565B - Containment Vent Effluent Iodine Monitor

   The purpose of this monitor is to measure gaseous iodine activity in the containment purge vent and provide data necessary to ensure that the release rate through the containment vent during purging is below specified limits.

   The containment purge flow is filtered and routed through the main plant vent, where activity is again monitored by the plant vent radioiodine monitor RE-12442B and then discharged to the atmosphere.

   This monitor initiates automatic closure of the containment purge supply and exhaust duct valves for high radiation levels.

6. RE-2565C - Containment Vent Effluent Radiogas Monitor

   The purpose of this monitor is to measure gaseous activity in the containment purge vent and provide data necessary to ensure that the release rate through the containment vent during purging is below specified limits.

   The containment purge flow is routed through the main plant vent, where the activity is again monitored by the plant vent gas monitor RE-12442C and then discharged to the atmosphere.

   This monitor initiates automatic closure of the containment purge supply and exhaust duct valves for high radiation levels.

7. RE-2562A - Containment Atmosphere Process Air Particulate Monitor

   The purpose of this monitor is to measure air particulate radioactivity in the containment atmosphere and provide information for determining allowable personnel access to the containment.

   The monitor can also be utilized to provide an indication of reactor coolant system leakage as part of the reactor coolant leak detection system. An intermediate radiation alarm alerts the operator that a predetermined activity level in the containment has been exceeded.

8. RE-2562B - Containment Atmosphere Iodine Cartridge

   The purpose of this monitor is to measure airborne iodine radioactivity in the containment atmosphere and provide information for determining allowable personnel access to the containment.

   The cartridge may also be used to provide an indication of reactor coolant system leakage.

9. RE-2562C - Containment Atmosphere Process Radiogas Monitor

   The purpose of this monitor is to measure gaseous radioactivity in the containment atmosphere and provide information for determining allowable personnel access to the containment.

   The monitor can also be utilized to provide an indication of reactor coolant system leakage as part of the reactor coolant leak detection system. An intermediate radiation alarm alerts the operator that a predetermined activity level in the containment has been exceeded.

10. Deleted

11. Deleted

12. Deleted

13. Deleted

14. Deleted

15. RE-12116 and RE-12117 - Control Room Air Intake Process Radiogas Monitor

    The purpose of this monitor is to continuously monitor the control room supply air for gaseous activity that could be present in the atmosphere following an accident.

    A high alarm signal will generate a control room isolation (CRI) signal which will automatically switch the control room ventilation system from the normal operating mode to the accident mode using safety grade filtration trains. In addition, once the CRI has been initiated, a signal to the onsite technical support center ventilation is provided to switch to its filtration mode.

16. RE-0013 - Waste Gas Processing System Process Radiogas Monitor

    The purpose of this monitor is to continuously monitor the gaseous activity of the waste gas processing system recirculation stream and provide data used to determine the radioactive inventory of the waste gas decay tanks and/or waste gas shutdown tanks.

17. Deleted

18. Deleted

19. Deleted

20. ARE-0014 - Waste Gas Processing System Effluent Radiogas Monitor

    The purpose of this monitor is to continuously monitor the gaseous activity discharged from the waste gas processing system and provide data necessary to ensure that gaseous activity releases are below specified levels.

    The activity is routed to the Unit 1 auxiliary building exhaust (and again monitored by RE-0039A) and, in turn, to the Unit 1 plant vent (and again monitored by RE-12442C).

    A high radiation alarm signal isolates the waste gas processing system discharge line.

21. RE-0039A, RE-0039B, and RE-0039C - Waste Gas Decay Tank and Compressor Area Ventilation Effluent Radiogas Monitor

    The purpose of this monitor is to continuously monitor the ventilation exhaust from the waste gas decay tank area (RE-0039A), the waste gas compressor areas along with the catalytic recombiner room in Unit 1 (RE-0039B), and catalytic recombiner room in Unit 2 (RE-0039C).

    A high radiation alarm indicates leakage in the waste gas processing system or catalytic recombiner areas being monitored.

22. ARE-2532A and B and ARE-2533A and B - Fuel Handling Building Effluent Radiogas Monitors

The purpose of this monitor is to continuously monitor the gaseous activity discharged from the fuel handling building and provide data necessary to ensure that gaseous activity releases are below specified limits.

The activity is filtered and routed to the plant vent, where it is again monitored by the plant vent monitors RE-12442A, B, and C prior to discharge to the atmosphere. In the event of a fuel handling accident inside the fuel handling building, ARE-2532 and ARE-2533 are capable of detecting the high activity in order to switch the fuel handling building ventilation system from the normal operating mode to the accident mode, which includes safety-related filtration units. (See subsection 9.4.2.)

23. Deleted.

24. Deleted.

25. ARE-16980 – Radwaste Processing Facility Vent Air Particulate Monitor

    The purpose of this monitor is to continuously monitor air particulate activity as it is discharged through the radwaste processing facility ventilation exhaust system to the atmosphere and to provide data to ensure that activity releases from the facility are below specified limits. This monitor is part of the PERMS system.

    Should activity exceed a specified level, a high radiation level alarm will be annunciated in the radwaste processing facility local control room and in the Unit 1 main control room to indicate that an increase in particulate release has occurred.

B. Liquid Process and Effluent Monitors

1. RE-810 - Steam Generator Primary-to-Secondary Noble Gas Leak Detection Monitor

   The purpose of this monitor is to monitor the condenser steam jet air ejector header to detect noble gases and determine combined leak rate (contributed by all four steam generators) and rate of change in primary-to-secondary leaks. Remote readout is provided at the integrated plant computer (IPC) and the Communications Console PERMS viewer application in the control room.

   The noble gas monitor, RE-810, is a stand-alone monitor that sends an output signal to the Communications Console which sends the information to the IPC via the BID. The IPC converts this signal to provide readout of primary-to-secondary leak rate in gallons per day (GPD), as well as rate of change (GPD/h). RE-810 monitor parameters and alarm setpoints are maintained locally at the monitor and are also changeable via the PERMS Communications Console or the PERMS display console (PDC). Parameters for GPD leak rate, GPD/h calculations, and alarm setpoints for both calculated values are maintained on the IPC.

2. RE-0017A and B - Component Cooling Water Process Monitor

The purpose of this monitor is to continuously monitor the component cooling loop for activity indicative of a leak of radioactive water into the component cooling system.

3. RE-0018 - Waste Liquid Effluent Monitor

   The purpose of this monitor is to monitor waste disposal system liquid releases from the plant and provide data necessary to ensure that liquid waste activity discharges are below specified limits.

   A high alarm signal initiates automatic valve closure on the liquid waste discharge line to prevent further activity release. Remote readout is provided at the liquid section of the waste processing system control panel.

4. RE-0019 - Steam Generator Sample Liquid Process Monitor

   The purpose of this monitor is to monitor the liquid phase of the secondary side of the steam generators for activity (which would be indicative of primary to secondary system leaks) and to provide backup information to the air ejector gas monitor RE-12839C.

   Remote readout and A/V alarm are provided at the isolation valve controls in the secondary side of sampling area.

5. RE-724 - Steam Generator Primary-to-Secondary N16 Leak Detector Monitor

   The purpose of this monitor is to monitor the main steam lines to detect N16 isotope and determine combined leak rate (contributed by all four steam generators) and rate of change in primary-to-secondary leaks. Remote readout is provided at the integrated plant computer (IPC).

   The N-16 monitor, RE-724, is a stand-alone monitor which sends an output signal to the IPC. The IPC converts this signal to provide readout of primary-to-secondary leak rate in gallons per day (GPD), as well as rate of change (GPD/h). RE-724 monitor parameters and alarm setpoints are maintained locally at the monitor. Parameters for GPD leakrate, GPD/h calculations and alarm setpoints for both calculated values are maintained on the IPC.

6. RE-0020A and B - Nuclear Service Water Process Monitor

   The purpose of this monitor is to continuously monitor the nuclear service water loop for activity and to prevent otherwise undetected activity release from the service water system, which represents a potential release path to the environment.

   During normal operation, these monitors will indicate one of the following conditions:

   a. Leak from equipment processing radioactive water into nuclear service cooling water.

b. Leak from equipment processing radioactive liquid into auxiliary component cooling water system coincident with leak to nuclear service cooling water.

c. Leak from equipment processing radioactive liquid into component cooling water system coincident with leaks to nuclear service cooling water.

7. RE-0021 - Steam Generator Blowdown Liquid Process Monitor

   The purpose of this monitor is to monitor the steam generator blowdown liquid for activity in the event of primary to secondary system leakage and provide data necessary to ensure that the steam generator blowdown processing system discharges to the environment are below desired limits.

   This monitor also provides indication of a steam generator blowdown processing system malfunction.

   A high alarm signal automatically closes the steam generator blowdown processing system isolation valves and discharge lines. Remote readout is provided at the steam generator blowdown system panel.

8. RE-0848 - Turbine Building Drain Liquid Effluent Monitor

   The purpose of this monitor is to continuously monitor the secondary side liquid drains for activity released in the event of primary to secondary system leakage.

   A high alarm signal stops the flow from the turbine building drain system to the waste water retention basin. Turbine building sump pump discharges are diverted to the "dirty" turbine building drain tank. Discharges from the "clean" turbine building drain tank are recirculated back to the clean tank.

9. RE-1950 - Auxiliary Component Cooling Water Process Monitor

   The purpose of this monitor is to continuously monitor the auxiliary component cooling loop for activity indicative of a leak of contaminated water into the auxiliary component cooling system.

10. RE-48000 - Chemical and Volume Control System Letdown Monitor

   The purpose of this monitor is to monitor the chemical and volume control system letdown liquid process and provide indication of abnormal activity levels in the reactor coolant system.

11.5.2.4 Alarm Setpoints

Each of the PERMS airborne and liquid monitors (excluding passive cartridge monitors RE-2562B, RE-12839A and B, and RE-12444A, B, F, and G) has two alarm setpoints; i.e., intermediate and high. When the radiation level being monitored reaches the intermediate setpoint, a visual indication alerts plant personnel of the monitor reading. If the high setpoint is reached, an alarm will be annunciated. The high alarm setpoint also triggers the control function for those monitors so equipped: gas monitors RE-2565A, B, and C, RE-12839C, RE-12116, RE-12117, ARE-0014, ARE-2532A and B, and ARE-2533A and B; and liquid monitors RE-0018, RE-0021, and RE-0848.

The primary-to-secondary leak detection (N16) monitor, RE-724, setpoints and the primary-to-secondary noble gas leak detection monitor, RE-810, leak rate and leak rate change setpoints are operator controlled through the IPC interface. The RE-810 radiation setpoint is operator controlled through the PERMS Comm Console. Variable operator-selected leak rate gallons per day (GPD) and leak rate changes (GPD/h) are provided in the main control room.

Setpoints are controlled by plant procedures and the Offsite Dose Calculational Manual (ODCM) where appropriate.

11.5.2.5 Inservice Inspection, Calibration, and Maintenance

The operability of each digital radiation monitoring system (DRMS) safety-related radiation detector/channel, except for RE-0005 and RE-0006, of the PERMS, PAMS, and ARMS is checked frequently with a source positioned remotely from the SRDC in the main control room or the DPM, or manually at the monitor. The operability of each DRMS nonsafety-related radiation detector/channel, except for RE-48000, of the PERMS, PAMS, and ARMS is checked frequently with a source positioned remotely from the communications console in the main control room, the DPM, or the chemistry PERMS station, or manually at the monitor.

Channel checks are routinely done to verify parameters and alarm conditions. Electronic testing means are used to calibrate instrument subcomponents for accurate range, setpoint, and indication periodically. The PERMS, PAMS, and ARMS radiation monitors are functionally tested periodically. This periodic test is used to verify the operability of alarms, automatic valves, and dampers, if applicable. This is performed by lowering the setpoint below background; introducing an external electronic input signal or other testing means as necessary to create an alarm/actuation condition; and verifying the appropriate response of alarms, valves, and dampers.

Calibration of all PERMS, PAMS, and ARMS digital radiation monitoring system (DRMS) detectors is normally performed every 18 months. Generally this procedure is performed on all but containment area monitors during an administratively selected period of the fuel cycle other than refueling to minimize the effect on the plant and to limit occupational exposure. Calibration of non-DRMS detectors/monitors (radwaste processing facility ARMS, and N16 and noble gas primary-to-secondary leak detection) is normally performed every 18 months with periodic testing under separate procedural controls from the DRMS. Calibration of all PERMS, PAMS, and ARMS digital radiation monitoring system (DRMS) detectors is performed at an interval in accordance with the Surveillance Frequency Control Program.

Calibration of all PERMS and PAMS detectors is performed using sources that allow indication of low and mid range. Calibration of low range ARMS monitors is performed using a source which will allow indication in a low, mid, and high range. Calibration of high range ARMS monitors is performed in accordance with manufacturer’s recommendations to prevent undesirable exposure to plant personnel.

An initial calibration was performed on PERMS, PAMS, and ARMS monitors with the use of NIST traceable standards of known isotopic concentration. Secondary sealed sources with fixed geometrical configurations were tested to determine relationship to the initial calibration standards. These secondary calibration sources or National Institute of Science and

Technology (NIST) traceable sources are used for periodic onsite recalibration to ensure that the response of each detector is in agreement with the response as determined in the original initial "type" calibration.

Provisions for minimizing sample deposition in liquid effluent radiation monitors through electropolishing, decontaminating, and flushing sample chambers have been made.

11.5.3 EFFLUENT MONITORING AND SAMPLING

11.5.3.1 Plant Sampling System

The plant sampling system is described below.

11.5.3.2 Sampling Requirements for Process and Effluent Radiation Monitoring System (PERMS)

The primary means of quantitatively evaluating the isotopic activity levels in process and effluent paths is a program of sampling and laboratory measurements. Gross activity measurements, as provided by the process and effluent monitors, are generally not acceptable for showing compliance with effluent release limits. However, the continuous gross monitors (both process and effluent) can be calibrated by normalizing against the results of specific radionuclide analysis, and in this way the gross measurement from the continuous monitors may be used to determine the isotopic activity concentration in process streams and the total isotopic activity released in effluent paths. Tables 11.5.3-1 and 11.5.3-2 present information for the sampling system associated with the PERMS monitors, for both gaseous and liquid monitors. These tables include the following information for each PERMS monitor:

- Sample identification (including fluid type, sample point location, and sample type).
- Purpose.

Sample points are located at the PERMS monitor location unless a sample point that will serve to calibrate the monitor already exists in the system.

The requirements of General Design Criterion 64, Monitoring Radioactivity Releases, are satisfied by this sampling program and the associated radiological monitors described in subsection 11.5.2. VEGP will conform to Regulatory Guide 4.15, Quality Assurance for Radiological Monitoring Programs (Normal Operations) - Effluent Streams and Environment.

11.5.4 PROCESS MONITORING AND SAMPLING

The requirements of General Design Criterion 60, Control of Releases of Radioactive Materials to the Environment, have been addressed. Monitors used to initiate automatic closure of isolation valves in liquid and gaseous systems are described in subsection 11.5.2.

The requirements of General Design Criteria 63, Monitoring Fuel and Waste Storage, have been addressed. Radiation monitors used in the radioactive waste processing systems are discussed in subsection 11.5.2.

11.5.5 POST-ACCIDENT RADIATION MONITORING

In addition to the process and effluent radiation monitoring system (PERMS) described in subsection 11.5.2, radiological monitors are also included in the post-accident monitoring system (PAMS) as required by General Design Criterion 64 and Regulatory Guide 1.97. The function of the radiation monitors in this system is to detect, compute, and indicate the radiation levels at selected plant locations and to actuate alarms should these levels exceed predetermined values. Thus, the control room operating staff is provided with the information on radiation levels needed to help assess and maintain the safety of the plant and its environs following the occurrence of any design basis accident.

The PAMS radiation monitors supplement the PERMS and area radiation monitoring system (ARMS) (subsections 11.5.2 and 12.3.4, respectively) and consist of 32 DRMS monitors (10 active and 6 passive monitors per unit). The following is a description of each monitor:

A. RE-12444A and RE-12444F - Plant Vent Air Particulate Sampler

The purpose of this monitor is to continuously sample air particulate activity as it is discharged to the environment through the main plant vent. The sample filter is measured to determine that particulate activity releases from the plant vent are below specified limits.

B. RE-12444B and RE-12444G - Plant Vent Effluent Iodine Sampler

The purpose of this monitor is to sample airborne iodine radioactivity in the plant vent.

C. RE-12444C,D,E - Plant Vent Radiogas Monitor

The purpose of these three monitors (normal, mid, and high range) is to continuously monitor the gaseous activity as it is discharged to the environment through the main plant vent and provide data necessary to ensure that the gaseous activity releases from the plant vent are below specified limits.

D. RE-13119, RE-13120, RE-13121, and RE-13122 - Main Steam Line Monitors

The purpose of this monitor is to detect steam generator tube leakage and to monitor this potential effluent release path. One detector is used on the main steam line of each steam generator upstream of the main steam line isolation and atmospheric dump valves.

E. RE-12839A - Condenser Air Ejector and Steam Packing Exhauster Passive Air Particulate Monitor

The purpose of this monitor is to continuously measure particulate activity in the discharge from the air ejector exhaust header of the condensers (which is indicative of a primary to secondary system leak) and steam packing exhauster monitor effluent path and to estimate magnitude of release of radioactivity.

F. RE-12839B - Condenser Air Ejector and Steam Packing Exhauster Passive Iodine Cartridge

The purpose of this monitor is to continuously measure iodine activity in the discharge from the air ejector exhaust header of the condensers (which is indicative of a primary to secondary system leak) and steam packing exhauster monitor effluent path and to estimate magnitude of release of radioactivity.

G. RE-12839C,D,E - Condenser Air Ejector and Steam Packing Exhauster Radiogas Monitor

The purpose of this monitor is to continuously measure gaseous activity in the discharge from the air ejector exhaust header of the condensers (which is indicative of a primary to secondary system leak) and steam packing exhauster effluent path and to estimate magnitude of release of radioactivity.

A high radiation alarm signal from this monitor will automatically divert the air ejector discharge flow through charcoal and high efficiency particulate air filtration banks.

Table 11.5.5-1 gives the design parameters for the PAMS radiation monitors. The plant vent and the condenser air ejector and steam packing exhauster radiogas monitors consist of multiple detectors, each to cover the required range. Table 11.5.5-2 presents the detector requirements for the PAMS radiation monitors. Table 11.5.5-3 lists the conditions of service for the PAMS radiation monitors.

Sampling requirements for the PAMS radiation monitors (where applicable) are included with the PERMS on table 11.5.3-1.

Alarm setpoints are controlled by plant procedures and the Offsite Dose Calculation Manual (ODCM) where appropriate.

Inservice inspection, calibration, and maintenance of PAMS radiation monitors are described in paragraph 11.5.2.5.

Some monitors described in this section are utilized for normal operations as well as for accident conditions. Likewise, some of the area radiation monitors described in subsection 12.3.4 are also used for post-accident monitoring. The grouping presented in this report, recognizing that some of these monitors are multifunctional, was so chosen to maintain clarity while avoiding duplication.

11.5.5.1 Potential Gaseous Accident Release Pathways

The potential gaseous accident release pathways are from:

A. The plant vent which includes discharges from the containment purge system, the auxiliary building HVAC system (which includes discharges from the gaseous radwaste system and the containment piping penetration area filter and exhaust system) and the fuel handling building HVAC.

B. The condenser air ejector and steam packing exhauster system.

C. The steam generator safety relief valves and atmospheric dump valves.

D. The auxiliary feedwater steam turbine exhaust vent.

E. The steam generator blowdown line break blowout panel.

The plant vent, condenser air ejector and steam packing exhauster, and main steam line monitor's capability and flowrate monitoring equipment are summarized in tables 11.5.5-1 through 11.5.5-3. The steam generator liquid monitor capability and flowrate monitoring equipment is summarized in tables 11.5.2-1 through 11.5.2-3 and 11.5.2-5.

The main steam line monitors are used to estimate the releases from the actuation of steam generator secondary safety relief valves, atmospheric steam dump valves, and the auxiliary feedwater steam turbine exhaust. The main steam line monitors are externally mounted Geiger-

Mueller tubes viewing the main steam line upstream of the safety valves, atmospheric dump valves, and steam supply line to the auxiliary feedwater steam turbine. The main steam line monitors are calibrated to account for the pipe thickness of the main steam line.

The activity release rate or integrated release from the steam generator safety valves and/or atmospheric dump valves during an accident may be estimated by using the appropriate readings from the main steam line monitor and the main steam line flowrate and pressure. (Main steam line flowrate and pressure instrumentation is discussed in section 7.5.) The activity release rate from the auxiliary feedwater turbine exhaust may be estimated by using the appropriate reading from the main steam line monitor and the flowrate determined from a heat balance on the auxiliary feedwater turbine.

To prevent overpressurization of the auxiliary building from a steam generator blowdown line break outside containment, a vent pathway from the auxiliary building to the atmosphere is provided with a blowout panel that is designed to open at 1 psid. Steam generator blowdown is automatically terminated if the temperature environment in the affected area or the flowrate through the line exceeds a predetermined setpoint, thereby limiting the actual release. An estimate of the activity released to the environment can be determined from the recorded readings from the steam generator liquid monitor and the appropriate steam generator blowdown line flow instrumentation. The flow instrumentation for each steam generator connection has a flowrate range of 0 to 250 gal/min.

11.5.5.2 Standard Review Plan Evaluation

Standard Review Plan (SRP) 11.5 states that the gaseous and liquid process streams or effluent release points should be monitored and sampled according to tables 1 and 2.

Although monitoring and sampling are not provided for all of the process systems specifically listed in tables 1 and 2 of SRP 11.5, all effluent streams are monitored and sampled prior to discharge from the plant. A footnote to tables 1 and 2 of SRP 11.5 states that provisions indicated within parentheses are required only for systems not monitored, sampled, or analyzed (as indicated) prior to release by downstream provisions. The process and effluent radiological monitoring instrumentation and sampling systems provided are adequate for compliance with applicable regulations and therefore meet the intent of the SRP. Also, the provisions made for post-accident monitoring and sampling meet the intent of SRP 11.5 requirements.

TABLE 11.5.2-1 (SHEET 1 OF 4)

DESIGN PARAMETERS FOR PROCESS AND EFFLUENT RADIATION MONITORS

Monitor	Monitor Type	Safety Classification	Mechanical Code	Seismic Qualification	Number of Channels per Unit	Pump	Process Flow (ft^3/min)	Automatic Control Function	Duct Size/Pipe Diameter Material, Schedule	
Gas										
Plant Vent Effluent (Low Range)										
RE-12442A (particulate)	Effluent, offline	NNS[(a)]	B 31.1	Sample probe and flowmeter only	1	X	150,000 (Unit 1) 90,000 (Unit 2)	None	120 in. x 60 in. duct (Unit 1) 60 in. x 78in. (Unit 2) ASTM[(b)] 526, 12 gauge	Non-Class 1E diesel backed
RE-12442B (iodine)	Effluent, offline	NNS	B 31.1		1	X				
RE-12442C (radiogas)	Effluent, offline	NNS	B 31.1		1	X				
Containment Vent Effluent										
RE-2565A (particulate)	Effluent, offline	NNS	B 31.1	No	1	X	5000/15,000	Yes; closes containment purge supply and exhaust ducts	32-in.-dia. ASTM A527, 14 gauge	Non-Class 1E diesel backed
RE-2565B (iodine)	Effluent, offline	NNS	B 31.1	No	1	X				
RE-2565C (radiogas)	Effluent, offline	NNS	B 31.1	No	1	X				
Containment Atmosphere										
RE-2562A (particulate)	Airborne, offline	NNS	B 31.1	Yes	1	X	NA	None	1-in.-dia. ASTM SA-312, -GR, TP 304L, 40S	Non-Class 1E diesel backed
RE-2562B (iodine)	Airborne, offline passive iodine cartridge	NNS	B 31.1	Yes	1	X	NA	None		
RE-2562C (radiogas)	Airborne, offline	NNS	B 31.1	Yes	1	X				

TABLE 11.5.2-1 (SHEET 2 OF 4)

Monitor	Monitor Type	Safety Classification	Mechanical Code	Seismic Qualification	Number of Channels per Unit	Pump	Process Flow (ft^3/min)	Automatic Control Function	Duct Size/Pipe Diameter Material, Schedule	Power Supply
Control Room Intake RE-12116 RE-12117	Airborne, inline	SC-3/1E	NA	Yes	2		3000	Yes; switches control room ventilation to safety grade filtration train and initiates TSC filtration train	60 in. x 48 in., duct ASTM A527, 14 gauge	Class 1E
Waste Gas Processing System RE-0013	Process, inline	NNS	ASME(c) III-3	Yes	1		40-50 nitrogen	None	2-in. dia. ASME SA-106, GR. B, 80	Non-Class1E
Waste Gas Processing System Effluent ARE-0014	Process, inline	NNS	ASME III-3	Yes	1 Common		2 nitrogen	Yes; isolates WGPS discharge	2-in. dia. SA-106 GR. B 80	Non-Class 1E
Waste Gas Decay Tank RE-0039A and B (C for Unit 2 only)	Process, inline	NNS	NA	No	2 (Unit 1) 3 (Unit 2)		500/12000 (Unit 1) 500/1100/800 (Unit 2)	None	16 in. x 16 in. ASTM A527, 16 gauge	Non-Class 1E
ARE-2532A and B ARE-2533A and B	Effluent, inline	SC-3/1E	NA	Yes	4 Common		12,700	Yes; switches building to accident mode of operation including safety-related filtration units	36 in. x 36 in. ASTM A527, 18 gauge	Class 1E
Primary-to-Secondary N16 Leak Detection RE-724	Process, offline	NNS	NA	No	1		Variable	None		Non-Class 1E

TABLE 11.5.2-1 (SHEET 3 OF 4)

Monitor	Monitor Type	Safety Classification	Mechanical Code	Seismic Qualification	Number of Channels per Unit	Pump	Process Flow (ft^3/min)	Automatic Control Function	Duct Size/Pipe Diameter Material, Schedule	Power Supply
Primary-to-Secondary Noble Gas Leak Detection RE-810	Process, offline	NNS	B 31.1	No	1		Variable	None	18-in. dia. ASTM A-106-GRB std wt	Non-Class 1E
Radwaste Processing Facility Vent ARE-16980 (particulate)	Effluent, offline	NNS	B 31.1	No	1 Common	X	16000	None	42 in., ASTM A-653, 14 gauge	Non-Class 1E
Liquid										
Component Cooling Water RE-0017A and B	Process, offline	NNS	ASME III-3	Yes	2		9000	None	20-in. dia., ASME SA-106 GR. B std wt	Non-Class 1E
Waste Liquid Effluent RE-0018	Effluent, offline	NNS	B 31.1	No	1		100		Initiates automatic valve closure of liquid waste discharge line	2-in. dia.,Non-Class 1E ASTM A-312 GR. TP 394L, 40S
Steam Generator Liquid RE-0019	Process, offline	NNS	B 31.1	No	1		360	None	4-in. dia ASTM A-106 GR. B, 40	Non-Class 1E
Nuclear Service Water RE-0020A and B	Process, offline	NNS	ASME III-3	Yes	2		14,000-16,000(A) 16,000 18,000(B)	None	24-in. dia., ASME SA-312 GR. TP 304L, 10	Non-Class 1E

TABLE 11.5.2-1 (SHEET 4 OF 4)

Monitor	Monitor Type	Safety Classification	Mechanical Code	Seismic Qualification	Number of Channels per Unit	Pump	Process Flow (ft^3/min)	Automatic Control Function	Duct Size/Pipe Diameter Material, Schedule	Power Supply
Steam Generator Blowdown RE-0021	Effluent, offline	NNS	B 31.1	No	1		360	Closes steam generator blowdown processing system isolation valves and discharge lines	4-in. dia., ASTM A-106 GR. B, 40	Non-Class 1E
Turbine Building Drain RE-0848	Effluent, offline	NNS	B 31.1	No	1		50-640	Stops flow from turbine and control bldg drains system to the waste water retention basin	6-in. dia., ASTM A-106 GR. B, 40	Non-Class 1E diesel backed
Auxiliary Component Cooling Water RE-1950	Process, offline	NNS	ASME III-3	Yes	1		5400 6600	None	16-in. dia., ASME SA-106 GR. B, 80	Non-Class1E
Chemical and Volume Control System Letdown RE-48000	Process, offline	NNS	ASME III-3	Yes	1		75-120	None	3-in. dia., ASME SA-376 TP 304 or 316SS Schedule 160	Non-Class1E

a. NSS - Nonnuclear safety.
b. ASTM - American Society of Testing Materials.
c. ASME - American Society of Mechanical Engineers.
d. Structural integrity only.
e. WGPS - Waste gas processing system.

TABLE 11.5.2-2 (SHEET 1 OF 3)

DETECTOR REQUIREMENTS FOR PROCESS AND EFFLUENT RADIATION MONITORS

Monitor	Location	Detector Type	Radiation Zone (mR/h)	Major Isotopes	Detectable Range (µCi/cm^3)
Gas					
Plant Vent Effluent (Low Range)					
RE-12442A (particulate)	Plant vent	Beta scintillation	0.25-2.5	I-131, I-133, Cs-134, Cs-137, Co-58, Co-60, Sr-90	3.5E-12 to 2.3E-7
RE-12442B (iodine)	Plant vent	Gamma scintillation	0.25-2.5	I-131, I-133, I-135	1.1E-10 to 2.3E-5
RE-12442C (radiogas)	Plant vent	Beta scintillation	0.25-2.5	Xe-133, Xe-135, Kr-85, Ar-41	5.0E-7 to 5.0E-2
Containment Vent Effluent					
RE-2565A (particulate)	Containment vent	Beta scintillation	0.25-2.5	I-131, I-133, Cs-134, Cs-137, Co-58, Co-60	3.5E-12 to 2.3E-7
RE-2565B (iodine)	Containment vent	Gamma scintillation	0.25-2.5	I-131, I-133, I-135	1.1E-10 to 2.3E-5
RE-2565C (noble gas)	Containment vent	Beta scintillation	0.25-2.5	Xe-133, Xe-135, Kr-85, Ar-41	1.0E-6 to 1.0E-1
Containment Atmosphere					
RE-2562A (particulate)	Containment bldg	Beta scintillation	0.25-2.5	I-131, Cs-137, Co-58, Co-60	3.5E-12 to 2.3E-7
RE-2562B (iodine)	Containment bldg	Passive iodine cartridge	0.25-2.5	I-131, I-133	NA
RE-2562C (radiogas)	Containment bldg	Beta scintillation	0.25-2.5	Xe-133, Xe-135, Kr-85, Ar-41	5.0E-7 to 5.0E-2

TABLE 11.5.2-2 (SHEET 2 OF 3)

Monitor	Location	Detector Type	Radiation Zone (mR/h)	Major Isotopes	Detectable Range ($\mu Ci/cm^3$)
Control Room Air Intake RE-12116, RE-12117	Control room	Thin walled G-M tubes	<0.25	Xe-133, Xe-135, Kr-85, I-131, I-133, Co-58, Co-60	1.0E-6 to 1.0E-1
Waste Gas Processing System RE-0013	Aux bldg	G-M tubes	2.5-15	Xe-133, Xe-135, Kr-85	9.9E-1 to 2.5E+3
Waste Gas Processing System Effluent ARE-0014	Aux bldg	G-M tube	0.25-2.5	Xe-133, Xe-135, Kr-85	1.0E-1 to 1.0E+4
Waste Gas Decay Tank RE-0039A, B, and C	Aux bldg	Thin walled G-M tube	0.25-2.5 2.5-15 2.5-15	Xe-133, Xe-135, Kr-85	1.0E-6 to 1.0E-1
Fuel Handling Building Effluent ARE-2532A and B ARE-2533A and B	Fuel handling bldg	Thin walled G-M tube	0.25-2.5	Xe-133, Xe-135, Kr-85	1.0E-6 to 1.0E-1
Primary-to-Secondary N16 Leak Detection RE-724	Turbine Building	Gamma Scintillation	<0.25	N-16	2.0E-8 to $2x10^{-3}$
Primary-to-Secondary Noble Gas Leak Detection RE-810	Turbine Building	Beta Scintillation	<0.25	Xe-133, Xe-135, Kr-85 Kr-88	3.74E-8 to 9.09E-2
Radwaste Processing Facility Vent ARE-16980 (particulate)	Radwaste Processing Facility	Beta Scintillation	<2.5	Co-58, Co-60, Cs-134, Cs-137	1.0E-13 to 1.0E-7

TABLE 11.5.2-2 (SHEET 3 OF 3)

Monitor	Location	Detector Type	Radiation Zone (mR/h)	Major Isotopes	Detectable Range ($\mu Ci/cm^3$)
Liquid					
RE-0017A and B (component water)	Aux bldg	Gamma scintillation	0.25	Co-58, Co-60, I-131, I-133, Cs-134, Cs-137	4.0E-7 to 4.0E-2
RE-0018 (waste liquid effluent)	Aux bldg	Gamma scintillation	2.5-15	Co-58, Co-60, I-131, I-133, Cs-134, Cs-137	1.0E-6 to 1.0E-1
RE-0019 (steam generator liquid)	Aux bldg	Gamma scintillation	2.5-15	Co-58, Co-60, I-131, I-133, Cs-134, Cs-137	4.0E-7 to 4.0E-2
RE-0020A and B (nuclear service water)	Aux bldg	Gamma scintillation	0.25-2.5	Co-58, Co-60, I-131, I-133, Cs-134, Cs-137	4.0E-7 to 4.0E-2
RE-0021 (steam generator blowdown)	Aux bldg	Gamma scintillation	0.25-2.5	Co-58, Co-60, I-131, I-133, Cs-134, Cs-137	4.0E-7 to 4.0E-2
RE-0848 (turbine bldg drain)	Turbine bldg	Gamma scintillation	0.25-2.5	Co-58, Co-60, I-131, I-133, Cs-134, Cs-137	4.0E-7 to 4.0E-2
RE-1950 (aux component cooling water)	Aux bldg	Gamma scintillation	0.25-2.5	Co-58, Co-60, I-131, I-133, Cs-134, Cs-137	4.0E-7 to 4.0E-2
RE-48000 (CVCS letdown)	Aux bldg	Gamma scintillation	100-250	Co-58, Co-60, I-131, I-133, Cs-134, Cs-137	2.0E-4 to 4.0E+1

TABLE 11.5.2-3 (SHEET 1 OF 2)

CONDITIONS OF SERVICE FOR PROCESS AND EFFLUENT RADIATION MONITORS

	PROCESS CONDITIONS			MONITOR LOCATION CONDITIONS			
Monitor	Sample Fluid	Operating Temperature (°F)	Operating Pressure (psig)	Temp (°F)	Pressure (psig)	Relative Humidity (%)	Building Location/ Elevation
Gas							
Plant vent effluent RE-12442A	Gas	40-104	0	Unit 1 32-120 Unit 2 40-120	0	35-90	Equip bldg 220 ft
Plant vent effluent RE-12442B	Gas	40-104	0	Unit 1 32-120 Unit 2 40-120	0	35-90	
Plant vent effluent RE-12442C	Gas	40-104	0	Unit 1 32-120 Unit 2 40-120	0	35-90	
Containment vent effluent RE-2565A	Gas	60-120	0	Unit 1 32-120 Unit 2 40-125	0	35-90	Equip bldg 220 ft
Containment vent effluent RE-2565B	Gas	60-120	0	Unit 1 32-120 Unit 2 40-125	0	35-90	
Containment vent effluent RE-2565C	Gas	60-120	0	Unit 1 32-120 Unit 2 40-125	0	35-90	
Containment atmosphere RE-2562A	Gas	60-120	(-1.5)-3	Unit 1 40-104 Unit 2 40-120	0	20-95	Aux bldg 170 ft 6 in.
Containment atmosphere RE-2562B	Gas	60-120	(-1.5)-3	Unit 1 40-104 Unit 2 40-120	0	20-95	
Containment atmosphere RE-2562C	Gas	60-120	(-1.5)-3	Unit 1 40-104 Unit 2 40-120	0	20-95	
Control room intake RE-12116, RE-12117	Gas	5-104	0	65-100	0	20-60	Control bldg 260 ft
Waste gas processing RE-0013	Gas	70-140	0.5-2	40-104	0	20-95	Aux bldg 188 ft 3 in.
Waste gas processing system RE-0014	Gas	70-140	0	40-104	0	20-95	Aux bldg 178 ft
Waste gas decay tank RE-0039A, RE-0039B, RE-0039C	Gas	40-104	0	40-104	0	20-95	Aux bldg 189 ft Aux bldg 190 ft 3 in.

TABLE 11.5.2-3 (SHEET 2 OF 2)

Monitor	PROCESS CONDITIONS Sample Fluid	Operating Temperature (°F)	Operating Pressure (psig)	MONITOR LOCATION CONDITIONS Temp (°F)	Pressure (psig)	Relative Humidity (%)	Building Location/ Elevation
Fuel handling building effluent ARE-2532A and B	Gas	40-104	0	40-104	(-0.25)-0 (in. WG)	20-95	Fuel handling bldg 253 ft
Fuel handling building effluent ARE-2533A and B	Gas	40-104	0	40-104	-0.25)-0 (in. WG)	20-95	Fuel handling bldg 283 ft
Radwaste processing facility vent effluent ARE-16980	Gas	50-100	0	40-104	0	20-95	Radwaste processing facility
Liquid							
Component cooling water RE-0017A and B	Liquid	108	24	40-104	0	20-95	Aux bldg 240 ft
Waste liquid effluent RE-0018	Liquid	100	125	40-104	0	20-95	Aux bldg 119 ft 3 in.
Steam generator liquid RE-0019	Liquid	130	300	40-104	0	20-95	Aux bldg 170 ft 6 in.
Nuclear service water RE-0020A and B	Liquid	108	90	40-104	0	20-95	Aux bldg 195 ft
Steam generator blowdown RE-0021	Liquid	130	300	40-104	0	20-95	Aux bldg 195 ft
Turbine bldg drain RE-0848	Liquid	104	20	40-104	0	40-85	Turbine bldg 195 ft
Aux component cooling water RE-1950	Liquid	105	26	40-104	0	20-95	Aux bldg 220 ft
Chemical and volume control system letdown RE-48000	Liquid	115	350	40-104	0	20-95	Aux bldg 195 ft

TABLE 11.5.2-4

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TABLE 11.5.2-5 (SHEET 1 OF 2)

FLOWRATE MONITORING FOR LIQUID AND GASEOUS NORMAL RELEASE PATHWAYS

Pathway	Equipment Number	Range	Control Room Indication	Local Indication
Liquid				
Radwaste effluent (batch)	FE-018	0 - 225 gal/min	Yes	Yes
Steam generator blowdown effluent (batch)	FE-021	125-1250 gal/min	Yes	Yes
Turbine building drain effluent	(a)	(a)	(a)	(a)
Control building sump effluent	(a)	(a)	(a)	(a)
Gaseous				
Plant vent (includes discharges from the containment purge system, gaseous radwaste system, fuel handling building HVAC, and auxiliary building HVAC)	1FE-12442A & 12442B	3,000-190,000 sf^3/min	Yes	Yes
	2FE-12442A & 12442B	1200-120,000 sf^3/min	Yes	Yes
	1FE-12835	3200-160,000 sf^3/min	Yes	No
	2FE-12835	2,000-100,000 sf^3/min	Yes	No
Condenser air ejector and steam packing exhaust	FE-12839	0-6500 sf^3/min	Yes	Yes
Containment purge system[(a)]	FE-2565	0-20,000 sf^3/min	Yes	Yes

TABLE 11.5.2-5 (SHEET 2 OF 2)

Pathway	Equipment Number	Range	Control Room Indication	Local Indication
Waste gas processing effluent[b]	AFE-0014	0.2-50.0 sf^3/min	Yes	Yes
Radwaste processing facility vent effluent	AFE-16980	480-18,280 sf^3/min	Yes	Yes

a. These pathways are not equipped with a release flowrate measuring device due to the fact that radioactivity exceeding preset limits is not sent to the environs but instead to the respective waste processing systems.

b. These pathways are also monitored downstream by the plant vent monitor.

TABLE 11.5.3-1 (SHEET 1 OF 2)

AIRBORNE RADIOLOGICAL SAMPLING CAPABILITY FOR PERMS

	Sample Identification			
Monitor	Fluid	Location	Type	Purpose
RE-12442A (gaseous release pathway)[(a)]	Ventilation air effluent	At monitor	Particulate	Provide quantitative activity release data. Calibration of continuous monitors.
RE-12442B (gaseous release pathway)[(a)]	Ventilation air effluent	At monitor	Iodine	
RE-12442C (gaseous release pathway)[(a)]	Ventilation air effluent	At monitor	Radiogas	
			Vapor sample	
RE-2565A	Containment ventilation air	At monitor	Particulate	Provide diagnostic backup data to RE-12442A, B, and C and RE-2562A, B, and C.
RE-2565B	Containment ventilation air	At monitor	Iodine	Determine effectiveness of containment vent filters.
RE-2565C	Containment ventilation air	At monitor	Radiogas	
			Vapor sample	
RE-2562A	Containment atmosphere	At monitor	Particulate	Determine personnel access time limit inside containment. Calibration of monitor.
RE-2562B	Containment atmosphere	At monitor	Passive iodine cartridge	Determine personnel access time limit inside containment.
RE-2562C	Containment atmosphere	At monitor	Radiogas	
			Vapor sample	
RE-12116 and RE-12117	Control room air intakes	Control room intake duct	Radiogas iodine	Determine isotopic activity in control room following accident.
RE-0013	Waste gas	At monitor	Radiogas	Determine activity inventory waste gas system.
RE-12839A (gaseous release pathway)[(a)]	Condenser air ejector effluent	At monitor	Passive particulate cartridge	Provide quantitative activity release data.

TABLE 11.5.3-1 (SHEET 2 OF 2)

AIRBORNE RADIOLOGICAL SAMPLING CAPABILITY FOR PERMS

	Sample Identification			
Monitor	Fluid	Location	Type	Purpose
RE-12839B (gaseous release pathway)[(a)]	Condenser air ejector effluent	At monitor	Passive iodine cartridge	
RE-12839C (gaseous release pathway)[(a)]	Condenser air ejector effluent	At monitor release data.	Radiogas Calibration	Provide quantitative activity of continuous monitors.
ARE-0014	Waste gas system effluent (batch)	Use system sample at decay tank and at monitor	Radiogas iodine	Determine identify and quantity in effluent.
RE-0039A, B, and C	Decay tank and compressor ventilation air	At monitor	Radiogas iodine	Provide quantitative isotopic activity data.
ARE-2532A and B ARE-2533A and B	Fuel handling building ventilation air	At monitor	Radiogas iodine vapor sample	Provide quantitative isotopic activity data.
RE-12444A, F RE-12444B, G RE-12444C (gaseous release pathway)	Ventilation air effluent	At monitor	Passive particulate cartridge Passive iodine cartridge Radiogas Vapor sample	Provide quantitative activity release data. Calibration of continuous monitors.
ARE-16980 (gaseous release pathway)[(a)]	Radwaste processing facility vent effluent	At monitor	Particulate	Provide quantitative activity release data. Calibration of continuous monitors.

a. These monitors require sampling in accordance with Regulatory Guide 1.21.

TABLE 11.5.3-2

LIQUID RADIOLOGICAL SAMPLING CAPABILITY FOR PERMS

Monitor	Sample Identification Fluid	 Location	 Type	Purpose
RE-0017A and B	Component cooling water liquid	At monitor	Liquid sample	Detect activity leakage into component cooling system. Calibration of continuous monitor.
RE-0018[a][b]	Waste liquid effluent (batch release)	Use system sample at waste monitor tank and at monitor.	Liquid sample	Provide quantitative activity release data. Calibration of continuous monitors.
RE-0019	Steam generator liquid	At monitor	Liquid sample	Detect primary-secondary system leaks; continuous monitor. Locate leaking steam generator. Determine magnitude of primary to secondary leakage.
RE-0020A and B[a]	Nuclear service water liquid effluent (continuous release)	At monitor	Liquid sample	Provide quantitative activity release data. Calibration of continuous monitors.
RE-0021[a]	Steam generator blowdown system liquid process and/or effluent	At monitor	Liquid sample	Provide quantitative activity release data. Calibration of continuous monitors.
RE-0848[a][b]	Turbine bldg liquid effluent (continuous)	Turbine bldg drain tanks and sumps, and at monitor	Liquid sample	Provide quantitative activity release data. Calibration of continuous monitors.
RE-1950	Aux component cooling water liquid	At monitor	Liquid sample	Detect activity leakage into aux component cooling water. Calibration of continuous monitor.
RE-48000	Letdown liquid	At monitor	Liquid sample	To detect abnormal activity levels in the reactor coolant system.

a. These monitors require sampling in accordance with Regulatory Guide 1.21.

b. Sampling also required to provide process concentration following activation of control function (See also subsection 7.3.6.)

TABLE 11.5.5-1 (SHEET 1 OF 2)

DESIGN PARAMETERS FOR POST-ACCIDENT RADIATION MONITORS

Monitor	Monitor Type	Safety Classification	Mechanical Code	Seismic Qualification	Number of Monitors per Unit	Moving Filter	Charcoal Cartridge Particulate Filter	Pump	Process Flow (ft^3/ min)	Automatic Control Function	Duct Size/Pipe Diameter Material, Schedule	Flow Instrumentation: Instrument No.	Flow Instrumentation: Range (sf/min)
Plant Vent Effluent													
RE-12444A RE-12444F (particulate)	Effluent passive particulate filter	Cat. 2	B 31.1	Yes[e]	1		X	X	150,000 (9000)[b]	None	120 in. x 60 in. duct (Unit 1) 60 in. x 78 in. duct (Unit 2) ASTM 526, 12 gauge	FE-12835	3200-160000
RE-12444B RE-12444G (iodine)	Effluent passive iodine cartridge	Cat. 2	B 31.1	Yes[e]	1		X	X	150,000 (9000)[b]	None			
RE-12444C,D,E (radiogas)	Effluent Beta Scintillation CdTe, CdTe	Cat. 2	B 31.1	Yes[e]	1			X	150,000 (9000)[b]	None			
Main Steam Line													
RE-13119 RE-13120 RE-13121 RE-13122	Strap on	SC-3/IE	NA	Yes[a]	1 1 1 1				NA	None	Pipe 28-0-5 in. dia. 2.5 in. WT 3.0 in. INS	FT-542/543 FT-512/513 FT-522/523 FT-532/533	0-5 million lb/h
Condenser Air Ejector and Steam Packing Exhauster													
RE-12839A	Effluent passive particulate filter	NNS[d]	B 31.1	No	1		X	X	2100	None	18-in. dia. ASTM A-106-GRB std wt	FE-12839	0-6500
RE-12839B	Effluent passive iodine cartridge	NNS	B 31.1	No	1		X	X	2100	None	See ARE-0016 specs		

TABLE 11.5.5-1 (SHEET 2 OF 2)

DESIGN PARAMETERS FOR POST-ACCIDENT RADIATION MONITORS

Monitor	Monitor Type	Safety Classification	Mechanical Code	Seismic Qualification	Number of Monitors per Unit	Moving Filter	Charcoal Cartridge Particulate Filter	Pump	Process Flow (ft^3/ min)	Automatic Control Function	Duct Size/Pipe Diameter Material, Schedule	Flow Instrumentation Instrument No.	Flow Instrumentation Range (sf/min)
RE-12839C,D,E	Effluent Beta Scintillation G-M, G-M	NNS	B 31.1	No	1			X	2100	Yes; diverts air ejector and steam packing exhauster discharge through charcoal filtration	18-in. dia. ASTM A-106-GRB std wt		

a. These monitors are qualified to perform their function following design basis seismic event.

b. Accident condition flow for Unit 1. Flow for Unit 2 is 90,000 ft/min for normal operation and 3200 ft/min for accident conditions.

c. ASTM - American Society of Testing Materials.

d. NNS - Nonnuclear safety.

e. Seismic qualification is limited to the isokinetic nozzles, flow transmitter, monitoring skid, and associated data processing module.

TABLE 11.5.5-2 (Sheet 1 of 2)

DETECTOR REQUIREMENTS FOR POST-ACCIDENT RADIATION MONITORS

Monitor	Location	Detector Type	Radiation Zone (MR/h)	Major Isotopes	Detectable Range ($\mu Ci/cm^3$)
Plant Vent Effluent					
RE-12444A RE-12444F (particulate)	Plant vent	Passive particulate Filter	0.25-2.5 (10-100 r/h)[(a)]	I-131, I-133, Cs-134, Cs-137, Co-58, Co-60, Sr-90	NA
RE-12444B RE-12444G (iodine)		Passive iodine cartridge	0.25-2.5 (10-100 r/h)	I-131, I-133	NA
RE-12444C, (radiogas)		Beta scintillation	0.25-2.5 (10-100 r/h)	Xe-133, Xe-135, Kr-85	$1.0E^{-7}$ to $3.4E^{-3}$ for Xe-133
RE-12444D	Plant vent	CdTe	0.25 - 2.5 (10 - 100 r/h)	Xe-133, Xe-135, Kr-85, Kr-88	$3.4E^{-3}$ to $4.0E^{+1}$
RE-12444E	Plant vent	CdTe	0.25 - 2.5 (10 - 100 r/h)	Xe-133, Xe-135, Kr-85, Kr-88	$4.0E^{+1}$ to $5.8E^{+4}$
Main Steam Line RE-13119 RE-13120 RE-13121 RE-13122	MSIV areas	Strap on gamma	0.25-2.5 (10-100 r/h)	Xe-133, Xe-135, Kr-85m, Kr-88	$1.0E^{-1}$ to $1.0E^{+3}$
Condenser Air Ejector and Steam Packing Exhauster					
RE-12839A (particulate)	Turbine bldg.	Passive particulate filter	<0.25 (10-100 r/h)	Cs-134, Cs-137, C0-58, Co-60	NA

TABLE 11.5.5-2 (Sheet 2 of 2)

DETECTOR REQUIREMENTS FOR POST-ACCIDENT RADIATION MONITORS

Monitor	Location	Detector Type	Radiation Zone (MR/h)	Major Isotopes	Detectable Range ($\mu Ci/cm^3$)
RE-12839B (iodine)	Turbine bldg.	Passive iodine cartridge	<0.25 (10-100 r/h)	I-131, I-133	NA
RE-12839C (radiogas)	Turbine bldg	Beta scintillation	<0.25 (10-100 r/h)	Xe-133, Xe-135, Kr-85 Kr-88	$5.0E^{-7}$ to $3.4E^{-3}$ for Xe-133
RE-12839D	Turbine bldg.	G-M tube	<0.25 (10 - 100 r/h)	Xe-133, Xe-135, Kr-85, Kr-88	$3.4E^{-3}$ to $4.0E^{+1}$
RE-12839E	Turbine bldg.	G-M tube	<0.25 (10 - 100 r/h)	Xe-133, Xe-135, Kr-85, Kr-88	$4.0E^{+1}$ to $1.0E^{+5}$

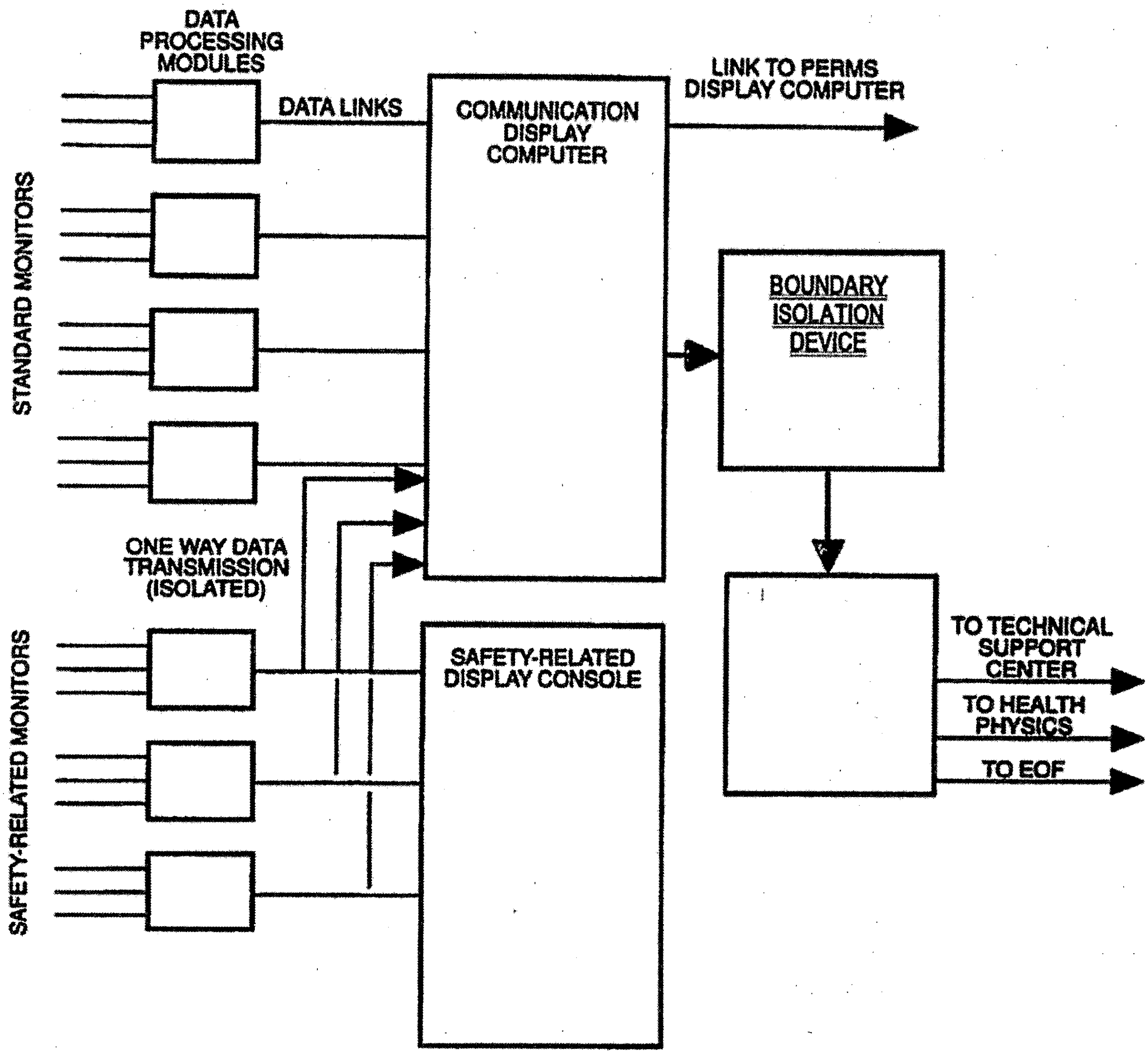
a. Values in parentheses for accident conditions.

TABLE 11.5.5-3

CONDITIONS OF SERVICE FOR POST-ACCIDENT RADIATION MONITORS

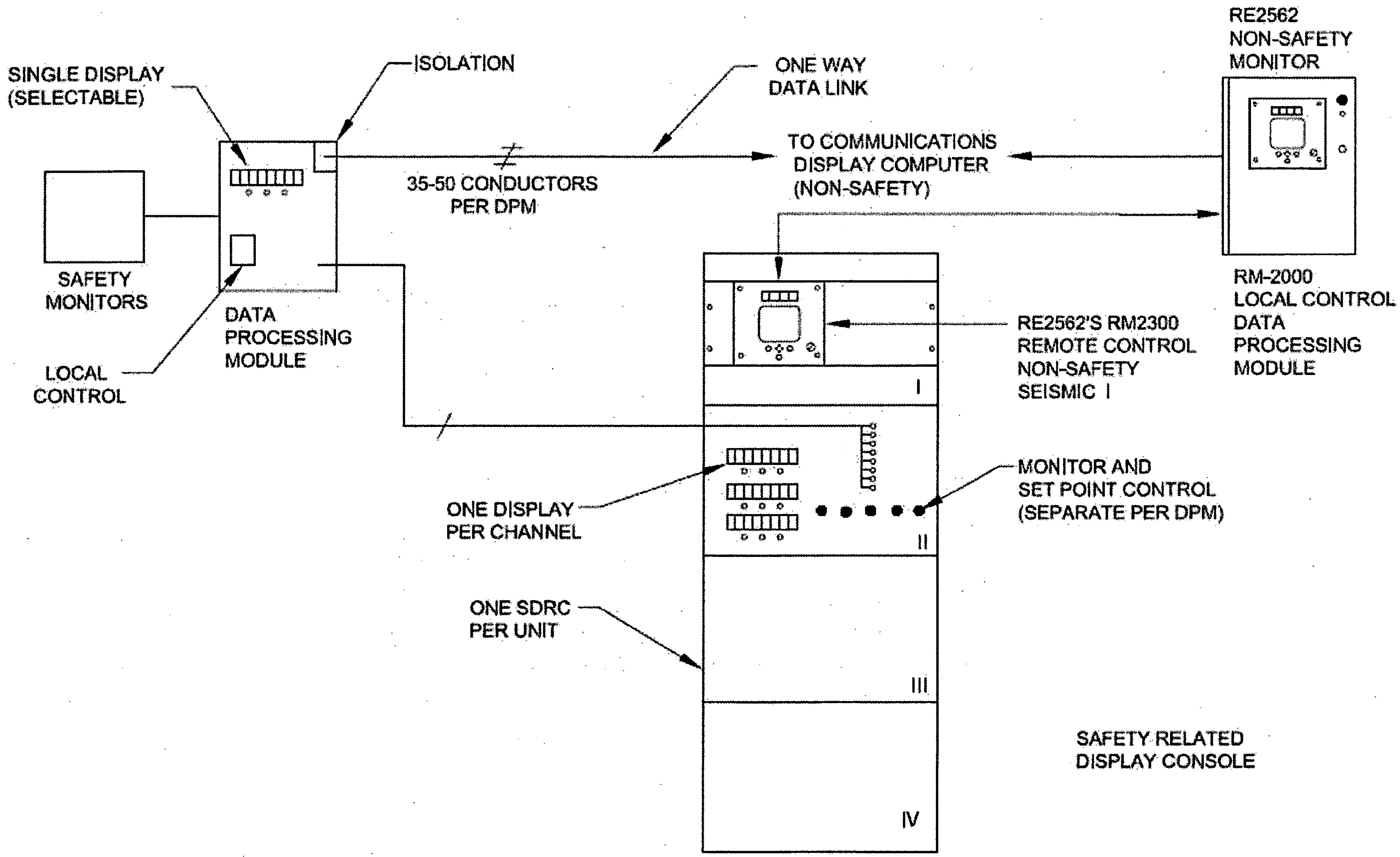
	PROCESS CONDITIONS			MONITOR LOCATION CONDITIONS			
Monitor	Sample Fluid	Operating Temperature (°F)	Operating Pressure (psig)	Temperature (°F)	Relative Pressure (psig)	Building Humidity (%)	Location/ Elevation
Plant Vent Effluent							
RE-12444A RE-12444F	Gas	40-104 (40-120)[a]	0 (0)[a]	32-120 (32-120)[a]	0 (0)[a]	35-90 (20-100)[a]	Equipment bldg 220 ft
RE-12444B RE-12444G	Gas	40-104 (40-120)	0 (0)	32-120 (32-120)	0 (0)	35-90 (20-100)	
RE-12444C,D,E	Gas	40-104 (40-120)	0 (0)	32-120 (32-120)	0 (0)	35-90 (20-100)	
Main Steam Line RE-13119 RE-13120 RE-13121 RE-13122	NA	NA	NA	47-115 (5-115)	0 (0)	10-100 (10-100)	Aux bldg (MSIV area) Control bldg (MSIV area)
Condenser Air Ejector and Steam Packing Exhauster							
RE-12839A	Gas	131-211[b]	0 (0)	40-104 (40-120)	0 (0)	40-85 (20-95)	Turbine bldg 270 ft
RE-12839B	Gas	131-211[b]	0 (0)	40-104 (40-120)	0 (0)	40-85 (20-95)	Turbine bldg 270 ft
RE-12839C,D,E	Gas	131-211[b]	0 (0)	40-104 (40-120)	0 (0)	40-85 (20-95)	Turbine bldg 270 ft

a. Values in parentheses for accident conditions.
b. Main process stream temperature upstream of heat exchanger on monitor skid.



REV 20 9/16

SOUTHERN COMPANY Energy to Serve Your World®	VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2	RADIATION MONITORING SYSTEM BLOCK DIAGRAM FIGURE 11.5.2–1



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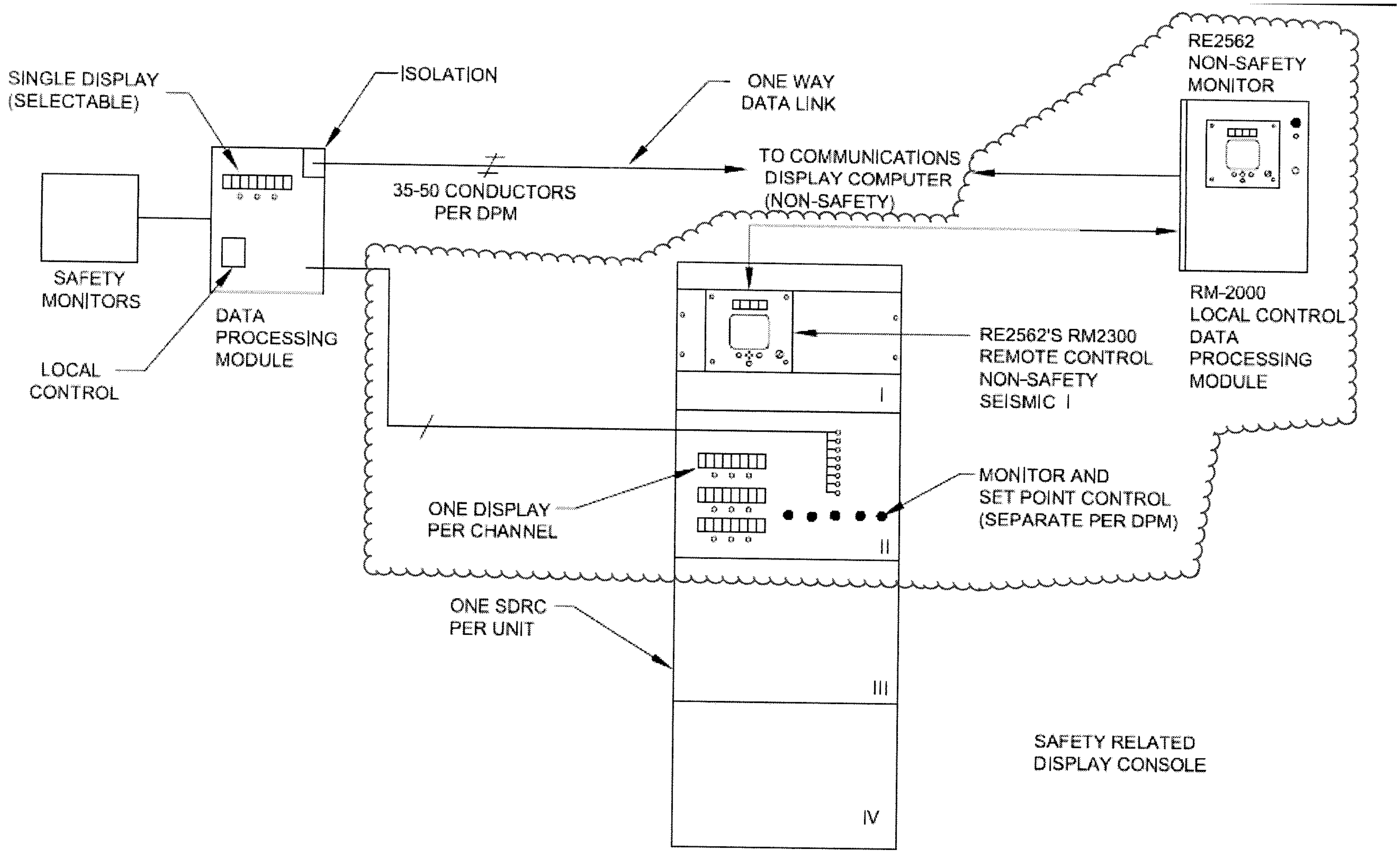
Energy to Serve Your World®

VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1

SAFETY SUBSYSTEM BLOCK DIAGRAM

FIGURE 11.5.2-2a



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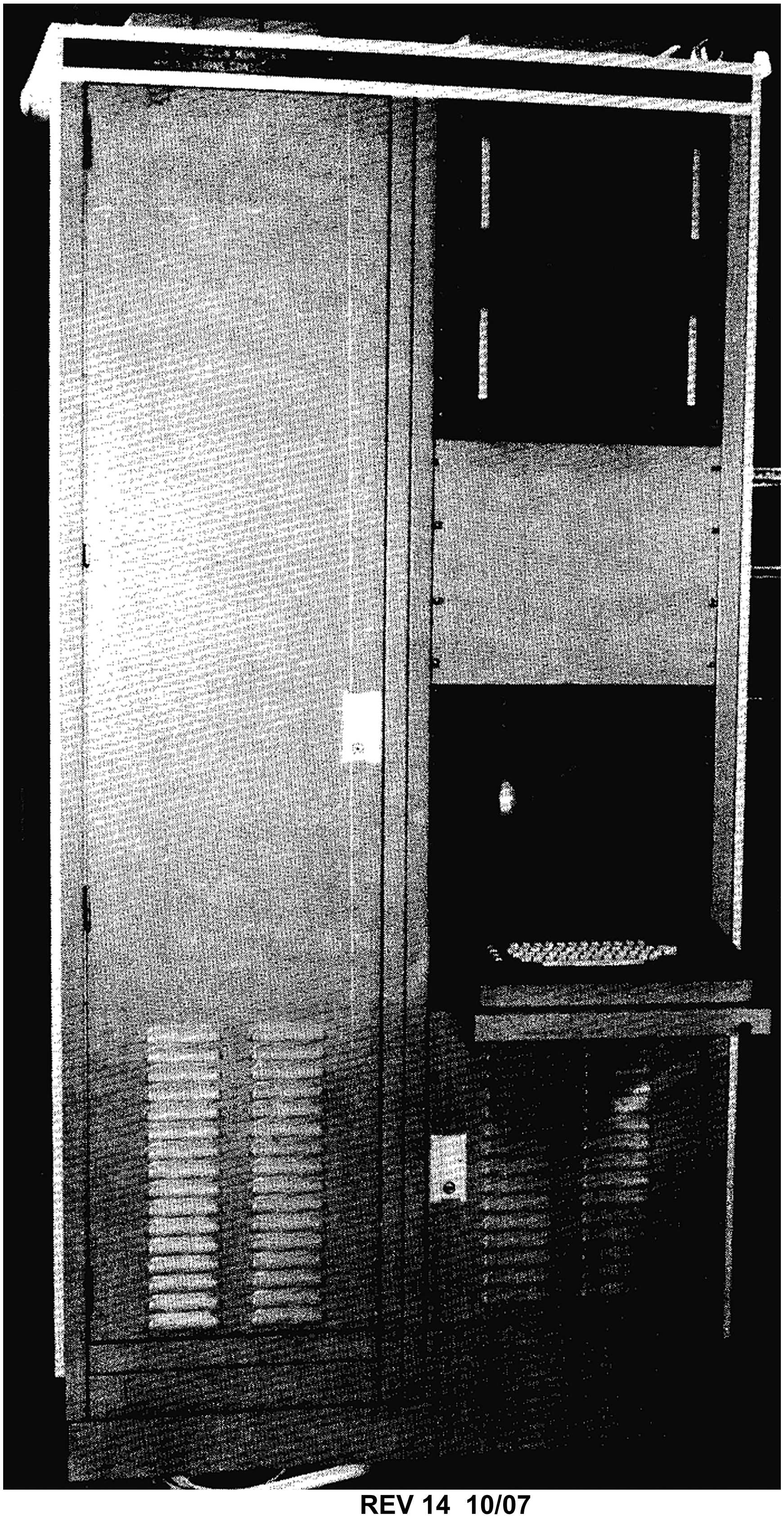
VOGTLE
ELECTRIC GENERATING PLANT
UNIT 2

SAFETY SUBSYSTEM BLOCK DIAGRAM

FIGURE 11.5.2-2b



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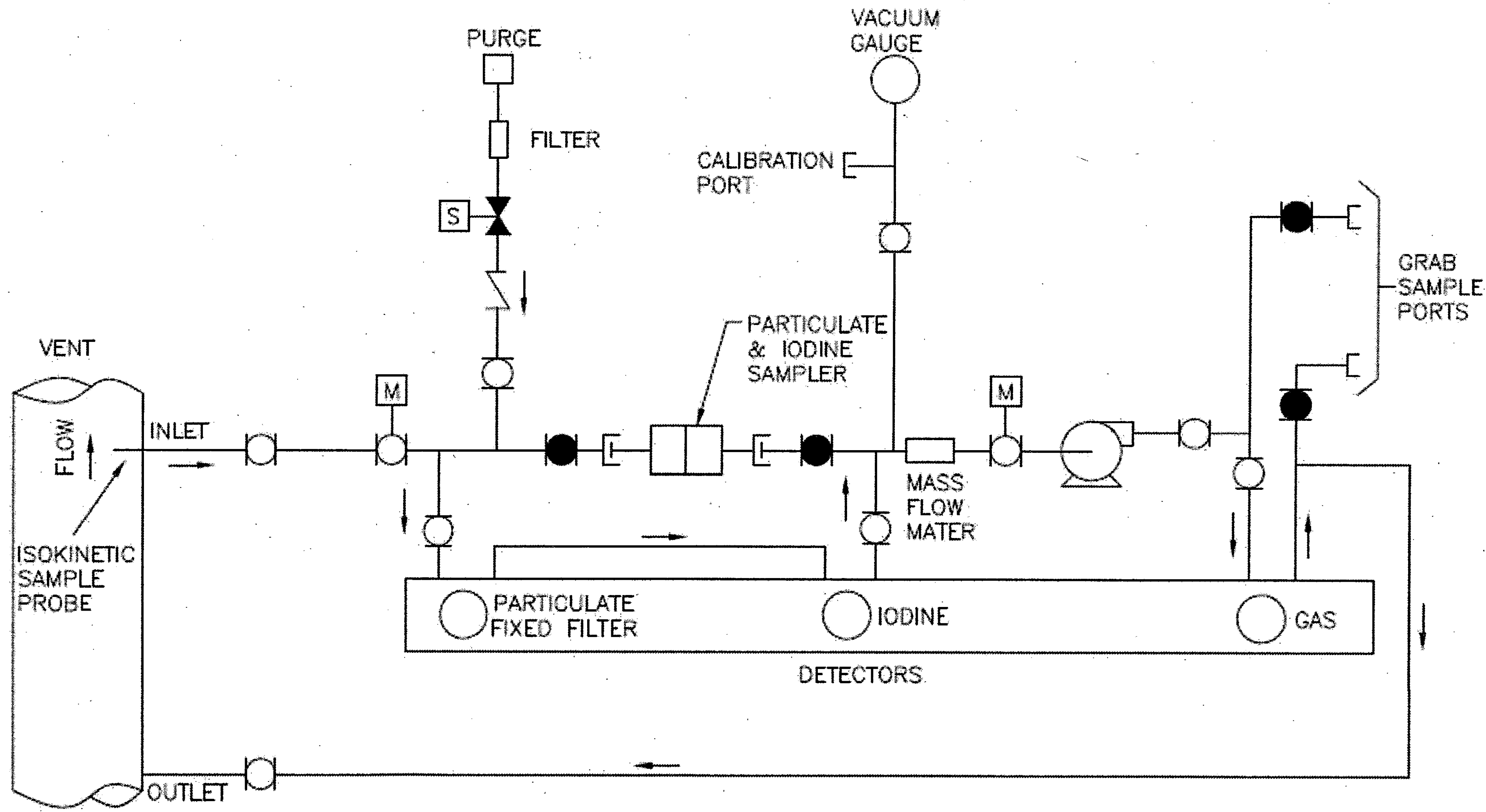
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

DRMS COMMUNICATION CONSOLE

FIGURE 11.5.2–3



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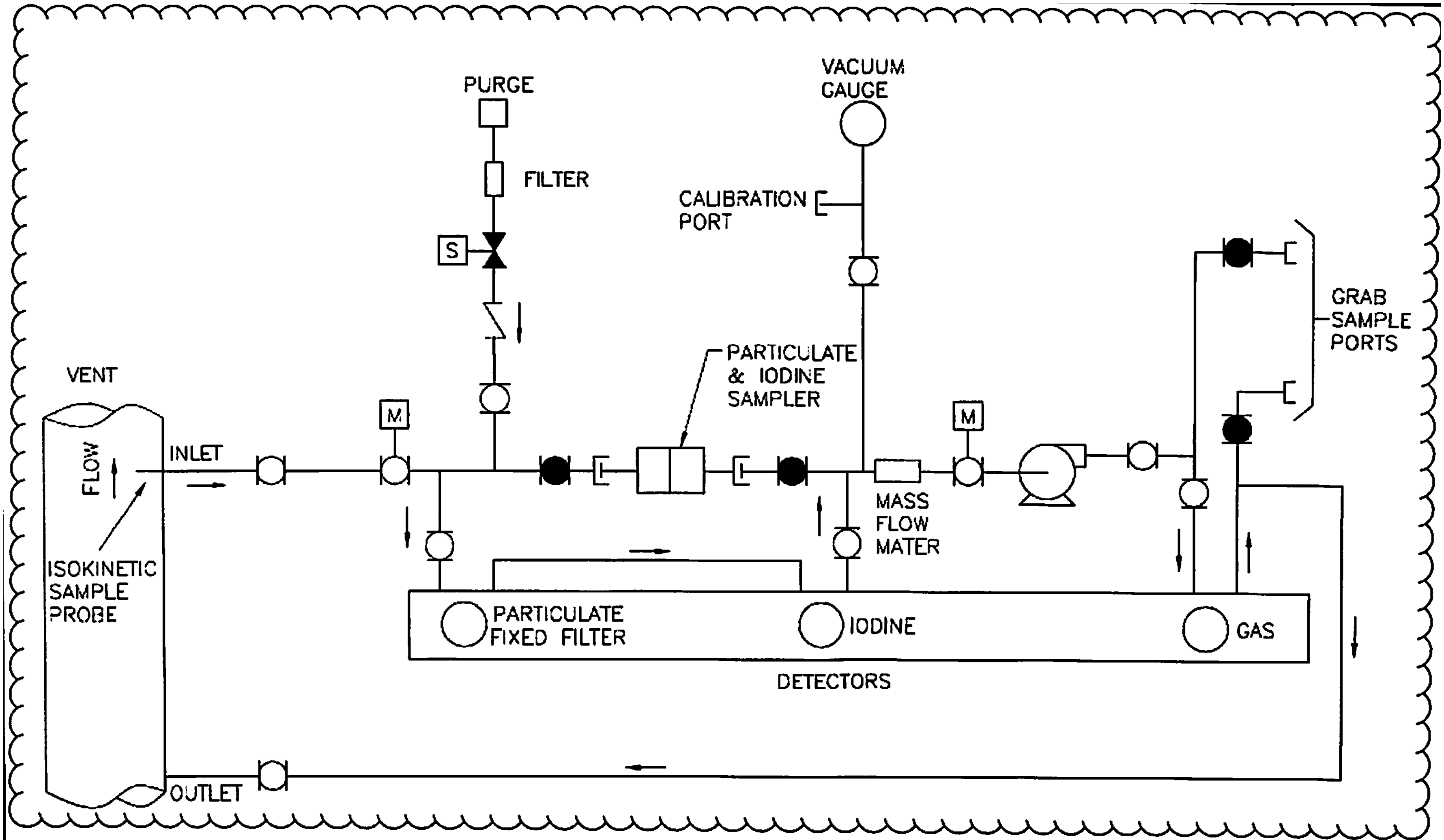
ELECTRIC GENERATING PLANT
UNIT 1

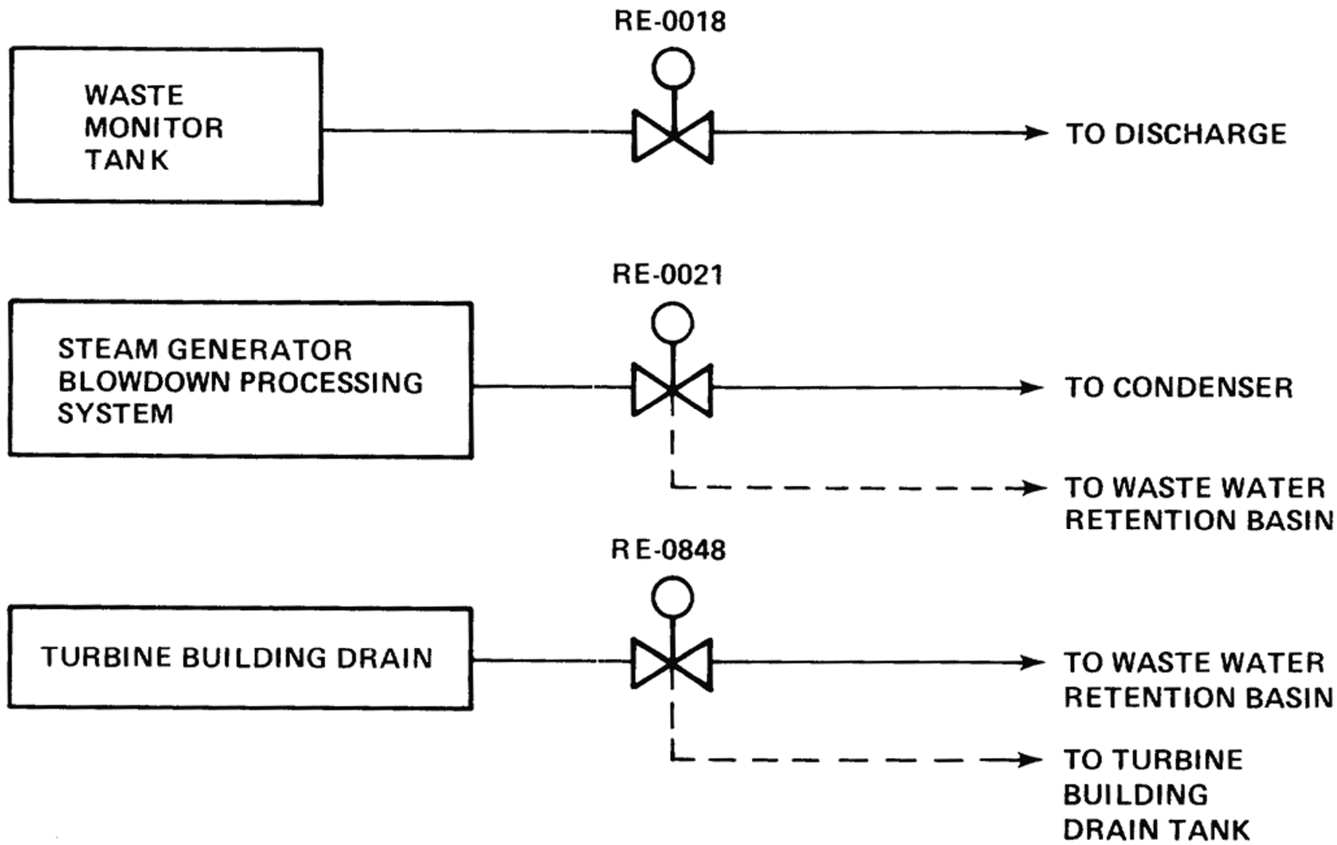
THREE-CHANNEL AIRBORNE MONITORING ASSEMBLY

FIGURE 11.5.2-4a



REV 19 4/15

SOUTHERN COMPANY *Energy to Serve Your World®*	VOGTLE ELECTRIC GENERATING PLANT UNIT 2	THREE-CHANNEL AIRBORNE MONITORING ASSEMBLY FIGURE 11.5.2-4b



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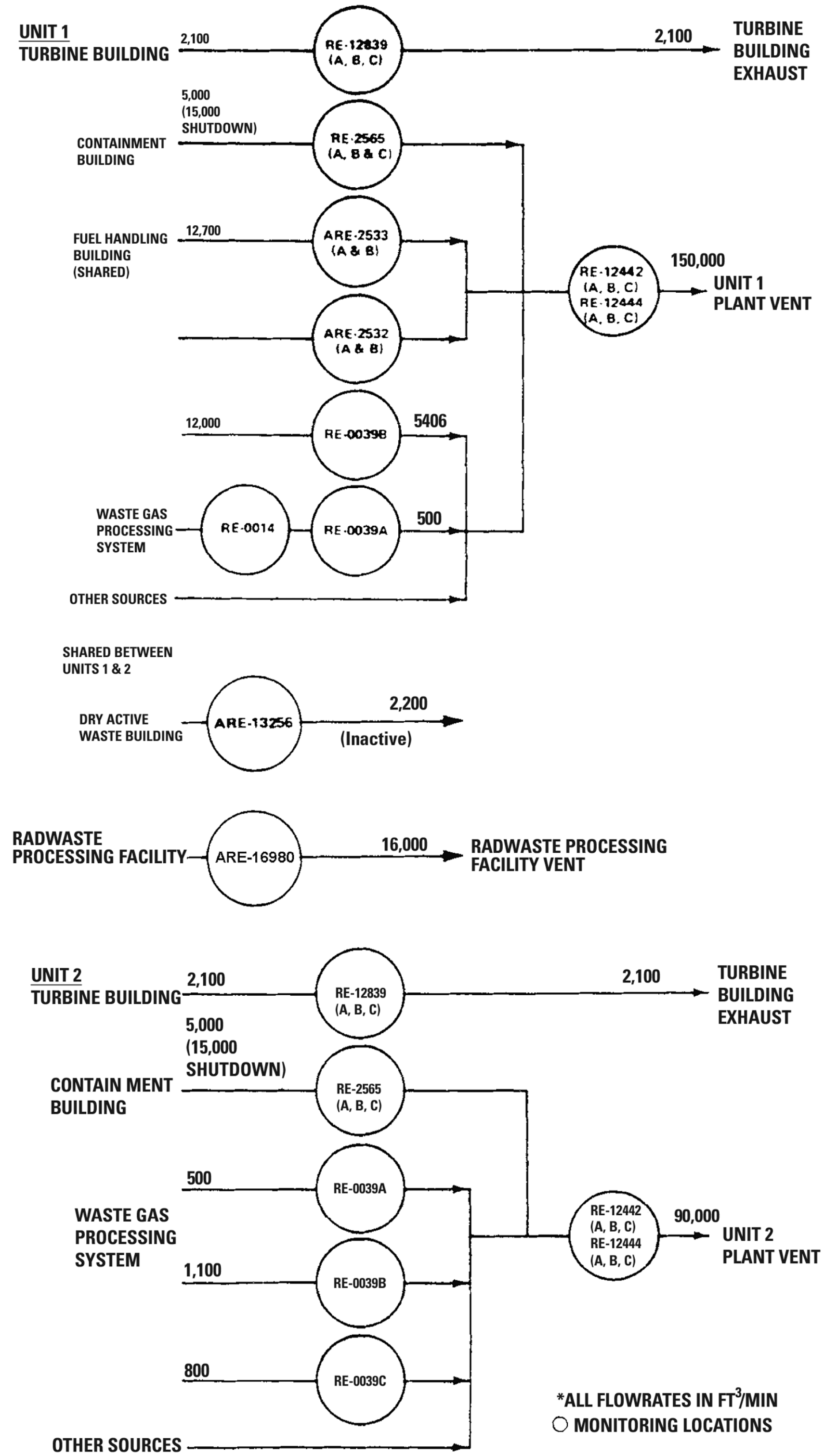
UNIT 1 AND UNIT 2

MONITORING LOCATIONS FOR
LIQUID RELEASE PATHWAYS

FIGURE 11.5.2–5



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UNIT 1 AND UNIT 2

PERMS NORMAL GASEOUS EFFLUENT RELEASE PATHWAY

FIGURE 11.5.2–6

12.0 RADIATION PROTECTION

12.1 ENSURING THAT OCCUPATIONAL RADIATION EXPOSURES ARE AS LOW AS REASONABLY ACHIEVABLE

12.1.1 POLICY CONSIDERATIONS

Administrative programs and procedures, in conjunction with facility design, ensure that the occupational radiation exposure to personnel will be kept as low as reasonably achievable (ALARA) as required by 10 CFR 20, Subpart B.

12.1.1.1 Design and Construction Policies

The ALARA philosophy was applied during the initial design of the plant and implemented via internal design reviews. Each design was given an ALARA review by engineers in the project organization of the architect/engineer. These reviewers included professional nuclear engineers and health physicists with a number of years of experience in ALARA nuclear power plant design and operation.

To comply with the ALARA policy, inspection and testing of plant shielding will be conducted to verify that the shielding performs its function of reducing radiation to design levels. During construction, a visual inspection was made to ensure that there were no major defects in the shield walls as they were poured. During initial power operations, radiation surveys will be conducted to ensure that there are no defects in the shielding that might seriously affect personnel exposures during normal operation and maintenance of the plant.

The ALARA reviews ensure that the design philosophies established in Regulatory Guide 8.8 are considered at the design stage. At the same time, these reviews ensure that a radiation exposure assessment similar to that recommended in Regulatory Guide 8.19 is performed. Design features are considered for potential exposure, and changes are recommended to reduce potentially high doses. Examples of changes effected by these ALARA reviews are:

- Additional shielding for level A filter valve galleries. Steel plates were provided to reduce direct radiation into corridor areas.

- Addition of a concrete slab over the pressurizer relief tank (PRT). The operating deck grating over the PRT was replaced with a concrete slab in order to minimize personnel exposure due to a sudden surge of reactor coolant to the PRT.

- Additional shielding for spent resin lines. Concrete floor slabs in pipe chases were locally thickened to reduce radiation levels from spent resin lines (during resin transfer) to less than 2.5 mrem/h in access corridors.

12.1.1.2 Operation Policies

It is the responsibility of the plant manager to implement a program to maintain occupational radiation exposures ALARA, consistent with recommendations of Section C.1 of Regulatory Guide 8.8, Revision 3, and Regulatory Guide 8.10, Revision 1.

To verify that the overall radiation protection program is functioning properly, the plant manager will review the written reports from audits of the ALARA program. Independent audits will be conducted by the quality assurance department, as delineated in administrative procedures.

The VEGP plant procedures manual is one of the major means of instituting the operational ALARA policy. It is available to each member of the plant staff, and each staff member will receive training covering the contents. This policy is further promoted in the health physics program, the training program, and plant procedures. (See section 12.5.) The "radiation protection plan," as defined in Section 5 of NUREG-0761, is a part of the plant procedures.

In addition to defining management's commitment to ALARA, the procedures manual designates the plant personnel who have the responsibility and authority to implement the ALARA program. The plant manager, as with all other aspects of plant operation, bears the final responsibility for implementing the ALARA program but will delegate this responsibility to the health physics manager. The authority to prevent unsafe practices and to direct steps to prevent any unnecessary radiation exposures also lies within the health physics manager's responsibilities. The health physics support supervisor will assist the health physics manager in ensuring that radiation exposures are maintained ALARA. The health physics support supervisor will report to the health physics manager who serves as the radiation protection manager referred to in Regulatory Guide 8.8. The plant health physicist or health physics support supervisor may also serve as the radiation protection manager if qualified to do so. The health physics foremen handle the day-to-day operation of the site radiation protection program and report to the health physics manager and health physics support supervisor. The foremen will implement the ALARA program and supervise the health physics technicians who perform the various surveys for radiation protection. For a more detailed discussion on the responsibility and authority of the key supervisory positions discussed above and the qualifications of the personnel who fill them, see chapter 13 and section 12.5.

It is the responsibility of the health physics manager to see that company employees and contractors are trained in radiation protection requirements to comply with 10 CFR 19, 10 CFR 20, 10 CFR 55, and the recommendations of Regulatory Guides 8.8, 8.10, and 8.13 and to verify that personnel follow the radiation protection procedures. To ensure compliance with this policy, the health physics manager, health physics support supervisor, and the health physics foremen are charged with the responsibility to promptly advise higher management of any unsafe practices which exceed their authority to correct. They have the authority to halt any operation which, in their judgment, is unsafe. It is also the prerogative of the working health physics technicians to halt any operation which, in their judgment, is unsafe pending review by the health physics foreman.

In addition to reviews by management, all employees are encouraged to express their concerns for ALARA through a formal suggestion program. The program provides the basis to evaluate proposed ALARA improvements. In addition, the plant modification program also contributes to the ALARA improvements.

Prior to unit startup, personnel assigned to that unit will be trained in radiation protection procedures and techniques. Personnel will be tested prior to startup of their respective units to verify that they understand how these procedures relate to the safe performance of their jobs. Plant personnel, whose assignments require it, will be trained and tested in radiation protection procedures and techniques every 2 years. This training program will ensure compliance with

10 CFR 19.12. Contractors who work in the radiologically controlled area of the plant after initial fuel loading will receive radiation protection training commensurate with the safe performance of their jobs, and they will be tested on this phase of their training. Construction personnel will also be instructed in site emergency procedures.

Maintenance, refueling, and radwaste system operating procedures which involve significant radiation exposures will be reviewed to verify adherence to ALARA policy prior to their use. This review will be performed in accordance with the established ALARA program. ALARA will also be considered when the system or plant modifications are reviewed by the plant review board. A qualified health physics technician or health physics foremen will observe implementation of selected activities to identify situations in which exposures can be reduced. Those operations with higher exposure potentials will receive greater attention.

12.1.1.3 Conformance with Regulatory Guide 8.8

VEGP's design has been reviewed to ensure operator doses will be ALARA. Many of the recommendations of Revision 3 (June 1978) to Regulatory Guide 8.8, Information Relevant to Ensuring That Occupational Radiation Exposures at Nuclear Power Stations Will Be As Low As Reasonably Achievable, which was issued 4 years after the construction permit issue date, have been incorporated in the design.

12.1.1.4 Conformance with Regulatory Guide 8.10

VEGP's plant procedures will ensure conformance with the requirements of Revision 1R of Regulatory Guide 8.10, Operating Philosophy for Maintaining Occupational Radiation Exposures As Low As Reasonably Achievable, except that the radiation worker training testing frequency will be determined per the Systematic Approach to Training process in accordance with VEGP’s training programs accredited by the National Academy for Nuclear Training.

12.1.2 DESIGN CONSIDERATIONS

This subsection discusses the methods and features by which the policy considerations of subsection 12.1.1 are applied. Provisions and designs for maintaining personnel exposures as low as reasonably achievable (ALARA) are presented.

The engineers and supervisors assigned to the VEGP project have, in general, performed similar design work on other nuclear power plants. Through this experience, they have developed knowledge of radiation protection aspects of design which have been applied to VEGP. Design engineers are also made aware of nuclear plant operating experiences through Nuclear Regulatory Commission (NRC) inspection and enforcement bulletins, generic letters, circulars, and information notices; NRC power reactor events; and internal procedures of the participating design organizations.

The VEGP is committed to the concept of ALARA. The project initiates formal reviews of the facility and equipment design to ensure that ALARA concerns are addressed. Designers employ the guidelines of NRC Regulatory Guide 8.8, Revision 3, and VEGP conforms to the intent of this regulatory guide. Design alterations are made when deficiencies are identified, as practicable. The ALARA reviews form the basis for identifying conformance and exceptions to the regulatory guide.

The basic philosophy of VEGP plant management, to ensure that occupational radiation exposures are ALARA, can be expressed as:

A. Design of structures, systems, and components to ensure increased reliability and maintainability, thereby effectively reducing the maintenance requirements on radioactive components.

B. Design of structures, systems, and components to reduce the radiation fields to ensure that operation, maintenance, and inspection activities are performed in the minimum radiation field feasible.

C. Design of structures, systems, and components to reduce the time spent in radiation fields during operation, maintenance, and inspection.

D. Design of structures, systems, and components to accommodate remote and semiremote operation, maintenance, and inspection procedures.

12.1.2.1 General Design Considerations for ALARA Exposures

General design considerations and methods employed to maintain in-plant radiation exposures ALARA consistent with the recommendations of NRC Regulatory Guide 8.8 have two objectives:

A. Minimizing the necessity for, and amount of, personnel time spent in radiation areas.

B. Minimizing radiation levels in routinely occupied plant areas in the vicinity of plant equipment expected to require personnel attention.

Equipment and facility designs are considered in maintaining exposures ALARA during plant operations, including:

- Normal operation.
- Maintenance and repairs.
- Refueling operations and fuel storage.
- Inservice inspection and calibrations.
- Radioactive waste handling and disposal.
- Other anticipated operational occurrences.

The actual design features employed are described in general in subsection 12.3.1.

Paragraphs 12.1.2.2 and 12.1.2.3 list general design considerations for maintaining exposures ALARA in accordance with Regulatory Guide 8.8.

12.1.2.2 Equipment General Design Considerations for ALARA

A. Equipment general design considerations to minimize the necessity for, and amount of, personnel time spent in a radiation area include:

1. Reliability, durability, construction, and design features of equipment, components, and materials to reduce or eliminate the need for repair or preventive maintenance.

2. Servicing convenience for anticipated maintenance or potential repair, including ease of disassembly and modularization of components for replacement or removal to a lower radiation area for repair.

3. Provisions, where practicable, to remotely or mechanically operate, repair, service, monitor, or inspect equipment.

4. Redundancy of equipment or components to reduce the need for immediate repair when radiation levels may be high and when no feasible method is available to reduce radiation levels.

B. Equipment general design considerations directed toward minimizing radiation levels proximate to equipment or components requiring personnel attention include:

1. Provisions for draining, flushing, or, if necessary, remote cleaning of equipment containing radioactive material.

2. Design of equipment, piping, connections, and valves to minimize the buildup of radioactive material and to facilitate flushing of crud traps.

3. Use of high quality valves, valve packings, and gaskets to minimize leakage and spillage of radioactive materials.

4. Provisions for minimizing the spread of contamination into equipment service areas, including direct drain connections.

5. Provisions for isolating equipment from radioactive process fluids.

6. Provision for a spent fuel pool cleanup system to maintain the radiation level of the fuel pool area within the Zone II limit. See table 12.3.1-1 for the description of radiation zones.

12.1.2.3 Facility Layout General Design Considerations for ALARA

A. Facility general design considerations to minimize the amount of personnel time spent in a radiation area include:

1. Locating equipment, instruments, and sampling stations which will require routine maintenance, calibration, operation, or inspection, for ease of access and minimum of required occupancy time in radiation areas.

2. Laying out plant areas to allow remote or mechanical operation, service, monitoring, or inspection of highly radioactive equipment.

3. Providing, where practicable, for transportation of equipment or components requiring service to a lower radiation area.

B. Facility general design considerations directed toward minimizing radiation levels in plant access areas and in the vicinity of equipment requiring personnel attention include:

1. Separating radiation sources and occupied areas where practicable (e.g., pipes or ducts containing potentially highly radioactive fluids do not pass

through occupied areas). Redundant components requiring maintenance which are a source of radiation are located in separate compartments to allow maintenance of one component while the other component is in operation. Equipment such as demineralizers and filters are separated by shielding from nonradioactive equipment to facilitate unrestricted maintenance on the latter. Areas between radiation sources and access and service areas are adequately shielded.

2. Providing labyrinth entrances to radioactive pump, equipment, and valve rooms. Adequate space is provided in labyrinth entrances for easy access. Highly radioactive passive components with minimal maintenance requirements are located in completely enclosed compartments and are provided with access via a shielded hatch or removable blocks.

3. Where appropriate, separating equipment or components in service areas with permanent shielding. Also provided are means and adequate space for utilizing movable shielding for sources within the service area when required.

4. Plant layout incorporates restrictions and control of access to the various radiation zones. In the case of an abnormal occurrence or accident, the zone restrictions may change due to increased dose rates. Special access controls would be implemented at these times to minimize personnel exposure.

   Access to a given radiation zone generally does not require passing through a higher radiation zone. For example, personnel are not normally required to pass through a Zone IV area to gain access to a Zone III area. Exception to this is taken within the containment. The plant design incorporates personnel escape routes which allow personnel to rapidly exit from an area in which the radiation exposure level has unexpectedly increased.

5. Locating equipment, instruments, and sampling sites in the lowest practicable radiation zone.

6. Providing means to control contamination, to facilitate decontamination of potentially contaminated areas, and for decontamination of service areas.

7. Providing backflushable and cartridge filter systems for specified high radioactive filtration requirements.

12.1.3 OPERATIONAL CONSIDERATIONS

In accordance with Southern Nuclear Operating Company (SNC) policy, and consistent with the recommendations of Regulatory Guides 8.8 and 8.10, the radiation exposure of plant personnel will be kept as low as reasonably achievable (ALARA) by means of the health physics program discussed in section 12.5. The radiation protection policies and practices contained therein are initiated through the training program discussed in section 13.2 and through the plant procedures manual discussed in subsection 12.1.1 and section 12.5.

The following will meet the requirements of NUREG-0737, Item III.D.1.1.

VEGP is committed to a program to reduce leakage from systems outside containment which could contain highly radioactive fluids during a serious transient or accident to an as low as practical level in accordance with the Technical Specifications.

This program includes inspection and leak rate tests with the exception of those systems that are known to contain highly radioactive fluids and/or when a leak is discovered in a high radiation area. In both of these exceptions, leak rate testing will not be performed due to the personnel hazard from a radiation exposure standpoint. The systems that are known to contain highly radioactive fluids will be pressurized and inspected for leaks. The procedures have been written for inspection and surveillances to require a work request upon identification of leakage. Leak rate measurements and/or inspections will be performed periodically at the intervals specified in the Technical Specifications. Quantitative leak testing will be performed on the gaseous system covered by this program.

12.1.3.1 Procedures

Procedures for radiation area jobs that routinely occur at an operating reactor will be written and approved for use at VEGP. For unusual or first-time operations which will involve significant radiation exposure, operating procedures or work control documents will normally be prepared by or with the assistance of the section doing the work. If the operation requires plant review board review (section 13.4), then it will be reviewed and approved by the board. If plant review board review is not required, the health physics superintendent or his designee will review the procedure for ALARA purposes according to the ALARA program.

Techniques which will be embodied in plant procedures, training, and/or work practices for ALARA purposes as appropriate are discussed below. Also discussed are the criteria and/or conditions for their use. These techniques will be written into plant procedures as appropriate and must be followed in accordance with SNC requirements for procedure compliance.

A large percentage of exposure at an operating reactor occurs during plant outages from maintenance and inspection activities and not from normal operating activities. This is to be expected since, during operation, instrumentation and valves can be operated from outside the shield walls, and operators only have to enter cubicles containing radioactive equipment for short periods of time to check equipment. Maintenance and inspection personnel, in order to perform their job, are often in close proximity to lines, valves, instruments, or other pieces of equipment which are radiation sources. For this reason, the ALARA-related procedures place emphasis on maintenance and non-routine operations.

12.1.3.2 General ALARA Techniques

Described below are several general ALARA techniques. These methods will be incorporated into preplanning of tasks and procedure development. Further information on ALARA techniques incorporated into procedures is given in section 12.5.

A. Permanent shielding is used, where possible, by having workers stay behind walls or in areas of low-level radiation when not actively working in the radiation area. On some jobs, temporary shielding may be used. Temporary shielding will be used only if the total exposure, which includes exposure received during installation and removal of the temporary shielding, will be effectively reduced.

B. Piping systems and other pieces of equipment which are subject to crud buildup have been equipped with connections which can be used for flushing the system to eliminate potential hot-spot buildup.

Prior to performing maintenance work, consideration will be given to flushing and/or chemically decontaminating the involved system or piece of equipment in order to reduce crud levels and, hence, personnel exposure.

C. Work involving whole body exposure rates in excess of 100 mrem/h and exposure of greater than 1 person-rem per job or removable contamination levels in excess of 10^6 dpm/100 cm^2 will be preplanned. Other jobs which involve significant total man-rem exposure, but at lower dose rates, may also be preplanned. The purpose of preplanning is to carefully prepare for the job so that it can be performed in a proper, safe manner with a minimum of personnel exposure.

D. On complex jobs or jobs with exceptionally high radiation levels, dry-run training may be utilized, including mockup training as appropriate. The intent of dry-run or mockup training is to improve worker efficiency and to lower exposure times.

E. As much as practicable, jobs will be performed outside radiation areas. This includes items such as reading instruction manuals or maintenance procedures, adjusting tools or jigs, repairing valve internals, and prefabricating components.

F. For long-term repair jobs, consideration will be given to setting up a communications network to assist supervising personnel in checking on work progress from a lower radiation area.

G. Special tools or jigs will be used when their use permits the job to be performed more efficiently or prevents errors, thus reducing the time spent in a radiation area. Special tools may also be used if their use would increase the distance from the source to the worker, thereby reducing the exposure received. These special tools will be used only if the total exposure, including that received during installation and removal, is significantly reduced.

H. Entry and exit points will be established in low-level radiation areas to allow personnel access and exit in as low a level of radiation as practicable. This is done because personnel may spend a significant amount of time changing protective clothing and respiratory equipment in those access areas. These access points will be setup to limit the spread of contamination from the jobsite to as small an area as practicable.

I. Protective clothing and respiratory equipment will be selected to minimize the discomfort of workers so that efficiency will be increased and less time will be spent in radiation areas. Protective clothing prescribed by health physics will be commensurate with the hazards involved, and the requirements cannot be modified by other personnel.

J. Contamination containments (i.e., glove bags, polybottles, tents, etc.) will be used, where practicable, to allow personnel to work on highly contaminated equipment while minimizing the spread of contamination during the work.

K. Individuals will be instructed to remain in low-level radiation areas as much as possible, consistent with performing their assigned jobs. In addition, on certain jobs, surveys will be provided to clearly delineate the areas of high radiation levels and hot spots to prevent inadvertent entry into an area of much higher radiation level.

L. Personnel will be assigned self-reading dosimeters to allow determination of accumulated exposure at any time during the job. In addition, personnel performing work in containment will be assigned dosimetry capable of alarming excessive dose rate and dose. Additional discussion of personnel dosimetry is provided in paragraph 12.5.3.6.

M. On jobs where general area radiation levels are unusually high, remotely readable dosimetry or a timekeeper will keep track of the total exposure time. This will ensure that personnel do not stay in a radiation field long enough to exceed applicable dose limits.

N. On major maintenance jobs, especially those which involve high or complex radiation levels, the job preplanning will include estimates of the man-rem needed to complete the job. At the completion of the work, a debriefing session will be held with the personnel who performed the work (when practical) in an effort to determine how the work could have been completed more efficiently, resulting in less accumulated exposure. All radiation aspects (i.e., radiation, contamination, airborne radioactivity, and external or internal personnel contamination) will be compiled and filed for future reference to provide guidance during preplanning of future similar work situations. This will incorporate experience gained in performing these tasks into future work efforts.

12.2 RADIATION SOURCES

This section discusses and identifies the sources of radiation that form the basis for shield design calculations and the sources of airborne radioactivity used for the design of personnel protection measures and dose assessment.

12.2.1 CONTAINED SOURCES

The shielding design source terms are based upon the three plant conditions of normal full-power operation, shutdown, and design basis accident events.

12.2.1.1 Sources for Full-Power Operation

The primary sources of radioactivity during normal full-power operation are direct core radiation, coolant activation processes, leakage of fission products from pinhole defects in fuel rod cladding, and activation of reactor coolant corrosion products. The design basis for the shielding source terms for fission products in this section is cladding defects in fuel rods producing 1 percent of the core thermal power. The design basis for activation and corrosion product activities is derived from measurements at operating plants and is independent of fuel defect level. The design basis of 0.25 percent fuel cladding defects for shielding source terms is used to establish shielding provisions for the radwaste processing facility.

12.2.1.1.1 Reactor Core

The primary radiation from the reactor core during normal operation is neutrons and gamma rays. Figures 12.2.1-1 and 12.2.1-2 show neutron and gamma multigroup fluxes on the inside surface of the primary shield wall at the core centerline. Gamma dose rate incident on the primary shield wall at the core midplane is shown in figure 12.2.1-3. These figures are based on nuclear parameters discussed in chapter 4. Table 12.2.1-1 lists core gamma sources after shutdown for shielding requirements during shutdown and inservice inspection.

12.2.1.1.2 Reactor Coolant System (RCS)

Sources of radiation in the RCS are fission products released from fuel and activation, and corrosion products that are circulated in the reactor coolant. These sources and their bases are discussed in section 11.1.

The activation product, N-16, is the predominant contributor to the activity in the reactor coolant pumps, steam generators, and reactor coolant piping during operation. The N-16 activity in each of the components depends on the total transit time to the component and the average residence time in the component. Table 12.2.1-2 presents the RCS N-16 activity as a function of transport time in a reactor coolant loop. The N-16 activity for the pressurizer is tabulated in table 12.2.1-3.

Fission and corrosion product activities circulating in the RCS and out-of-core crud deposits comprise the remaining significant radiation sources during full-power operation. The fission and corrosion product activities circulating in the reactor coolant are given in section 11.1. The fission and corrosion product source strengths in the reactor coolant pressurizer liquid and vapor phases are given in table 12.2.1-4. The isotopic composition and specific activity of typical out-of-core crud deposits are given in table 12.2.1-5. Typically, 1 mg of deposited crud material is found on 1 cm^2 of a relatively smooth surface. This may be as much as 50 times higher in crud trap areas. Crud trap areas are generally locations of high turbulence, areas of high momentum change, gravitational sedimentation areas, high affinity material areas, and possibly thin boundary layer regions.

The N-16 activity is not a factor in the radiation sources for systems and components located outside containment due to its short, 7.11-s, half-life and the greater than 1-min transport time before the letdown flow exits the containment.

12.2.1.1.3 Main Steam Supply System

For the purpose of evaluating the steam generator blowdown processing system, radioactivity in the main steam supply system is based on a steam generator tube leakage rate of 1 gal/min concurrent with 1 percent failed fuel. Continuous operation with primary-to-secondary leakage is assumed. The RCS radionuclide concentrations used are those in table 11.1-2. The resulting steam generator secondary side water and steam fission and corrosion product concentrations are discussed in subsection 10.4.8.

It is assumed that the maximum blowdown rate to the blowdown demineralizers is 1.2 percent of the main steam rate.

12.2.1.1.4 Auxiliary Systems

12.2.1.1.4.1 Chemical and Volume Control System (CVCS). Radiation sources in the CVCS consist of radionuclides carried in the reactor coolant. The design of the CVCS ensures that most of the N-16 has decayed before the letdown stream leaves the containment by placing a delay mechanism in the letdown flowpath. All CVCS system heat exchangers other than the regenerative heat exchanger and excess letdown heat exchanger are located in the auxiliary building.

The shielding design is based on the maximum activity in each component. These sources are listed in table 12.2.1-6.

A. CVCS Heat Exchangers

The regenerative and excess letdown heat exchangers are located in the containment building. They provide the initial cooling for the reactor coolant letdown. Their radiation sources include N-16.

The magnitude of the N-16 source strength is highly sensitive to the location of these heat exchangers with respect to the RCS loop piping. Therefore, the N-16 source strengths for these heat exchangers are based on the average value in the crossover leg between the steam generator and the reactor coolant pump. The shielding design takes into account the N-16 decay from the crossover leg to each heat exchanger.

The letdown heat exchanger provides second-stage cooling for the reactor coolant prior to entering the demineralizers. The seal water heat exchanger cools the water from several sources, including the reactor coolant discharged from the excess letdown heat exchangers. The tube side of the letdown reheat

heat exchanger heats letdown water before the water enters the boron thermal regeneration demineralizers during boron release.

The radiation source strengths for the boron thermal regeneration moderating, chiller, and shell side of the letdown reheat units account for radionuclides removed by the demineralizers upstream of the units. The source strengths for the tube side of the regenerative heat exchanger account for the removal of radionuclides by the CVCS demineralizers and the volume control tank.

B. CVCS Ion Exchangers

The mixed bed demineralizer is in continuous use and removes fission products in cation and anion form. It also is highly effective in removing corrosion products. The cation bed demineralizer is used intermittently to remove lithium for pH control. It also is highly effective in removing the monovalent cations, cesium and rubidium. The short-lived isotopes are assumed to build up to saturation activities on both beds. For the long-lived isotopes, the activity retained is assumed to be evenly distributed between the two demineralizers.

The boron thermal regeneration demineralizers are used to regulate the boron concentration in the reactor coolant water. They are used during load follow operations and in removing boron from the coolant as the nuclear fuel is depleted. These demineralizers collect radioactive anions, such as iodine and bromine, which were not retained by the mixed bed demineralizer. The radionuclides retained are assumed to be evenly distributed between two of the five 90-ft3 beds during load follow operation.

C. CVCS Filters

The design criterion for CVCS filter shielding is based primarily on operating experience.

The source strengths for the reactor coolant filter correspond to an exposure rate of 500 rem/h at contact; the source strengths for the remaining filters correspond to an exposure rate of 100 rem/h at contact. These dose rates are arrived at assuming the filters are homogeneous sources with the dimensions and composition given in table 12.2.1-6.

D. Tanks

1. Volume Control Tank

The radiation sources in the volume control tank are based on a nominal operating level in the tank of 200 ft^3 in the liquid phase and 200 ft^3 in the vapor phase and on the stripping fractions given in table 11.1-1, assuming no volume control tank purge.

2. Boric Acid Storage Tank

Removal from service of recycle evaporator has eliminated this source.

12.2.1.1.4.2 Steam Generator Blowdown System. The source terms for the steam generator blowdown system are provided in tables 12.2.1-7 through 12.2.1-13.

12.2.1.1.4.3 Boron Recycle System. The source terms for the boron recycle system are listed in table 12.2.1-15.

12.2.1.1.5 Nuclear Service Cooling Water System and Component Cooling Water System

The nuclear service cooling water and component cooling water systems are normally nonradioactive or, because of inleakage, of very low level activity. The process radiation monitor for these systems is described in section 11.5. For shielding and dose assessment purposes, the nuclear service cooling water and component cooling water systems do not yield substantive doses.

12.2.1.1.6 Spent Fuel Storage and Transfer

The predominant radioactivity sources in the spent fuel storage and transfer areas in the fuel handling building are the spent fuel assemblies. The source strengths employed to determine the minimum water depth above spent fuel and shielding walls around the spent fuel pool, as well as shielding of the spent fuel transfer tube, are given in table 12.2.1-18. For shielding design, the spent fuel pool is assumed to contain the design maximum number of fuel assemblies. Limitations are placed on fuel storage and transfer operations to limit occupational doses to personnel in the fuel handling building. These limitations are administratively controlled.

12.2.1.1.7 Spent Fuel Pool Cooling and Cleanup System

Sources in the spent fuel pool cooling and cleanup system are a result of transfer of radioactive isotopes from the reactor coolant into the spent fuel pool during refueling operations. The reactor coolant activities for fission, corrosion, and activation products are decayed for the amount of time required to remove the reactor vessel head following shutdown; are reduced by operation of the CVCS purification ion exchangers; and are diluted by the total volumes of the water in the reactor vessel, refueling pool, and spent fuel pool. This activity then undergoes subsequent decay and accumulation on the spent fuel pool cooling and cleanup system filters and demineralizer.

The design and normal fission and corrosion product activities in the spent fuel pools are given in table 12.2.1-19. The source terms for the spent fuel pool demineralizers and filters are provided in tables 12.2.1-20 through 12.2.1-23.

12.2.1.1.8 Main Steam Supply and Power Conversion Systems

Potential radioactivity in the main steam supply and power conversion systems is a result of steam generator tube leaks and fuel cladding defects as discussed in paragraph 12.2.1.1.3. This radioactivity is sufficiently low so that no radiation shielding for equipment in secondary systems, other than portions of the steam generator blowdown system (paragraph 12.2.1.5), is required to meet the radiation zone requirements.

12.2.1.1.9 Liquid Systems

Radioactive inputs to the radwaste system sources include fission and activation product radionuclides produced in the core and reactor coolant. The components of the radwaste systems contain varying degrees of activity.

The concentrations of radionuclides present in the process fluids at various locations in the radwaste systems, such as pipes, tanks, filters, ion exchangers, and evaporators, are discussed in section 11.1. Shielding for each component of the radwaste systems as based on maximum activity conditions is shown in tables 12.2.1-25 through 12.2.1-30. Radiation sources in the various pumps in this system are assumed to be identical to the liquid sources in the tank from which the pump takes suction.

12.2.1.1.10 Gaseous Radwaste System

Radiation sources for each component of the waste gas system are based on operation with the maximum activity conditions as given in sections 11.1 and 11.3.

Table 12.2.1-33 lists the distribution of the radioactive gas inventory associated with operation of the gaseous radwaste system. A volume control tank purge rate of 0.7 sf^3/min is assumed. The values represent the design activity distribution with 1-percent defective fuel and the purge system operating. They are used in defining the maximum activities in the waste gas system.

The volume control tank stripping fractions used in establishing the activity distributions are given in table 12.2.1-34. These stripping fractions are calculated using the equation given in table 11.1-1 and assuming a 0.7-sf^3/min volume control tank purge rate.

The Kr-85 inventory in the gaseous radwaste system for various full-power operating times is listed in table 12.2.1-35. These values assume no leakage from the system.

The radioactive gases removed from the RCS at the volume control tank are continuously recirculated through a gas decay tank and other gaseous radwaste system equipment, including the hydrogen recombiners and waste gas compressors. The gamma ray source strengths for this equipment are derived from cold shutdown procedures during which the radioactive gases are stripped from the RCS. Since the gases are continuously recirculated, the gamma ray source strengths for the hydrogen recombiner, waste gas compressor, and gas decay tank are identical. Table 12.2.1-36 lists the activities for the recirculation equipment.

12.2.1.1.11 Instrument Calibration Source

For the purpose of instrument calibration, a Cs-137 source is provided in the instrument calibration lab. This source is shielded to keep personnel exposures limits within ALARA limits.

12.2.1.2 Sources for Shutdown

In the reactor shutdown condition, the only additional significant sources requiring permanent shielding consideration are the spent reactor fuel, the residual heat removal system, and the incore detector system. Individual components may require shielding during shutdown due to deposited crud material. Estimates of accumulated crud are given in paragraph 12.2.1.1. The radiation sources in the RCS and other systems addressed in paragraph 12.2.1.1 are bounded by the sources given for full power operation with the exception of a short time period, i.e., less than 24 h, following shutdown, during which the fission product spiking phenomenon and crud bursts can result in increased radiation sources. The spiking phenomenon involves the release of a portion of the accumulated water soluble salts from the interior cladding surface, e.g., iodine and cesium, and gases, e.g., xenon and krypton, of defective fuel rods during the shutdown and coolant depressurization.[1] Crud bursts are the resuspension or solubilization of a portion of the accumulated deposited corrosion products into the RCS during shutdown such as during oxygenization of the reactor coolant. However, special shielding considerations to accommodate these increases should be unnecessary due to several factors, including:

A. The spike or crud burst release is of short duration (generally less than 6 h).

B. The CVCS is generally in operation at full reactor coolant purification capability during the shutdown (130 gal/min).

C. The reactor coolant activity is limited by the Technical Specifications to approximately one-fourth of the design basis activity levels.

The maximum gamma ray source strengths in the residual heat removal system are given in table 12.2.1-37 for 4 and 8 h after reactor shutdown and the contributing nuclides are identified in tables 12.2.1-38 and 12.2.1-39. The system may be placed in operation at approximately 4 h following a shutdown at the maximum shutdown rate. The system removes decay heat from the reactor for the duration of the shutdown. The sources given are maximum values with credit for 4 and 8 h of fission and corrosion product decay and purification.

Core average gamma ray source strengths are given in table 12.2.1-1. These source strengths are used in the evaluation of radiation levels within and around the shutdown reactor. These sources are based on a 3-region core with the regions operated at 300, 600, and 900 effective full-power days, respectively. Spent fuel gamma ray source strengths are given in table 12.2.1-18. These source strengths are used in the evaluation of radiation levels for spent fuel handling, storage, and shipping. These sources are based on 1 core region operated for 900 effective full-power days. All of these sources may be put on a per unit volume of homogenized core basis by multiplying by the power density of 109.2 W/cm^3. Core average and spent fuel neutron source strengths are given in table 12.2.1-40.

The only source materials, byproduct material, or special nuclear material requiring shielding consideration is the neutron source material used in primary and secondary source rods. Each of the two primary source rods contains a californium-252 spontaneous fission neutron source which emits approximately 6×10^8 neutrons/s when initially placed in the reactor core. The gamma ray and neutron source strengths for the secondary source rods, irradiated for 400 days, are given in table 12.2.1-41. The source rods will be stored in the spent fuel pool after use.

The absorber material used in the control rods is hafnium or silver-indium-cadmium. The gamma ray source strengths associated with the absorber material are listed in table 12.2.1-42 for various times after shutdown. The values are based on an irradiation period of 4 years. There are no significant gamma ray sources associated with the BC absorber.

The absorber material used in the burnable poison rods is borosilicate glass. There are no significant gamma ray sources associated with this absorber.

The material used for the control rod cladding, primary source rod cladding, secondary source rod cladding, burnable poison rod cladding, and inner sheath is type 304 stainless steel with a maximum cobalt content of 0.12 weight percent. The gamma ray source strengths associated with the stainless steel are listed in table 12.2.1-43 for various times after shutdown. The values are based on an irradiation time of 15 years.

Irradiated incore detector and drive cable maximum gamma ray source strengths are given in table 12.2.1-44. These source strengths are used in determining shielding requirements when detectors are being moved during or following a flux mapping of the reactor core. These source strengths are given for a detector irradiation period of 30 days and a drive cable irradiation period of 400 days and are given in terms of per cm^3 of detector and drive cable. Irradiated incore detector drive cable average gamma ray source strengths are given in table 12.2.1-45. These source strengths are used in determining shielding requirements when the detectors are not in use and for shipment when the detectors have failed. The values are given in terms of per cm^3 of drive cable after an irradiation period of 400 days. Irradiated incore flux thimble gamma ray source strengths are given in table 12.2.1-46. These source strengths are used in determining shielding requirements during refueling operations when the flux thimbles are withdrawn from the reactor core. The values are given in terms of per cm^3 stainless steel for an irradiation period of 15 years. The flux thimbles are made of type 316 stainless steel with a maximum cobalt impurity content of 0.12 weight percent.

12.2.1.3 Sources for Design Basis Accident

The radiation sources of importance for the design basis accident are the containment source and the residual heat removal and containment spray system sources.

The fission product radiation sources considered to be released from the fuel to the containment following a maximum credible accident are based on the assumptions given in TID-14844.[(2)] These assumptions are consistent with those provided in Regulatory Guide 1.4 and Section II.B.2 of NUREG-0737. The integrated gamma ray and beta particle source strengths for various time periods following the postulated accident are given in table 12.2.1-47.

The residual heat removal system and shielding are designed to allow limited access to the residual heat removal pumps following a design basis accident. The sources are based on the assumptions in TID-14844 with only the nongaseous activity being retained in the sump water, which flows in the residual heat removal loop. Noble gases formed by the decay of halogens in the sump water are assumed to be released to the containment and not retained in the water. Credit has been taken for dilution by the RCS volume plus the contents of the refueling water storage tank. Gamma ray source strengths for radiation sources circulating in the residual heat removal loop and associated equipment are given in table 12.2.1-48.

Isotopic fission product sources from the design basis accident, based on the assumptions in TID-14844, are given in table 12.2.1-49 and chapter 15.

12.2.1.4 Stored Radioactivity

The principal sources of activity not enclosed by the power block buildings are:

- The independent spent fuel storage installation.
- The refueling water storage tank.
- The reactor makeup water storage tank.
- The condensate storage tanks.
- The storage of used spent fuel racks.
- The radwaste processing facility.

- Designated radioactive material storage areas.
- The outage storage building.

The content of these tanks is processed by the spent fuel pool purification system, liquid waste processing system, or boron recycle system until the activity in the fluids is sufficiently low to allow the shielding afforded by the concrete tank walls to result in surface dose rate less than 0.25 mR/h.

Radionuclide inventories of the refueling water storage tank, reactor makeup water storage tank, and condensate storage tanks are presented in tables 12.2.1-50 through 12.2.1-52. There are no other significant amounts of radioactive fluids permanently stored outside the power block and radwaste buildings.

Spent fuel is stored in the spent fuel pool until it is placed in a spent fuel cask for temporary storage in the independent spent fuel storage installation pending transport offsite. Storage space is allocated in the radwaste processing facility for storage of spent filter cartridges and packaged spent resins. Radioactive wastes stored inside plant structures are shielded so that there is Zone I access outside the structures. If it becomes necessary to temporarily store radioactive wastes/ materials outside plant structures, radiation protection measures are to be taken by the radiation protection staff to ensure compliance with 10 CFR 20 and to be consistent with the recommendations of Nuclear Regulatory Commission Regulatory Guide 8.8.

Normally low level contaminated DAW is collected in bags and metal containers and is shipped to an offsite vendor for processing and disposal. In the event that an offsite disposal facility is not available then low level DAW can be stored onsite at various locations. DAW with higher levels of contamination or DAW that cannot be sent offsite due to its construction (noncompressible material, nonburnable material, metals, and so forth) are placed in metal containers and stored onsite at various locations in the owner controlled area. This DAW may also be shipped offsite in the future if an NRC approved facility becomes available.

The outage storage building will store materials with low levels of radioactive contamination that are not classified as radioactive waste. Storage will be controlled to comply with the requirements of 10 CFR 20.

The used spent fuel racks will be stored on site in an area other than those already designated or described. This storage location will comply with the requirements of 10 CFR 20.

12.2.1.5 Field Run Pipe Routing

The procedures for routing of radioactive piping are discussed in paragraph 12.3.1.1.2.

12.2.1.6 References

1. Lutz, R. J., and Chubb, W., "Iodine Spiking - Cause and Effect," ANS Transactions, Vol 28, p 649, June 1978.
2. DiNunno, J. J., et al., "Calculation of Distance Factors for Power and Test Reactor Sites," TID-14844, March 1962.

12.2.2 AIRBORNE RADIOACTIVE MATERIAL SOURCES

This section deals with the models, parameters, and sources required to evaluate airborne concentration of radionuclides during plant operations in various plant radiation areas where personnel occupancy is expected.

These sources include the concentrations of radionuclides in the primary system, secondary system, spent fuel pool, and refueling pool. Sources of airborne radioactive material in equipment cubicles, corridors, or operating areas normally occupied by operating personnel may be obtained from the reactor coolant activities given in section 11.1.

The assumptions and parameters required to evaluate the isotopic airborne concentrations in the various applicable regions are listed in table 12.2.2-1. The chemical and volume control system and the residual heat removal system are designed to provide the capability to purify reactor coolant through the purification demineralizer after reactor shutdown and cooldown. This mode of operation will ensure that the effect of activity spikes does not significantly contribute to the containment airborne activity during refueling operations.

Sources resulting from the removal of the reactor vessel head and the movement of spent fuel are dependent on a number of operating characteristics (e.g., coolant chemistry, fuel performance, etc.) and operating procedures followed during and after shutdown. The permissible coolant activity levels following depressurization are based on the noble gases evolved from the reactor coolant system water upon removal of the reactor vessel head. The endpoint limit for coolant cleanup and degasification is established based on maximum permissible concentration considerations and containment ventilation system capabilities of the plant. Operating plant experience has indicated that coolant Xe-133 concentrations of less than 0.05 μCi/g have posed no problem to operating personnel during vessel head removal.

The exposure rates at the surface of the reactor cavity and spent fuel pool water are dependent on the purification capabilities of the reactor cavity and spent fuel pool cleanup systems. A water total activity level of less than 0.005 μCi/g for the dominant gamma-emitting isotopes at the time of refueling has been shown in operating experience to maintain the dose rate at the water surface at less than 2.5 mrem/h.

The detailed listing of the expected airborne isotopic concentrations in all the various plant regions is presented in table 12.2.2-2. The final design of the plant ensures that all the expected airborne isotopic concentrations in all the applicable regions are well below the maximum permissible concentration for the critical organ for the appropriate isotope for occupational workers, adjusted on the basis of expected weekly occupancy in the regions.

12.2.2.1 Model for Calculating Airborne Concentrations

For those regions characterized by a constant leakrate of the radioactive source at constant source strength and a constant exhaust rate of the region, the peak or equilibrium airborne concentration of the radioisotope in the regions is calculated using the following equation:

$$C_i(t) = \frac{(LR)_i A_i (PF)_i (1 - e\lambda_{Ti} t)}{V_{\lambda_{Ti}}}$$

where:

$(LR)_i$ = leak or evaporation rate of the i^{th} radioisotope in the applicable region (g/s).

A_i = activity concentration of the i^{th} leaking or evaporating radioisotope (μCi/g).

$(PF)_i$ = partition factor or the fraction of the leaking activity that is airborne for the i^{th} radioisotope.

λ_{Ti} = total removal rate constant for the i^{th} radioisotope from the applicable region (s^{-1}).

λ_{Ti} = $\lambda_{di} + \lambda_e$, the removal rate constants in s^{-1} due to radioactive decay for the i^{th} radioisotope and the exhaust from the applicable region, respectively.

t = time elapsed from the start of the leak and the time at which the concentration is evaluated (s).

V = free volume of the region in which the leak occurs (cm^3).

$C_i(t)$ = airborne concentration of the i^{th} radioisotope at time t in the applicable region ($\mu Ci/cm^3$).

From the above equation, it is evident that the peak or equilibrium concentration, $C_{\Sigma gi}$, of the i^{th} radioisotope in the applicable region will be given by the following expression:

$$C_{\Sigma gi} = (LR)_i\, A_i\, (PF) / V\lambda_{Ti}$$

With high exhaust rates, this peak concentration will be reached within a few hours.

12.2.2.2 Sources Resulting from Design Basis Accidents

The radiation sources from design basis accidents include the design basis inventory of radioactive isotopes in the reactor coolant, plus postulated fission product releases from the fuel. Accident parameters and sources are discussed and evaluated in chapter 15.

TABLE 12.2.1-1

CORE AVERAGE GAMMA RAY SOURCE STRENGTHS AT VARIOUS TIMES AFTER SHUTDOWN

Energy Group (MeV/gamma)	Source Strength at Time after Shutdown (MeV/W-s)				
	12 h	24 h	100 h	1 Week	1 Month
0.20 - -.40	1.8×10^{9}	1.5×10^{9}	8.2×10^{8}	5.8×10^{8}	1.4×10^{8}
0.40 - 0.90	1.1×10^{10}	9.4×10^{9}	6.2×10^{9}	5.3×10^{9}	3.3×10^{9}
0.90 - 1.35	1.9×10^{9}	1.2×10^{9}	5.8×10^{8}	4.2×10^{8}	1.2×10^{8}
1.35 - 1.80	3.8×10^{9}	3.3×10^{0}	2.7×10^{9}	2.3×10^{9}	6.6×10^{8}
1.80 - 2.20	2.6×10^{8}	1.8×10^{8}	1.2×10^{8}	9.2×10^{7}	3.7×10^{7}
2.20 - 2.60	2.5×10^{8}	1.9×10^{8}	1.6×10^{8}	1.4×10^{8}	4.0×10^{7}
2.60 - 3.00	6.5×10^{6}	3.3×10^{6}	2.7×10^{6}	2.4×10^{6}	6.9×10^{5}
3.00 - 4.00	4.5×10^{6}	1.3×10^{6}	1.1×10^{6}	9.4×10^{5}	2.7×10^{5}
	3 Month	6 Months	1 Year	5 Years	
0.20 - 0.40	4.6×10^{7}	2.6×10^{7}	1.5×10^{7}	1.6×10^{4}	
0.40 - 0.90	1.8×10^{9}	9.0×10^{8}	3.2×10^{8}	8.0×10^{7}	
0.90 - 1.35	3.0×10^{7}	1.8×10^{7}	1.3×10^{7}	4.8×10^{6}	
1.35 - 1.80	3.8×10^{7}	1.1×10^{7}	8.1×10^{6}	1.3×10^{6}	
1.80 - 2.20	1.6×10^{7}	1.2×10^{7}	7.5×10^{6}	2.2×10^{5}	
2.20 - 2.60	1.5×10^{6}	1.2×10^{4}	0	0	
2.60 - 3.00	2.7×10^{5}	2.0×10^{2}	0	0	
3.00 - 4.00	1.0×10^{4}	0	0	0	

TABLE 12.2.1-2

RADIATION SOURCES
REACTOR COOLANT NITROGEN-16 ACTIVITY

Position in Loop	Loop Transit Time(s)	Nitrogen-16 Activity (μCi/g)
Leaving core	0.0	189
Leaving reactor vessel	1.1	170
Entering steam generator	1.4	164
Leaving steam generator	5.4	112
Entering reactor coolant pump	6.0	106
Entering reactor vessel	6.8	98
Entering core	9.0	86
Leaving core	9.7	189

Nitrogen-16 Energy Emission

Energy (MeV/gamma)	Intensity (%)
1.75	0.13
2.74	9.76
6.13	60.0
7.12	5.0

TABLE 12.2.1-3

NITROGEN-16 RADIATION SOURCES - PRESSURIZER

Discrete Energy (MeV/gamma)	Specific Source Strength (MeV/g-s)
1.75	1.1×10^3
2.74	9.9×10^3
6.13	2.0×10^6
7.12	1.7×10^5

TABLE 12.2.1-4

RADIATION SOURCES - PRESSURIZER

Liquid Phase (1080 ft^3)

Enery Group (MeV/gamma)	Specific Source Strength (MeV/g-s)
0.20 - 0.40	4.7×10^{5}[(a)]
0.40 - 0.90	6.7×10^{5}
0.90 - 1.35	2.9×10^{5}
1.35 - 1.80	1.7×10^{5}
1.80 - 2.20	1.7×10^{5}
2.20 - 2.60	1.8×10^{5}
2.60 - 3.00	2.2×10^{4}
3.00 - 4.00	9.0×10^{3}

Vapor Phase (720 ft^3)

Enery Group (MeV/gamma)	Specific Source Strength (MeV/cm^3-s)
0.20 - 0.40	9.4×10^{5}[(a)]
0.40 - 0.90	8.2×10^{4}
0.90 - 1.35	2.1×10^{3}
1.35 - 1.80	3.4×10^{3}
1.80 - 2.20	4.9×10^{3}
2.20 - 2.60	1.0×10^{4}
2.60 - 3.00	5.3×10^{1}

a. Includes 80-keV Xe-133.

TABLE 12.2.1-5

ISOTOPIC COMPOSITION AND SPECIFIC ACTIVITY OF TYPICAL OUT-OF-CORE CRUD DEPOSITS[a]

Composition (Nuclide)	Activity (μCi/mg) of Deposited Crud for Effective Full Power Years of Plant Operation			
	1 Year	2 Years	5 Years	10 Years
Mn-54	1.0	1.1	1.3	1.4
Fe-59	0.5	0.5	0.5	0.5
Co-58	12.0	12.0	12.0	12.0
Co-60	1.5	2.3	4.0	6.0

a. In addition to corrosion products, about 1.0 μg of mixed actinides and fission products may be present for each 1 g of deposited crud.

TABLE 12.2.1-6 (SHEET 1 OF 15)

RADIATION SOURCES - CVCS
REGENERATIVE HEAT EXCHANGER SOURCE STRENGTHS

	Source Strength (MeV/g-s)	
Energy Group (MeV/gamma)	Tube Side	Shell Side
0.2 - 0.4	3.9×10^{5}[(a)]	4.7×10^{5}[(a)]
0.4 - 0.9	3.3×10^{5}	6.7×10^{5}
0.9 - 1.35	1.6×10^{5}	2.9×10^{5}
1.35 - 1.8	6.9×10^{4}	1.8×10^{5}
1.8 - 2.2	1.1×10^{5}	1.7×10^{5}
2.2 - 2.6	7.1×10^{4}	1.8×10^{5}
2.6 - 3.0	2.1×10^{4}	1.1×10^{5}
3.0 - 4.0	6.2×10^{3}	9.0×10^{3}
6.0 - 7.0	-	1.7×10^{7}
7.0 - 7.5	-	1.4×10^{6}

TABLE 12.2.1-6 (SHEET 2 OF 15)

RADIATION SOURCES - CVCS
REGENERATIVE HEAT EXCHANGER SPECIFIC ACTIVITY

	Activity (μCi/g)	
Nuclide	Tube Side	Shell Side
Kr-88	1.3	3.7
Kr-89	-	1.1×10^{-1}
Xe-133	2.6×10^{2}	2.7×10^{2}
Xe-135	5.3	7.3
I-132	-	2.8
I-133	-	4.2
I-135	-	2.3
Rb-88	4.8	4.8
Cs-134	2.3	2.3
Cs-136	2.9	2.9
Cs-138	9.6×10^{-1}	9.6×10^{-1}
N-16	-	1.1×10^{2}

TABLE 12.2.1-6 (SHEET 3 OF 15)

RADIATION SOURCES - CVCS
EXCESS LETDOWN HEAT EXCHANGER (TUBE SIDE)
SOURCE STRENGTHS

Energy Group (MeV/gamma)	Source Strength (MeV/g-s)
0.2 - 0.4	4.7×10^{5}[(a)]
0.4 - 0.9	6.7×10^{5}
0.9 - 1.35	2.9×10^{5}
1.35 - 1.8	1.8×10^{5}
1.8 - 2.2	1.7×10^{5}
2.2 - 2.6	1.8×10^{5}
2.6 - 3.0	1.1×10^{5}
3.0 - 4.0	9.0×10^{3}
6.0 - 7.0	1.7×10^{7}
7.0 - 7.5	1.4×10^{6}

TABLE 12.2.1-6 (SHEET 4 OF 15)

RADIATION SOURCES - CVCS
EXCESS LETDOWN HEAT EXCHANGER (TUBE SIDE)
SPECIFIC ACTIVITY

Nuclide	Activity (μCi/g)
Kr-88	3.7
Kr-89	1.1×10^{-1}
Xe-133	2.7×10^{2}
Xe-135	7.3
I-132	2.8
I-133	4.2
I-135	2.3
Rb-88	4.8
Cs-134	2.3
Cs-136	2.9
Cs-138	9.6×10^{-1}
N-16	1.1×10^{2}

TABLE 12.2.1-6 (SHEET 5 OF 15)

RADIATION SOURCES - CVCS
LETDOWN HEAT EXCHANGER AND SEAL WATER HEAT
EXCHANGER (TUBE SIDE) SOURCE STRENGTHS

Energy Group (MeV/gamma)	Source Strength (MeV/cm^3-s)
0.2 - 0.4	4.7×10^5 (a)
0.4 - 0.0	6.7×10^5
0.9 - 1.35	2.9×10^5
1.35 - 1.8	1.7×10^5
1.8 - 2.2	1.7×10^5
2.2 - 2.6	1.8×10^5
2.6 - 3.0	2.2×10^4
3.0 - 4.0	9.0×10^3

TABLE 12.2.1-6 (SHEET 6 OF 15)

RADIATION SOURCES - CVCS
LETDOWN HEAT EXCHANGER AND SEAL WATER HEAT
EXCHANGER (TUBE SIDE) SPECIFIC ACTIVITY

Nuclide	Activity (µCi/g)
Kr-88	3.7
Kr-89	1.1×10^{-1}
Xe-133	2.7×10^{2}
Xe-135	7.3
I-132	2.8
I-133	4.2
I-135	2.3
Rb-88	4.8
Cs-134	2.3
Cs-136	2.9
Cs-138	9.6×10^{-1}

TABLE 12.2.1-6 (SHEET 7 OF 15)

RADIATION SOURCES - CVCS
LETDOWN REHEAT HEAT EXCHANGER SOURCE STRENGTHS

	Source Strength (MeV/g-s)	
Energy Group (MeV/gamma)	Tube Side	Shell Side
0.2 - 0.4	4.7×10^{5}[(a)]	4.3×10^{5}[(a)]
0.4 - 0.9	6.7×10^{5}	3.7×10^{5}
0.9 - 1.35	2.9×10^{5}	1.8×10^{5}
1.35 - 1.8	1.7×10^{5}	1.1×10^{5}
1.8 - 2.2	1.7×10^{5}	1.7×10^{5}
2.2 - 2.6	1.8×10^{5}	1.8×10^{5}
2.6 - 3.0	2.2×10^{4}	2.2×10^{4}
3.0 - 4.0	9.0×10^{3}	8.2×10^{3}

TABLE 12.2.1-6 (SHEET 8 OF 15)

RADIATION SOURCES - CVCS
LETDOWN REHEAT HEAT EXCHANGER SPECIFIC ACTIVITY

	Activity (µCi/g)	
Nuclide	Tube Side	Shell Side
Kr-88	3.7	3.7
Kr-89	1.1×10^{-1}	1.1×10^{-1}
Xe-133	2.7×10^{2}	2.7×10^{2}
Xe-135	7.3	7.3
I-132	2.8	-
I-133	4.2	-
I-135	2.3	-
Rb-88	4.8	4.8
Cs-134	2.3	2.3
Cs-136	2.9	2.9
Cs-138	9.6×10^{-1}	9.6×10^{-1}

TABLE 12.2.1-6 (SHEET 9 OF 15)

RADIATION SOURCES - CVCS
LETDOWN CHILLER HEAT EXCHANGER (TUBE SIDE) AND
MODERATING HEAT EXCHANGER SOURCE STRENGTHS

Energy Group (MeV/gamma)	Source Strength (MeV/cm^3-s)
0.2 - 0.4	4.3×10^{5}[(a)]
0.4 - 0.9	3.7×10^{5}
0.9 - 1.35	1.8×10^{5}
1.35 - 1.8	1.1×10^{5}
1.8 - 2.2	1.7×10^{5}
2.2 - 2.6	1.8×10^{5}
2.6 - 3.0	2.2×10^{4}
3.0 - 4.0	8.2×10^{3}

LETDOWN CHILLER HEAT EXCHANGER (TUBE SIDE) AND
MODERATING HEAT EXCHANGER SPECIFIC ACTIVITY

Nuclide	Activity (µCi/g)
Kr-88	3.7
Kr-89	1.1×10^{-1}
Xe-133	2.7×10^{2}
Xe-135	7.3
Rb-88	4.8
Cs-134	2.3
Cs-136	2.9
Cs-138	9.6×10^{-1}

TABLE 12.2.1-6 (SHEET 10 OF 15)

RADIATION SOURCES - CVCS
MIXED BED DEMINERALIZER SOURCE STRENGTHS
FOR 30 FT^3 OF RESIN

Energy Group (MeV/gamma)	Source Strength (MeV/cm^3-s)
0.2 - 0.4	1.8×10^8
0.4 - 0.9	7.7×10^8
0.9 - 1.35	7.7×10^7
1.35 - 1.8	2.6×10^7
1.8 - 2.2	1.4×10^6
2.2 - 2.6	8.4×10^5
2.6 - 3.0	1.9×10^5
3.0 - 4.0	5.7×10^4

MIXED BED DEMINERALIZER SPECIFIC ACTIVITY

Nuclide	Activity ($\mu Ci/cm^3$)
Br-84	6.5×10^{-1}
I-131	1.4×10^4
I-132	1.7×10^2
I-135	3.9×10^2
Rb-88	3.3×10^1
Cs-134	8.5×10^3
Cs-136	7.8×10^2
Ba-137m	6.7×10^3
Cs-138	1.3×10^1
Co-60	1.5×10^2
La-140	3.4×10^1

TABLE 12.2.1-6 (SHEET 11 OF 15)

RADIATION SOURCES - CVCS
CATION BED DEMINERALIZER SOURCE STRENGTHS
FOR 20 FT^3 OF RESIN

Energy Group (MeV/gamma)	Source Strength (MeV/cm^3-s)
0.2 - 0.4	1.1×10^7
0.4 - 0.9	9.8×10^8
0.9 - 1.35	6.5×10^7
1.35 - 1.8	2.3×10^7
1.8 - 2.2	8.3×10^5
2.2 - 2.6	2.8×10^5
2.6 - 3.0	2.8×10^5
3.0 - 4.0	6.6×10^4

CATION BED DEMINERALIZER SPECIFIC ACTIVITY

Nuclide	Activity ($\mu Ci/cm^3$)
Rb-88	5.0×10^1
Cs-134	1.3×10^4
Cs-136	1.2×10^3
Cs-138	1.9×10^1
Ba-137m	1.0×10^4

TABLE 12.2.1-6 (SHEET 12 OF 15)

RADIATION SOURCES - CVCS
BORON THERMAL REGENERATION DEMINERALIZER SOURCE STRENGTHS
FOR 150 FT^3 OF RESIN

Energy Group (MeV/gamma)	Source Strength (MeV/cm^3-s)
0.2 - 0.4	2.7×10^6
0.4 - 0.9	1.8×10^6
0.9 - 1.35	4.4×10^5
1.35 - 1.8	2.4×10^5
1.8 - 2.2	2.1×10^4
2.2 - 2.6	1.4×10^4

BORON THERMAL REGENERATION DEMINERALIZER
SPECIFIC ACTIVITY

Nuclide	Activity (μCi/cm^3)
I-131	2.3×10^2
I-132	4.4
I-133	3.7×10^1
I-135	1.0×10^1

TABLE 12.2.1-6 (SHEET 13 OF 15)

RADIATION SOURCES - CVCS
CHEMICAL AND VOLUME CONTROL SYSTEM FILTER
SOURCE STRENGTHS ASSUMED TO ESTABLISH
SHIELDING REQUIREMENTS

Reactor Coolant Filter

Energy Group (MeV/gamma)	Source Strength (MeV/cm^3-s)
0.4 - 0.9	5.7×10^7
0.9 - 1.35	1.5×10^7

Seal Water Injection Filter

Energy Group (MeV/gamma)	Source Strength (MeV/cm^3-s)
0.4 - 0.9	4.8×10^7
0.9 - 1.35	1.2×10^7

Seal Water Return Filter

Energy Group (MeV/gamma)	Source Strength (MeV/cm^3-s)
0.4 - 0.9	1.1×10^7
0.9 - 1.35	3.0×10^6

DIMENSIONS AND COMPOSITION OF CVCS FILTERS
ASSUMED TO ESTABLISH SHIELDING REQUIREMENTS

Filter	Source Dimensions (in.)	Source Composition volume %
Reactor coolant and seal water return	Radius - 3.375 Length - 19	Air – 62 Water - 38
Seal water injection	Radius - 1.3 Length - 20	Air - 90 Stainless steel - 10

TABLE 12.2.1-6 (SHEET 14 OF 15)

RADIATION SOURCES - CVCS
VOLUME CONTROL TANK LIQUID PHASE SOURCE STRENGTHS
FOR A 200-FT^3 LIQUID PHASE

Energy Group (MeV/gamma)	Source Strength (MeV/cm^3-s)
0.2 - 0.4	3.9×10^{5}[(a)]
0.4 - 0.9	3.3×10^{5}
0.9 - 1.35	1.6×10^{5}
1.35 - 1.8	6.9×10^{4}
1.8 - 2.2	1.1×10^{5}
2.2 - 2.6	7.1×10^{4}
2.6 - 3.0	2.1×10^{4}
3.0 - 4.0	6.2×10^{3}

TABLE 12.2.1-6 (SHEET 15 OF 15)

RADIATION SOURCES - CVCS
VOLUME CONTROL TANK VAPOR PHASE SOURCE STRENGTHS
FOR A 200-FT^3 VAPOR PHASE

Energy Group (MeV/gamma)	Source Strength (MeV/cm^3-s)
0.2 - 0.4	5.6×10^6[(a)]
0.4 - 0.9	4.8×10^5
0.9 - 1.35	1.1×10^5
1.35 - 1.8	3.3×10^5
1.8 - 2.2	5.9×10^5
2.2 - 2.6	1.2×10^6
2.6 - 3.0	6.8×10^3
3.0 - 4.0	3.3×10^3

a. Includes 80-keV Xe-133.

TABLE 12.2.1-7

STEAM GENERATOR BLOWDOWN DEMINERALIZER SOURCE STRENGTHS FOR 75 FT^3 OF RESIN

Energy Group (MeV/gamma)	Source Strength (MeV/cm^3-s)
0.2 - 0.4	1.3×10^6
0.4 - 0.9	1.3×10^8
0.9 - 1.35	8.0×10^6
1.35 - 1.8	3.1×10^6
1.8 - 2.2	8.5×10^3
2.2 - 2.6	2.3×10^3

TABLE 12.2.1-8

STEAM GENERATOR BLOWDOWN DEMINERALIZER
SPECIFIC ACTIVITY

Nuclide	Activity ($\mu Ci/cm^3$)
I-131	7.8×10^{1}
I-132	3.5
I-135	1.1
Cs-134	1.8×10^{3}
Cs-136	1.3×10^{2}
Ba-137m	1.3×10^{3}
La-140	2.0×10^{-1}

TABLE 12.2.1-9

STEAM GENERATOR BLOWDOWN SPENT RESIN STORAGE TANK
SOURCE STRENGTHS

Energy Group (MeV/gamma)	Source Strength Resin	Liquid (MeV/g-s)
0.2 - 0.4	1.3×10^{6}	1.3×10^{2}
0.4 - 0.9	1.3×10^{8}	1.3×10^{4}
0.9 - 1.35	8.0×10^{6}	8.0×10^{2}
1.35 - 1.8	3.1×10^{6}	3.1×10^{2}
1.8 - 2.2	8.5×10^{3}	8.5×10^{-1}
2.2 - 2.6	2.3×10^{3}	2.3×10^{-1}

TABLE 12.2.1-10

STEAM GENERATOR BLOWDOWN SPENT RESIN STORAGE TANK
SPECIFIC ACTIVITY

	Activity (μCi/g)	
Nuclide	Resin	Liquid
I-131	7.8×10^{1}	7.8×10^{-3}
I-132	3.5	3.5×10^{-4}
I-135	1.1×10^{-1}	1.1×10^{-5}
Cs-134	1.8×10^{3}	1.8×10^{-1}
Cs-136	1.3×10^{2}	1.3×10^{-2}
Ba-137m	1.3×10^{3}	1.3×10^{-1}
La-140	2.0×10^{-1}	2.0×10^{-5}

TABLE 12.2.1-11

STEAM GENERATOR BLOWDOWN PREFILTER SOURCE STRENGTHS

Energy Group (MeV/gamma)	Source Strength (MeV/cm^3-s)
0.2 - 0.4	1.4×10^2
0.4 - 0.9	1.9×10^3
0.9 - 1.35	8.7×10^2
1.35 - 1.8	1.3×10^2
1.8 - 2.2	2.2×10^1
2.2 - 2.6	8.6

TABLE 12.2.1-12

STEAM GENERATOR BLOWDOWN OUTLET AND SPENT RESIN SLUICE FILTER SOURCE STRENGTHS

Energy Group (MeV/gamma)	Source Strength (MeV/cm^3-s)
0.2 - 0.4	1.3×10^{6}
0.4 - 0.9	1.3×10^{8}
0.9 - 1.35	8.0×10^{6}
1.35 - 1.8	3.1×10^{6}
1.8 - 2.2	8.5×10^{3}
2.2 - 2.6	2.3×10^{3}

TABLE 12.2.1-13

DIMENSIONS AND COMPOSITION OF STEAM GENERATOR BLOWDOWN PURIFICATION SYSTEM FILTERS

Filter	Source Dimensions (in.)	Source Composition (volume %)
Blowdown prefilter[a]	Radius - 5.0	Air - 66
	Length - 36	Water - 32
		Stainless steel - 2
Blowdown outlet and spent resin sluice	Radius - 3.375	Air - 62
	Length - 19	Water - 38

a. These are the parameters assumed to establish shielding requirements.

TABLE 12.2.1-14

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TABLE 12.2.1-15

RECYCLE HOLDUP TANK SPECIFIC ACTIVITIES

Nuclide[(a)]	Liquid Phase 112,000 gal (μCi/g)	Vapor Phase 500 ft^3 (μCi/cm^3)
Kr-83m	-	1.1
Kr-85m	-	4.7
Kr-85	-	1.7×10^{1}
Kr-87	1.3	2.9
Kr-88	3.7	8.5
Kr-89	1.1×10^{-1}	2.5×10^{-1}
Xe-131m	-	5.1
Xe-133m	-	4.0×10^{1}
Xe-133	2.7×10^{2}	6.2×10^{2}
Xe-135m	-	1.1
Xe-135	7.3	1.7×10^{1}
Xe-137	-	4.0×10^{-1}
Xe-138	6.4×10^{-1}	1.5
I-132	2.8×10^{-1}	-
I-135	2.3×10^{-1}	-
Rb-88	4.8×10^{-1}	-
Cs-134	-	-
Cs-136	2.9×10^{-1}	-
Cs-138	9.6×10^{-2}	-

a. The nuclides listed by activity are the significant contributors to the source strengths and represent greater than 1 percent of gamma source strengths.

TABLE 12.2.1-16

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TABLE 12.2.1-17

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TABLE 12.2.1-18

SPENT FUEL GAMMA RAY SOURCE STRENGTHS AT VARIOUS TIMES AFTER SHUTDOWN

Energy Group (MeV/gamma)	Source Strength at Time after Shutdown (MeV/W-s)				
	12 h	24 h	100 h	1 Week	1 Month
0.20 - 0.40	1.8×10^{9}	1.6×10^{9}	8.4×10^{5}	6.0×10^{5}	1.6×10^{8}
0.40 - 0.90	1.1×10^{10}	9.7×10^{9}	6.5×10^{9}	5.6×10^{9}	3.6×10^{9}
0.90 - 1.35	2.0×10^{9}	1.4×10^{9}	7.0×10^{8}	5.3×10^{8}	1.6×10^{8}
1.35 - 1.80	3.6×10^{9}	3.2×10^{9}	2.6×10^{9}	2.2×10^{9}	6.4×10^{8}
1.80 - 2.20	3.2×10^{8}	2.5×10^{8}	1.7×10^{8}	1.4×10^{8}	5.7×10^{7}
2.20 - 2.60	2.4×10^{8}	1.8×10^{8}	1.5×10^{8}	1.3×10^{8}	3.8×10^{7}
2.60 - 3.00	5.9×10^{6}	3.2×10^{6}	2.6×10^{8}	2.3×10^{6}	6.6×10^{5}
3.00 - 4.00	4.2×10^{6}	1.2×10^{6}	1.0×10^{6}	9.0×10^{5}	2.6×10^{5}
	3 Months	6 Months	1 Year	5 Years	
0.20 - 0.40	5.3×10^{7}	3.1×10^{7}	1.8×10^{7}	2.2×10^{6}	
0.40 - 0.90	2.0×10^{9}	1.1×10^{9}	4.8×10^{8}	1.3×10^{8}	
0.90 - 1.35	4.6×10^{7}	3.0×10^{7}	2.3×10^{7}	8.7×10^{6}	
1.35 - 1.80	4.6×10^{7}	1.8×10^{7}	1.3×10^{7}	2.4×10^{6}	
1.80 - 2.20	2.0×10^{7}	1.4×10^{7}	8.8×10^{6}	2.5×10^{5}	
2.20 - 2.60	1.5×10^{4}	1.1×10^{4}	0	0	
2.60 - 3.00	2.6×10^{4}	2.0×10^{2}	0	0	
3.00 - 4.00	1.0×10^{4}	0	0	0	

TABLE 12.2.1-19 (SHEET 1 OF 2)

SPENT FUEL POOL SPECIFIC ACTIVITIES
FOR DETERMINING SHIELD WALL THICKNESSES[(a)]

Nuclide	Concentration for Maximum Failed Fuel (µCi/g)	Concentration for Expected Failed Fuel (µCi/g)
I-130	4.4901-11	4.9425-12
I-131	3.9336-05	3.9248-06
I-133	4.2069-07	4.0335-08
Rb-86	4.3508-07	1.7611-09
Cs-134	5.8617-05	6.4578-07
Cs-136	5.1219-05	2.4711-07
Cs-137	3.8453-05	4.8762-07
Cr-51	2.4155-04	2.4155-04
Mn-54	5.0467-05	5.0467-05
Fe-55	5.1036-08	4.0749-08
Fe-59	2.4597-05	2.4597-05
Co-58	4.9747-04	4.9747-04
Co-60	7.5971-05	7.5971-05
Sr-89	1.0019-07	8.2401-09
Sr-90	3.0765-09	2.5874-10
Y-90	1.4140-10	5.1408-12
Y-91	1.3462-08	1.5375-09
Zr-95	2.4832-05	2.4832-05

TABLE 12.2.1-19 (SHEET 2 OF 2)

Nuclide	Concentration for Maximum Failed Fuel (μCi/g)	Concentration for Expected Failed Fuel (μCi/g)
Nb-95	1.4518-08	1.1500-09
Mo-99	3.3173-06	3.6894-07
Ru-103	2.4432-05	2.4432-05
Ru-106	3.5438-09	2.5288-10
Te-125m	6.6049-09	6.8091-10
Te-127m	7.1135-08	6.8825-09
Te-129m	4.2167-07	3.1090-08
Te-131m	1.3749-08	1.3754-09
Te-132	1.6719-06	1.6204-07
Ba-140	7.3745-08	3.8577-09
La-140	1.9926-09	2.2546-10
Ce-141	1.4593-06	1.4593-06
Ce-143	4.2130-10	3.2189-11
Ce-144	7.5669-07	7.5669-07
Pr-143	1.1310-08	9.0264-10
Np239	--	3.9583-09
Ag-110m	3.5226-08	--

a. These activities are used to verify shield wall thicknesses. For dose assessment (section 12.4), activities in the pool are assumed to be limited administratively so that pool surface dose rates are less than 2.5 mrem/h.

TABLE 12.2.1-20

SPENT FUEL POOL DEMINERALIZER SOURCE STRENGTHS
FOR 30 FT^3 OF RESIN

Energy Group (MeV/gamma)	Source Strength (MeV/cm^3-s)
0.2 - 0.4	2.5×10^4
0.4 - 0.9	1.6×10^6
0.9 - 1.35	7.5×10^5
1.35 - 1.8	1.5×10^4

TABLE 12.2.1-21

SPENT FUEL POOL DEMINERALIZER SPECIFIC ACTIVITY

Nuclide	Activity (μCi/cm^3)
Cr-51	23.0
Mn-54	4.5
Fe-59	2.3
Co-58	45.0
Co-60	7.0
Zr-95	2.3
Ru-103	2.3

TABLE 12.2.1-22

SPENT FUEL POOL FILTER SOURCE STRENGTHS

Energy Group (MeV/gamma)	Source Strength (MeV/cm^3-s)
0.4 - 0.9	1.1×10^7
0.9 - 1.35	3.0×10^6

TABLE 12.2.1-23

SPENT FUEL POOL FILTER DIMENSIONS AND COMPOSITION
ASSUMED TO ESTABLISH SHIELDING REQUIREMENTS

Filter	Source Dimensions (in.)	Source Composition (volume %)
Spent fuel pool	Radius - 3.375	Air - 62
	Length - 19	Water - 38

TABLE 12.2.1-24

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TABLE 12.2.1-25

WASTE MONITOR TANK DEMINERALIZER
SPECIFIC ACTIVITY

Nuclide	Activity ($\mu Ci/cm^3$)
I-131	1.2×10^2
I-132	2.0×10^1
I-133	1.0×10^2
I-135	3.0×10^1
Cs-134	1.0×10^2
Cs-136	1.2×10^2

TABLE 12.2.1-26

REACTOR COOLANT DRAIN TANK LIQUID PHASE SPECIFIC ACTIVITY

Nuclide	Activity (μCi/g)
Kr-88	3.7
Kr-89	1.1×10^{-1}
Xe-133	2.7×10^{2}
Xe-135	7.3
I-132	2.8
I-138	4.2
I-135	2.3
Rb-88	4.8
Cs-134	2.3
Cs-136	2.9
Cs-138	9.6×10^{-1}

TABLE 12.2.1-27

REACTOR COOLANT DRAIN TANK VAPOR PHASE SPECIFIC ACTIVITY

Nuclide	Activity ($\mu Ci/cm^3$)
Kr-85m	8.7×10^{-1}
Kr-85	1.6×10^{2}
Kr-87	1.6×10^{-1}
Kr-88	9.9×10^{-1}
Xe-131m	2.1×10^{1}
Xe-133m	6.6×10^{1}
Xe-133	1.8×10^{3}
Xe-135	6.0

TABLE 12.2.1-28

FLOOR DRAIN TANK, WASTE MONITOR TANKS, AND
WASTE HOLDUP TANK SPECIFIC ACTIVITY

Nuclide	Activity (μCi/g)
I-131	2.8
I-132	2.8
I-133	4.2
I-135	2.3
Cs-134	2.3
Cs-136	2.9

TABLE 12.2.1-29

CHEMICAL DRAIN TANK SPECIFIC ACTIVITY

Nuclide	Activity (μCi/g)
Co-60	1.9×10^{-3}
Cs-134	1.1
Ba-137m	7.0×10^{-1}

TABLE 12.2.1-30

SPENT RESIN STORAGE TANK SPECIFIC ACTIVITY

	Activity (μCi/g)	
Nuclide	Resin	Liquid
Br-84	6.5×10^{-1}	6.5×10^{-5}
I-131	1.4×10^{4}	1.4
I-132	1.7×10^{2}	1.7×10^{-2}
I-135	3.9×10^{2}	3.9×10^{-2}
Rb-88	3.3×10^{1}	3.3×10^{-3}
Cs-134	8.5×10^{3}	8.5×10^{-1}
Cs-136	7.8×10^{2}	7.8×10^{-2}
Ba-137m	6.7×10^{3}	6.7×10^{-1}
Cs-138	1.3×10^{1}	1.3×10^{-3}
Co-60	1.5×10^{2}	1.5×10^{-2}
La-140	3.4×10^{1}	3.4×10^{-3}

TABLE 12.2.1-31

DELETED

TABLE 12.2.1-32

DELETED

TABLE 12.2.1-33

RADIOACTIVE GAS DISTRIBUTION - GASEOUS RADWASTE SYSTEM OPERATING

Nuclide	Reactor Coolant Activity (Ci/g)	Reactor Coolant Inventory (Ci)	Volume Control Tank Inventory (Ci)	Gas Decay Tank Inventory (Ci)	Total Ci
Kr-83m	4.5×10^{-2}	1.0×10^{2}	1.5×10^{1}	4.5	1.2×10^{2}
Kr-85	5.9×10^{-2}	1.4×10^{2}	4.0	[a]	[a]
Kr-85m	1.9	4.4×10^{2}	8.1×10^{1}	5.8×10^{1}	5.8×10^{2}
Kr-87	1.3	2.9×10^{2}	2.7×10^{1}	5.4	3.2×10^{2}
Kr-88	3.6	8.2×10^{2}	1.2×10^{2}	5.5×10^{1}	1.0×10^{3}
Kr-89	1.1×10^{-1}	2.4×10^{1}	1.4×10^{1}	1.1×10^{-3}	2.4×10^{1}
Xe-131m	2.5×10^{-1}	5.7×10^{1}	1.4×10^{1}	6.2×10^{2}	6.9×10^{2}
Xe-133	6.2×10^{-1}	1.4×10^{1}	3.3×10^{3}	6.6×10^{4}	8.3×10^{2}
Xe-133m	7.4	1.7×10^{3}	3.9×10^{2}	3.3×10^{3}	5.4×10^{3}
Xe-135	6.1	1.4×10^{3}	3.3×10^{2}	4.8×10^{2}	2.2×10^{3}
Xe-135m	4.8×10^{-1}	1.1×10^{2}	4.6×10^{1}	1.9	1.6×10^{2}
Xe-137	1.7×10^{-1}	4.0×10^{1}	2.7×10^{-1}	2.7×10^{-3}	4.0×10^{1}
Xe-138	6.4×10^{-1}	1.5×10^{2}	3.4	1.3×10^{-1}	1.5×10^{2}

a. The Kr-85 activity does not reach equilibrium. (See table 12.2.1-34.)

TABLE 12.2.1-34

VOLUME CONTROL TANK NOBLE GAS STRIPPING FRACTIONS[a]

Nuclide	Stripping Fraction
Kr-83m	0.78
Kr-85m	0.65
Kr-85	0.43
Kr-87	0.83
Kr-88	0.72
Kr-89	0.99
Xe-131m	0.34
Xe-133m	0.37
Xe-133	0.35
Xe-135m	0.93
Xe-135	0.47
Xe-137	0.98
Xe-138	0.94

a. Assuming 0.7-sf^3/min volume control tank purge rate.

TABLE 12.2.1-35

WASTE GAS PROCESSING SYSTEM Kr-85 INVENTORY

Time (effective full-power years)	GRS Kr-85 Inventory (Ci)
1	2.7×10^3
2	5.2×10^3
4	1.0×10^4
6	1.3×10^4
10	2.0×10^4
20	3.1×10^4
30	3.7×10^4
40[a]	3.9×10^4

[a] The renewed operating licenses authorized a 20-year period of extended operation for both VEGP units, resulting in a total plant operating life of 60 years. Since the inventory in the Waste Gas Decay Tanks (WGDTs) has been routinely released during the first 20 years of operation and is expected to continue to be routinely released during future operation, the inventory of the WGDTs accumulated during the first 20 years of operation will be released prior to entering the period of extended operation. Therefore, the stated design capacity of the GWPS remains sufficient, and the analysis of the maximum fission product inventory in the GWPS over a 40-year plant life remains bounding for a 60-year plant life.

TABLE 12.2.1-36

HYDROGEN RECOMBINER, WASTE GAS COMPRESSOR,
AND GAS DECAY TANK SPECIFIC ACTIVITY

Nuclide	Activity ($\mu Ci/cm^3$)
Kr-85m	9.5
Kr-85	8.0×10^1
Kr-87	1.3
Kr-88	1.2×10^1
Xe-131m	2.0×10^1
Xe-133m	1.5×10^2
Xe-133	2.3×10^3
Xe-135	4.9×10^1

TABLE 12.2.1-37

RADIATION SOURCES
RESIDUAL HEAT REMOVAL SYSTEM

Energy Group (MeV/gamma)	Specific Source Strength (MeV/g-s) 4 h after Shutdown	8 h after Shutdown
0.20 - 0.40	4.0×10^{5}[(a)]	3.5×10^{5}[(a)]
0.40 - 0.90	2.9×10^{5}	1.7×10^{5}
0.90 - 1.35	1.3×10^{5}	7.3×10^{4}
1.35 - 1.80	3.9×10^{4}	1.5×10^{4}
1.80 - 2.20	4.0×10^{4}	1.1×10^{4}
2.20 - 2.60	4.2×10^{4}	1.1×10^{4}
2.60 - 3.00	3.2×10^{3}	8.5×10^{2}
3.00 - 4.00	1.2×10^{3}	3.3×10^{2}

a. Includes 80 keV Xe-133.

TABLE 12.2.1-38

RESIDUAL HEAT REMOVAL LOOP SPECIFIC
ACTIVITY - 4 HOURS AFTER SHUTDOWN

Nuclide	Activity (μCi/g)
Kr-88	1.0
Xe-133	2.6×10^{2}
Xe-135	5.0
I-132	7.2×10^{-1}
I-133	2.4
I-135	9.7×10^{-1}
Rb-88	1.1
Cs-134	1.5
Cs-136	1.9

TABLE 12.2.1-39

RESIDUAL HEAT REMOVAL LOOP SPECIFIC ACTIVITY - 8 HOURS AFTER SHUTDOWN

Nuclide	Activity (μCi/g)
Kr-88	2.7×10^{-1}
Xe-133	2.5×10^{2}
I-132	3.0×10^{-1}
I-133	1.4
I-135	4.1×10^{-1}
Cs-134	9.7×10^{-1}
Cs-136	1.2
Rb-88	3.0×10^{-1}

TABLE 12.2.1-40

CORE AVERAGE AND SPENT FUEL NEUTRON SOURCE STRENGTHS AT VARIOUS TIMES AFTER SHUTDOWN[a]

Time after Shutdown	Core Average (neut/W-s)	Spent Fuel (neut/W-s)
12 h	9.6	24
24 h	9.6	24
100 h	9.5	24
1 week	9.5	23
1 month	9.1	23
3 months	8.3	21
6 months	7.4	19
1 year	6.3	16
5 years	4.8	12

a. Eighty-three to 93 percent of the neutron source strength is due to the spontaneous fission of curium-242 and curium-244. The curium spontaneous fission neutron spectrum is quite similar to that of californium-252. The californium-252 spontaneous fission neutron spectrum may be expressed by the Watt formula as follows:

$$\chi(E) = 0.37 \exp(-0.88E) \sinh\left(\sqrt{2.0\,E}\right)$$

where E is the neutron energy and x(E) is normalized so that

$$\int_0^\infty \chi(E)\, dE = 1$$

TABLE 12.2.1-41

IRRADIATED Sb-Be SECONDARY SOURCE ROD SOURCE STRENGTHS

Energy Group (MeV/gamma)	Gamma Ray Source Strength at Time after Shutdown (MeV/cm -s) 1 Day	1 Week	1 Month	6 Months	1 Year	5 Years
0.20 - 0.40	3.0×10^{10}	2.9×10^{10}	2.5×10^{10}	1.1×10^{10}	3.7×10^{9}	2.2×10^{7}
0.40 - 0.90	1.1×10^{13}	7.0×10^{12}	4.6×10^{12}	8.1×10^{11}	9.7×10^{10}	1.8×10^{8}
0.90 - 1.35	6.7×10^{11}	4.8×10^{11}	3.4×10^{11}	6.0×10^{10}	7.0×10^{9}	-
1.35 - 1.80	7.6×10^{12}	7.1×10^{12}	5.5×10^{11}	9.7×10^{11}	1.2×10^{11}	-
1.80 - 2.20	9.8×10^{11}	9.1×10^{11}	7.0×10^{11}	1.2×10^{10}	1.5×10^{10}	-

Neutron Source Strength at Time after Shutdown (neut/cm -s)

1 Day	1 Week	1 Month	6 Months	1 Year	5 Years
4.5×10^{8}	4.2×10^{8}	3.2×10^{8}	5.8×10^{7}	6.8×10^{6}	-

a. The secondary source cross-sectional area is 0.582 cm^2 per rod. The Sb-Be material density is 3.38 g/cm^3.

b. The secondary source rod cross-sectional area is 0.582 cm^2 per rod. The average neutron energy is 30 keV. The Sb-Be material density is 3.38 g/cm^3.

TABLE 12.2.1-42 (SHEET 1 of 2)

CONTROL ROD SOURCE STRENGTHS

Hafnium Control Rod[(a)]

400-Day Irradiation

Source Strength at Time after Shutdown (MeV/cm^3 -s)

Energy Group (MeV)	1 Day	1 Week	1 Month	6 Months	1 Year
0.20 - 0.40	2.2×10^{10}	2.0×10^{10}	1.4×10^{10}	1.3×10^{9}	1.0×10^{8}
0.40 - 0.90	1.9×10^{11}	1.7×10^{11}	1.2×10^{11}	1.0×10^{10}	5.0×10^{8}
0.90 - 1.35	2.6×10^{10}	2.5×10^{10}	2.2×10^{9}	8.9×10^{9}	2.9×10^{9}

15-Year Irradiation

Source Strength at Time after Shutdown (MeV/cm^3 -s)

Energy Group (MeV)	1 Day	1 Week	1 Month	6 Months	1 Year
0.20 - 0.40	4.7×10^{10}	4.3×10^{10}	2.9×10^{10}	3.8×10^{9}	6.2×10^{8}
0.40 - 0.90	3.5×10^{11}	3.2×10^{11}	2.2×10^{11}	1.9×10^{10}	9.3×10^{8}
0.90 - 1.35	2.8×10^{11}	2.7×10^{11}	2.4×10^{11}	9.7×10^{10}	3.1×10^{10}

TABLE 12.2.1-42 (SHEET 2 of 2)

Irradiated Ag-In-Cd Control Rod [b,c]

Energy Group (MeV/gamma)	Source Strength at Time after Shutdown (MeV/cm^3 -s) 1 Day	1 Week	1 Month	6 Months	1 Year	5 Years
0.20 - 0.40	2.3×10^8	2.3×10^8	2.2×10^8	1.4×10^8	8.5×10^7	1.5×10^6
0.40 - 0.90	1.1×10^{12}	1.1×10^{12}	1.0×10^{12}	6.6×10^{11}	4.0×10^{11}	7.1×10^9
0.90 - 1.35	2.0×10^{11}	1.9×10^{11}	1.8×10^{11}	1.2×10^{11}	7.2×10^{10}	1.3×10^9
1.35 - 1.80	3.7×10^{11}	3.7×10^{11}	3.4×10^{11}	2.3×10^{11}	1.4×10^{11}	2.5×10^9

a. Source strengths are expressed per cm^3 of absorber. Density of the hafnium absorber is 13.31 g/cm^3.

b. The absorber cross-sectional area is 0.589 cm^2/rod.

c. The absorber material density is 10.17 g/cm^3.

TABLE 12.2.1-43

IRRADIATED TYPE 304 STAINLESS STEEL (0.12 WEIGHT PERCENT COBALT)
CLADDING SOURCE STRENGTHS[a]

Source Strength at Time after Shutdown (Mev/cm -s)

Energy Group (MeV/gamma)	1 Day	1 Week	1 Month	6 Months	1 Year	5 Years
0.20 - 0.40	7.1×10^{9}	6.1×10^{9}	3.4×10^{9}	8.3×10^{7}	9.9×10^{5}	-
0.40 - 0.90	3.1×10^{10}	2.9×10^{10}	2.6×10^{10}	1.2×10^{10}	6.4×10^{9}	2.3×10^{8}
0.90 - 1.35	2.4×10^{11}	2.3×10^{11}	2.3×10^{11}	2.1×10^{11}	2.0×10^{11}	1.2×10^{10}
1.35 - 1.80	1.9×10^{8}	1.8×10^{8}	1.4×10^{8}	3.3×10^{7}	5.4×10^{6}	-

a. The various cladding cross-sectional areas per rod are:

Ag-In-Cd control rod	0.136 cm^2
Cf-252 primary source rod	0.136 cm^2
Sb-Be secondary source rod	0.136 cm^2
Burnable poison rod (including inner sheath)	0.159 cm^2
Hafnium control rod	0.136 cm^2

TABLE 12.2.1-44

IRRADIATED INCORE DETECTOR AND DRIVE CABLE MAXIMUM WITHDRAWAL SOURCE STRENGTHS[a]

Energy Group (MeV/gamma)	Incore Detector (MeV/cm^3-s)	Drive Cable (MeV/cm^3-s)
0.20 - 0.40	3.8×10^{10}	6.0×10^{8}
0.40 - 0.90	1.6×10^{11}	5.1×10^{10}
0.90 - 1.35	1.1×10^{11}	1.6×10^{10}
1.35 - 1.80	1.1×10^{11}	3.1×10^{8}
1.80 - 2.20	2.9×10^{10}	3.8×10^{10}
2.20 - 2.60	3.1×10^{10}	1.3×10^{9}
2.60 - 3.00	1.6×10^{10}	1.3×10^{9}
3.00 - 4.00	2.1×10^{10}	3.1×10^{8}
4.00 - 5.00	1.5×10^{10}	-
5.00 - 6.00	1.4×10^{9}	-

a. The effective diameter and length of the incore detector are 0.48 and 5.33 cm, respectively. The effective cross-sectional area of the drive cable is 0.095 cm^2.

TABLE 12.2.1-45

IRRADIATED INCORE DETECTOR DRIVE CABLE SOURCE STRENGTHS[a]

	Source Strength at Time after Shutdown (MeV/cm -s)						
Energy Group (MeV/gamma)	8 h	1 Day	1 Week	1 Month	6 Months	1 Year	5 Years
0.20 - 0.40	5.8×10^{8}	5.7×10^{8}	4.9×10^{8}	2.8×10^{8}	9.8×10^{6}	2.8×10^{5}	-
0.40 - 0.90	1.7×10^{10}	1.2×10^{10}	1.2×10^{10}	1.1×10^{10}	7.8×10^{9}	5.0×10^{9}	2.0×10^{8}
0.90 - 1.35	1.6×10^{10}	1.6×10^{10}	1.5×10^{10}	1.2×10^{10}	5.6×10^{9}	4.2×10^{9}	2.5×10^{9}
1.35 - 1.80	2.1×10^{7}	1.1×10^{7}	1.1×10^{7}	8.5×10^{6}	2.0×10^{6}	3.3×10^{5}	-
1.80 - 2.20	4.5×10^{9}	6.1×10^{7}	-	-	-	-	-
2.20 - 2.60	1.5×10^{8}	2.0×10^{6}	-	-	-	-	-
2.60 - 3.00	1.6×10^{8}	2.1×10^{6}	-	-	-	-	-
3.00 - 4.00	3.6×10^{7}	5.0×10^{5}	-	-	-	-	-

a. The drive cable effective cross-sectional area is 0.095 cm^2.

TABLE 12.2.1-46

IRRADIATED TYPE 316 STAINLESS STEEL (0.12 WEIGHT PERCENT COBALT)
FLUX THIMBLE SOURCE STRENGTHS[a]

Energy Group (MeV/gamma)	Source Strength at Time after Shutdown (MeV/cm -s)						
	12 h	1 Day	1 Week	1 Month	6 Months	1 Year	5 Years
0.20 - 0.40	6.5×10^{9}	6.4×10^{9}	5.5×10^{9}	3.1×10^{9}	7.5×10^{7}	9.0×10^{5}	-
0.40 - 0.90	5.7×10^{10}	3.7×10^{10}	3.3×10^{10}	2.8×10^{10}	1.2×10^{10}	6.2×10^{9}	2.2×10^{8}
0.90 - 1.35	2.4×10^{11}	2.4×10^{11}	2.4×10^{11}	2.3×10^{11}	2.2×10^{11}	2.0×10^{11}	1.2×10^{11}
1.35 - 1.80	2.9×10^{8}	2.2×10^{8}	2.1×10^{8}	1.7×10^{7}	3.9×10^{7}	6.4×10^{6}	-
1.80 - 2.20	2.1×10^{10}	8.2×10^{8}	-	-	-	-	-
2.20 - 2.60	6.8×10^{8}	2.7×10^{7}	-	-	-	-	-
2.60 - 3.00	7.2×10^{8}	2.9×10^{7}	-	-	-	-	-
3.00 - 4.00	1.7×10^{8}	6.7×10^{2}	-	-	-	-	-

a. The flux thimble cross-sectional area is 0.270 cm^2.

TABLE 12.2.1-47 (SHEET 1 OF 2)

INTEGRATED GAMMA RAY AND BETA SOURCE STRENGTHS AT VARIOUS TIMES FOLLOWING A DESIGN BASIS ACCIDENT (TID-14844 RELEASE FRACTIONS)

	Source Strength at Time after Release (MeV/W)				
Energy Group (MeV/gamma)	0.5 h	1 h	2 h	8 h	1 Day
0.20 - 0.40	1.2×10^{12}	2.2×10^{12}	3.9×10^{12}	1.3×10^{13}	3.2×10^{13}
0.40 - 0.90	8.2×10^{12}	1.5×10^{13}	2.4×10^{13}	5.3×10^{13}	8.6×10^{13}
0.90 - 1.35	3.5×10^{12}	6.2×10^{12}	1.1×10^{13}	2.5×10^{13}	3.8×10^{13}
1.35 - 1.80	3.5×10^{12}	6.2×10^{12}	1.0×10^{13}	2.1×10^{13}	3.0×10^{13}
1.80 - 2.20	1.9×10^{12}	3.3×10^{12}	5.4×10^{12}	1.1×10^{13}	1.3×10^{13}
2.20 - 2.60	2.2×10^{12}	3.9×10^{12}	6.6×10^{12}	1.3×10^{13}	1.5×10^{13}
2.60 - 3.00	3.2×10^{11}	5.4×10^{11}	8.4×10^{11}	1.3×10^{12}	1.5×10^{12}
3.00 - 4.00	2.3×10^{11}	3.4×10^{11}	4.7×10^{11}	6.8×10^{11}	7.4×10^{11}
4.00 - 5.00	9.4×10^{10}	1.0×10^{11}	1.1×10^{11}	1.4×10^{11}	1.5×10^{11}
5.00 - 6.00	1.0×10^{9}	1.0×10^{9}	1.0×10^{9}	1.0×10^{9}	1.0×10^{9}
Beta	1.2×10^{13}	2.2×10^{13}	3.6×10^{13}	8.6×10^{11}	1.5×10^{14}

TABLE 12.2.1-47 (SHEET 2 OF 2)

Energy Group (MeV/gamma)	Source Strength at Time after Release (MeV/W) 1 Week	1 Month	6 Months	1 Year
0.20 - 0.40	1.2×10^{14}	2.3×10^{14}	2.5×10^{14}	2.6×10^{14}
0.40 - 0.90	1.7×10^{14}	2.7×10^{14}	5.1×10^{14}	5.9×10^{14}
0.90 - 1.35	4.9×10^{13}	5.3×10^{13}	5.2×10^{13}	6.0×10^{13}
1.35 - 1.80	4.8×10^{13}	7.3×10^{13}	8.4×10^{13}	8.6×10^{13}
1.80 - 2.20	1.4×10^{13}	1.5×10^{13}	1.8×10^{13}	1.9×10^{13}
2.20 - 2.60	1.6×10^{13}	1.8×10^{13}	1.9×10^{13}	1.9×10^{13}
2.60 - 3.00	1.7×10^{12}	1.7×10^{12}	1.7×10^{12}	1.7×10^{12}
3.00 - 4.00	7.4×10^{11}	7.5×10^{11}	7.6×10^{11}	7.6×10^{11}
4.00 - 5.00	1.5×10^{11}	1.5×10^{11}	1.5×10^{11}	1.5×10^{11}
5.00 - 6.00	1.0×10^{9}	1.0×10^{9}	1.0×10^{9}	1.0×10^{9}
Beta	3.8×10^{14}	6.5×10^{14}	1.0×10^{15}	1.3×10^{15}

TABLE 12.2.1-48

SOURCE STRENGTH IN THE RESIDUAL
HEAT REMOVAL LOOP AT VARIOUS TIMES FOLLOWING AN EQUIVALENT
FULL-CORE MELTDOWN ACCIDENT

Source Strength at Time after Release (MeV/W)

Energy Group (MeV/gamma)	0 h	0.5 h	1 h	2 h	8 hr
0.20 - 0.40	2.8×10^{-1}	1.9×10^{-1}	1.7×10^{-1}	1.5×10^{-1}	1.3×10^{-1}
0.40 - 0.90	3.2	2.5	2.0	1.4	5.6×10^{-1}
0.90 - 1.35	1.6	9.9×10^{-1}	8.6×10^{-1}	6.8×10^{-1}	3.1×10^{-1}
1.35 - 1.80	1.2	6.7×10^{-1}	5.7×10^{-1}	4.5×10^{-1}	2.0×10^{-1}
1.80 - 2.20	1.1	6.9×10^{-2}	5.4×10^{-2}	4.0×10^{-2}	1.4×10^{-2}
2.20 - 2.60	2.4×10^{-1}	4.7×10^{-2}	3.8×10^{-2}	2.9×10^{-2}	1.3×10^{-2}
2.60 - 3.00	3.0×10^{-1}	5.0×10^{-3}	2.4×10^{-3}	6.3×10^{-4}	-
3.00 - 4.00	1.6×10^{-1}	3.3×10^{-2}	1.6×10^{-2}	4.5×10^{-3}	-
4.00 - 5.00	2.8×10^{-1}	4.5×10^{-4}	2.3×10^{-4}	-	-
5.00 - 6.00	1.4×10^{-3}	-	-	-	-
	1 Day	1 Week	1 Month	6 Months	1 Year
0.20 - 0.40	1.2×10^{-1}	6.7×10^{-2}	9.7×10^{-3}	1.9×10^{-4}	1.1×10^{-4}
0.40 - 0.90	2.9×10^{-1}	5.5×10^{-2}	2.6×10^{-2}	6.4×10^{-3}	2.3×10^{-3}
0.90 - 1.35	7.3×10^{-2}	3.2×10^{-3}	8.3×10^{-4}	1.3×10^{-4}	9.6×10^{-5}
1.35 - 1.80	5.4×10^{-2}	1.7×10^{-2}	4.7×10^{-3}	8.2×10^{-5}	5.8×10^{-5}
1.80 - 2.20	3.2×10^{-3}	6.6×10^{-4}	2.6×10^{-4}	8.5×10^{-5}	5.4×10^{-5}
2.20 - 2.60	3.4×10^{-3}	9.8×10^{-4}	2.8×10^{-4}	-	-

TABLE 12.2.1-49 (SHEET 1 OF 3)

POST-ACCIDENT SOURCE TERMS
SPECIFIC ACTIVITIES FOR TID-14844 RELEASE FRACTIONS

Nuclide	Containment Atmosphere (Ci/cm^3)	Reactor Coolant (Ci/cm^3)	Containment Sump Ci/cm^3)
Sr-89 N/A	2.49×10^{-3}	5.16×10^{-4}	
Sr-90 N/A	2.24×10^{-4}	4.65×10^{-5}	
Y-90	N/A	2.36×10^{-4}	4.91×10^{-5}
Y-91	N/A	3.38×10^{-3}	7.01×10^{-4}
Zr-95 N/A	4.91×10^{-3}	1.02×10^{-3}	
Zr-97 N/A	4.91×10^{-3}	1.02×10^{-3}	
Nb-95 N/A	4.91×10^{-3}	1.02×10^{-3}	
Nb-95m	N/A	6.14×10^{-5}	1.27×10^{-5}
Nb-97 N/A	5.22×10^{-3}	1.08×10^{-3}	
Mo-99 N/A	5.53×10^{-3}	1.15×10^{-3}	
Tc-99m	N/A	4.91×10^{-3}	1.02×10^{-3}
Ru-103	N/A	5.22×10^{-3}	1.08×10^{-3}
Ru-106	N/A	1.81×10^{-3}	3.76×10^{-4}
Rh-103m	N/A	5.22×10^{-3}	1.08×10^{-3}
Rh-105	N/A	3.38×10^{-3}	7.01×10^{-4}
Rh-106	N/A	2.12×10^{-3}	4.40×10^{-4}
Ag-110m	N/A	1.97×10^{-5}	4.08×10^{-6}
Ag-111	N/A	1.75×10^{-4}	3.63×10^{-5}
Sb-125	N/A	3.01×10^{-5}	6.25×10^{-6}
Sb-127	N/A	3.38×10^{-4}	7.01×10^{-5}
Te-127	N/A	3.38×10^{-4}	7.01×10^{-5}

TABLE 12.2.1-49 (SHEET 2 OF 3)

Nuclide	Containment Atmosphere (Ci/cm^3)	Reactor Coolant (Ci/cm^3)	Containment Sump Ci/cm^3)
Te-129	N/A	1.01×10^{-3}	2.10×10^{-4}
Te-129m	N/A	2.73×10^{-4}	5.67×10^{-5}
Te-132	N/A	4.30×10^{-3}	8.92×10^{-4}
Cs-134	N/A	7.06×10^{-4}	1.47×10^{-4}
Cs-136	N/A	1.97×10^{-4}	4.08×10^{-5}
Cs-137	N/A	3.38×10^{-4}	7.01×10^{-5}
Ba-137m	N/A	3.01×10^{-4}	6.25×10^{-5}
Ba-140	N/A	5.22×10^{-3}	1.08×10^{-3}
La-140 N/A	5.53×10^{-3}	1.15×10^{-3}	
Ce-141	N/A	4.91×10^{-3}	1.02×10^{-3}
Ce-143	N/A	4.30×10^{-3}	8.92×10^{-4}
Ce-144	N/A	3.69×10^{-3}	7.65×10^{-4}
Pr-143 N/A	4.30×10^{-3}	8.92×10^{-4}	
Pr-144 N/A	3.69×10^{-3}	7.65×10^{-4}	
Nd-147	N/A	1.93×10^{-3}	4.02×10^{-4}
Nd-154	N/A	1.29×10^{-4}	2.68×10^{-5}
Pm-147	N/A	3.38×10^{-4}	7.01×10^{-5}
Pm-148	N/A	6.14×10^{-4}	1.27×10^{-4}
Pm-148m	N/A	2.73×10^{-4}	5.67×10^{-5}
Pm-149	N/A	1.93×10^{-3}	4.02×10^{-4}
Pm-154	N/A	1.63×10^{-4}	3.38×10^{-5}
Sm-153	N/A	1.93×10^{-3}	4.02×10^{-4}

TABLE 12.2.1-49 (SHEET 3 OF 3)

Nuclide	Containment Atmosphere (Ci/cm^3)	Reactor Coolant (Ci/cm^3)	Containment Sump Ci/cm^3)
Eu-156	N/A	1.01×10^{-3}	2.10×10^{-4}
Kr-83m	1.54×10^{-4}	3.69×10^{-2}	N/A
Kr-85m	3.47×10^{-4}	8.29×10^{-2}	N/A
Kr-85 8.48×10^{-6}	2.03×10^{-3}		N/A
Kr-87 6.29×10^{-4}	1.50×10^{-1}		N/A
Kr-88 8.99×10^{-4}	2.15×10^{-1}		N/A
Kr-89 1.12×10^{-3}	2.67×10^{-1}		N/A
Xe-131m	8.99×10^{-6}	2.15×10^{-3}	N/A
Xe-133m	3.72×10^{-4}	8.91×10^{-2}	N/A
Xe-133	2.44×10^{-3}	5.83×10^{-1}	N/A
Xe-135m	5.14×10^{-4}	1.23×10^{-1}	N/A
Xe-135	5.39×10^{-4}	1.29×10^{-1}	N/A
Xe-138	2.05×10^{-3}	4.91×10^{-1}	N/A
Br-82 N/A	4.61×10^{-4}	9.56×10^{-5}	
Br-83 N/A	1.84×10^{-2}	3.82×10^{-3}	
Br-84 N/A	3.07×10^{-2}	6.37×10^{-3}	
I-130 5.65×10^{-6}	2.70×10^{-3}	5.61×10^{-4}	
I-131 3.15×10^{-4}	1.50×10^{-1}	3.12×10^{-2}	
I-132 4.62×10^{-4}	2.21×10^{-1}	4.59×10^{-2}	
I-133 6.42×10^{-4}	3.07×10^{-1}	6.37×10^{-2}	
I-134 7.06×10^{-4}	3.38×10^{-1}	7.01×10^{-2}	
I-135 6.04×10^{-4}	2.89×10^{-1}	5.99×10^{-2}	

TABLE 12.2.1-50

REFUELING WATER STORAGE TANK
SPECIFIC ACTIVITIES

Nuclide	Specific Activity (μCi/g)	Nuclide	Specific Activity (μCi/g)	Nuclide	Specific Activity (μCi/g)
Br-83	-	Te-129m	1.1×10^{-3}	Pr-143	3×10^{-5}
Kr-83m	-	Te-129	7.5×10^{-4}	Pr-144	7.7×10^{-8}
Kr-85m	2.9×10^{-9}	Te-132	3×10^{-3}	Cr-51	1.3×10^{-5}
Kr-85	4×10^{-2}	I-129	4.1×10^{-8}	Mn-54	5.4×10^{-6}
Kr-88	-	I-130	8.4×10^{-9}	Fe-59	1.4×10^{-6}
Rb-88	-	I-131	9×10^{-2}	Co-58	3×10^{-5}
Sr-89	2.7×10^{-4}	I-132	3.1×10^{-3}	Co-60	2.6×10^{-5}
Sr-90	8.6×10^{-5}	I-133	2.3×10^{-4}	H-3	1.8
Sr-91	-	Xe-131m	9.6×10^{-2}	Ag-110m	1.5×10^{-4}
Sr-92	-	Xe-133m	8×10^{-2}	Ag-110	1.5×10^{-4}
Y-90	8.5×10^{-5}	Xe-133	6.0	Rh-106	1.8×10^{-5}
Y-91m	2.4×10^{-8}	Xe-135m	1.9×10^{-3}	Te-127m	2.1×10^{-4}
Y-91	4.2×10^{-5}	Xe-135	2.2×10^{-3}	Te-127	2.1×10^{-4}
Y-92	-	Cs-134	4.8×10^{-1}	Rh-103m	1.4×10^{-6}
Y-93	-	Cs-135	3.4×10^{-9}	Te-125m	1.8×10^{-5}
Zr-95	1.5×10^{-6}	Cs-136	1.3×10^{-1}	Fe-55	4.8×10^{-4}
Nb-95m	2.2×10^{-7}	Cs-137	1.1		
Nb-95	3.8×10^{-5}	Ba-137m	1.1		
Mo-99	5.5×10^{-3}	Ba-140	1.8×10^{-4}		
Tc-99m	7×10^{-3}	La-140	2×10^{-4}		
Tc-99	5×10^{-8}	Ce-141	8.1×10^{-8}		
Ru-103	1.4×10^{-6}	Ce-143	3.7×10^{-7}		
Ru-106	1.8×10^{-5}	Ce-144	7.7×10^{-8}		

TABLE 12.2.1-51

REACTOR MAKEUP WATER STORAGE TANK
SPECIFIC ACTIVITIES

Nuclide	Specific Activity (μCi/g)	Nuclide	Specific Activity (μCi/g)	Nuclide	Specific Activity (μCi/g)
Kr-83m	4.6×10^{-3}	Rb-86	1.1×10^{-5}	Zr-95	6.5×10^{-8}
Kr-85m	2×10^{-3}	Rb-88	2.4×10^{-3}	Nb-95	6.5×10^{-8}
Kr-95	7.3×10^{-3}	Rb-89	1.1×10^{-4}	Mo-99	7.5×10^{-5}
Kr-87	1.3×10^{-3}	Cs-134	1.2×10^{-3}	Tc-99m	6.9×10^{-5}
Kr-88	3.6×10^{-3}	Cs-136	1.5×10^{-3}	Ru-103	5.7×10^{-8}
Kr-89	1.1×10^{-4}	Cs-137	7.5×10^{-4}	Ru-106	1.4×10^{-8}
Xe-131m	2.2×10^{-3}	Ba-137m	1.4×10^{-4}	Rh-103m	5.7×10^{-8}
Xe-133m	1.7×10^{-2}	Cs-138	9.6×10^{-5}	Rh-106	1.4×10^{-8}
Xe-133	2.7×10^{-1}	H-3	3.5	Ag-110m	1.4×10^{-7}
Xe-135m	4.8×10^{-4}	Cr-51	5.5×10^{-7}	Te-125m	2.8×10^{-8}
Xe-135	7.2×10^{-3}	Mn-54	4.4×10^{-8}	Te-127m	2.9×10^{-7}
Xe-137	1.7×10^{-4}	Mn-56	2×10^{-6}	Te-127	1.2×10^{-6}
Xe-138	6.4×10^{-4}	Fe-55	2×10^{-7}	Te-129m	1.9×10^{-6}
Br-83	9.5×10^{-6}	Fe-59	5.2×10^{-8}	Te-129	1.8×10^{-6}
Br-84	4.7×10^{-7}	Co-58	1.5×10^{-6}	Te-131m	2.6×10^{-6}
Br-85	6×10^{-7}	Co-60	1.9×10^{-7}	Te-131	1.2×10^{-6}
I-127	-	Sr-89	4.3×10^{-7}	Te-132	2.9×10^{-5}
I-129	-	Sr-90	1.2×10^{-8}	Te-134	3×10^{-6}
I-130	2.1×10^{-5}	Sr-91	6.2×10^{-7}	Ba-140	4.2×10^{-7}
I-131	2.8×10^{-3}	Sr-92	1.3×10^{-7}	La-140	1.4×10^{-7}
I-132	2.8×10^{-3}	Y-90	3.4×10^{-9}	Ce-141	6.3×10^{-8}
I-133	4.2×10^{-3}	Y-91m	3.3×10^{-7}	Ce-143	5×10^{-8}
I-134	5.7×10^{-4}	Y-91	5.7×10^{-8}	Ce-144	3.9×10^{-8}
I-135	2.3×10^{-3}	Y-92	1.2×10^{-7}	Pr-143	6.3×10^{-8}
		Y-93	3.8×10^{-8}	Pr-144	3.9×10^{-8}

TABLE 12.2.1-52

CONDENSATE STORAGE TANK
SPECIFIC ACTIVITIES

Nuclide	Specific Activity (µCi/g)	Nuclide	Specific Activity (µCi/g)
Br-83	2.5×10^{-7}	Y-90	5.3×10^{-11}
Br-84	3.5×10^{-8}	Y-91m	2.9×10^{-9}
Br-85	4.3×10^{-10}	Y-91	6.1×10^{-10}
I-129	4.2×10^{-13}	Y-92	4×10^{-10}
I-130	1.3×10^{-7}	Y-93	2.3×10^{-10}
I-131	2.7×10^{-5}	Zr-95	6.6×10^{-10}
I-132	9.3×10^{-6}	Nb-95	6.6×10^{-10}
I-133	3.1×10^{-5}	Mo-99	6.9×10^{-7}
I-134	7×10^{-7}	Tc-99m	6.5×10^{-7}
I-135	1.1×10^{-5}	Ru-103	5.8×10^{-10}
Rb-86	1.95×10^{-7}	Ru-106	1.4×10^{-10}
Rb-88	1.05×10^{-6}	Rh-103m	5.8×10^{-10}
Rb-89	4×10^{-8}	Rh-106	1.4×10^{-10}
Cs-134	2.05×10^{-5}	Ag-110m	1.4×10^{-9}
Cs-136	2.6×10^{-5}	Te-125m	2.8×10^{-10}
Cs-137	1.35×10^{-5}	Te-127m	3.0×10^{-9}
Ba-137m	1.25×10^{-5}	Te-127	8.1×10^{-9}
Cs-138	2.75×10^{-7}	Te-129m	1.9×10^{-8}
H-3	1.8	Te-129	1.3×10^{-8}
Cr-51	5.6×10^{-9}	Te-131m	2.2×10^{-8}
Mn-54	4.5×10^{-10}	Te-131	4.4×10^{-9}
Mn-56	5.7×10^{-9}	Te-132	2.8×10^{-7}
Fe-55	2×10^{-9}	Te-134	2.9×10^{-9}
Fe-59	5.3×10^{-10}	Ba-140	4.2×10^{-9}
Co-58	1.5×10^{-8}	La-140	1.8×10^{-9}
Co-60	1.9×10^{-9}	Ce-141	6.4×10^{-10}
Sr-89	7.8×10^{-9}	Ce-143	4.3×10^{-10}
Sr-90	2.2×10^{-10}	Ce-144	3.9×10^{-10}
Sr-91	5×10^{-9}	Pr-143	6.4×10^{-10}
Sr-92	4.3×10^{-10}	Pr-144	3.9×10^{-10}

TABLE 12.2.2-1 (SHEET 1 OF 2)

PARAMETERS AND ASSUMPTIONS FOR CALCULATING AIRBORNE RADIOACTIVE CONCENTRATIONS

Leak Rates (lb/day)	
Equivalent reactor coolant leak into containment during power for noble gases	5100
Equivalent reactor coolant leak into containment for halogens	5.1
Equivalent reactor coolant leak into auxiliary building	160
Equivalent reactor coolant leak into letdown heat exchanger valve gallery	7.4
Equivalent steam generator steam leak into turbine building	40,800
Evaporation Rates (g/min)	
From refueling pool into containment atmosphere	2960
From spent fuel pool into fuel building atmosphere	2670

Partition Factors	Noble Gases	Halogens	Particulates	Tritium
Auxiliary building	1	0.0075	0.0001	0.1
Fuel handling building	Negligible	Negligible	Negligible	1

TABLE 12.2.2-1 (SHEET 2 OF 2)

Ventilation Rates (ft^3/min)	
Containment during power	5000
Containment during refueling	15,000
Fuel handling building, in the vicinity of each spent fuel pool, during refueling	10,890
Auxiliary building	72,000
Auxiliary building letdown heat exchanger valve gallery	90
Turbine building	1.1×10^6
Volumes of the Regions (ft^3)	
Containment	2.75×10^6
Fuel handling building	5.0×10^5
Auxiliary building	1.9×10^6
Auxiliary building letdown heat exchanger valve gallery	2730
Turbine building	5.3×10^6
Miscellaneous Information	
Failed fuel percentage for fission products	0.12
Reactor coolant specific activities	Table 11.1-7
Steam generator steam activities	Table 11.1-7
Plant capacity factor (percent)	80

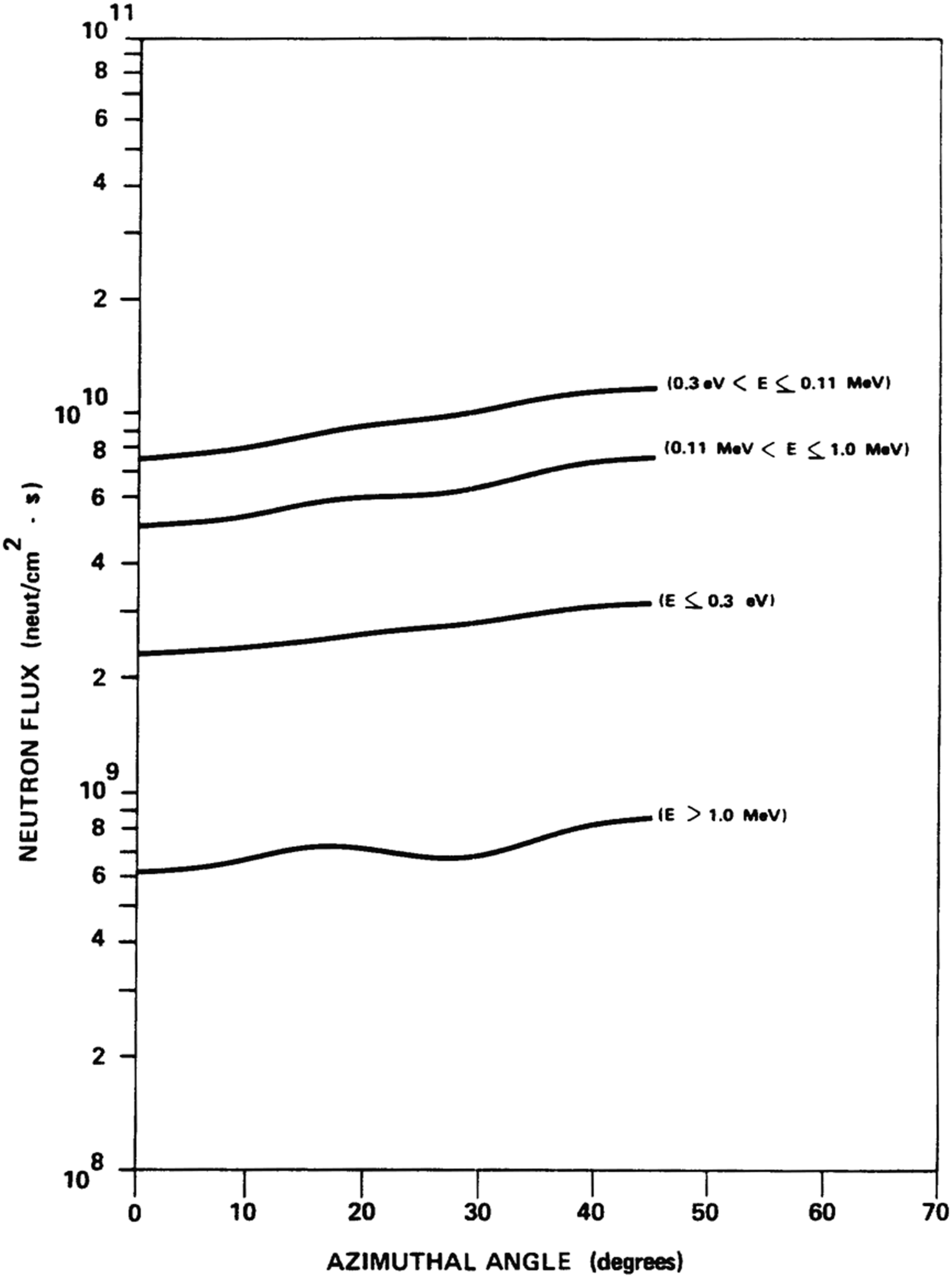
TABLE 12.2.2-2 (SHEET 1 OF 2)

AIRBORNE RADIOACTIVITY CONCENTRATIONS
(μCi/cm^3)

Nuclide	Containment (100%Power)	Containment (Refueling)	Fuel Handling Building (Refueling)	Turbine Building
H-3	1.50E-6	2.50E-6	2.50E-6	4.54E-10
N-16	-			-
Ar-41	9.97E-7			-
Mn-54	2.95E-10			-
Fe-59	1.01E-10			-
Co-58	1.01E-9			-
Co-60	4.56E-10			-
Br-83	1.67E-11			5.24E-16
Br-84	2.62E-12			7.74E-17
Br-85	2.98E-14			3.50E-19
Kr-83m	5.33E-8			2.46E-15
Kr-85m	4.46E-7			1.20E-14
Kr-85	5.40E-8			5.90E-16
Kr-87	1.16E-7			6.76E-15
Kr-88	6.60E-7			2.24E-14
Kr-89	5.25E-10			2.64E-16
Rb-86	-			3.36E-18
Rb-88	-			2.88E-16
Sr-89	2.28E-11			-
Sr-90	3.99E-12			-
I-130	1.70E-11			5.96E-16
I-131	3.05E-9			9.98E-14
I-132	3.26E-10			3.96E-14
I-133	3.45E-9			1.09E-13
I-134	7.22E-11			2.26E-15
I-135	1.20E-9			3.72E-14
Xe-131m	1.65E-7			1.86E-15
Xe-133m	8.63E-7			1.09E-14
Xe-133	4.39E-5			4.98E-14
Xe-135m	6.06E-9			1.33E-15
Xe-135	1.85E-6			3.42E-14
Xe-137	1.12E-9			5.22E-16
Xe-138	1.98E-8			4.24E-15
Cs-134	2.95E-10			7.72E-16
Cs-136	-			4.32E-16
Cs-137	5.10E-10			6.36E-16
Ba-137m	-			1.89E-16

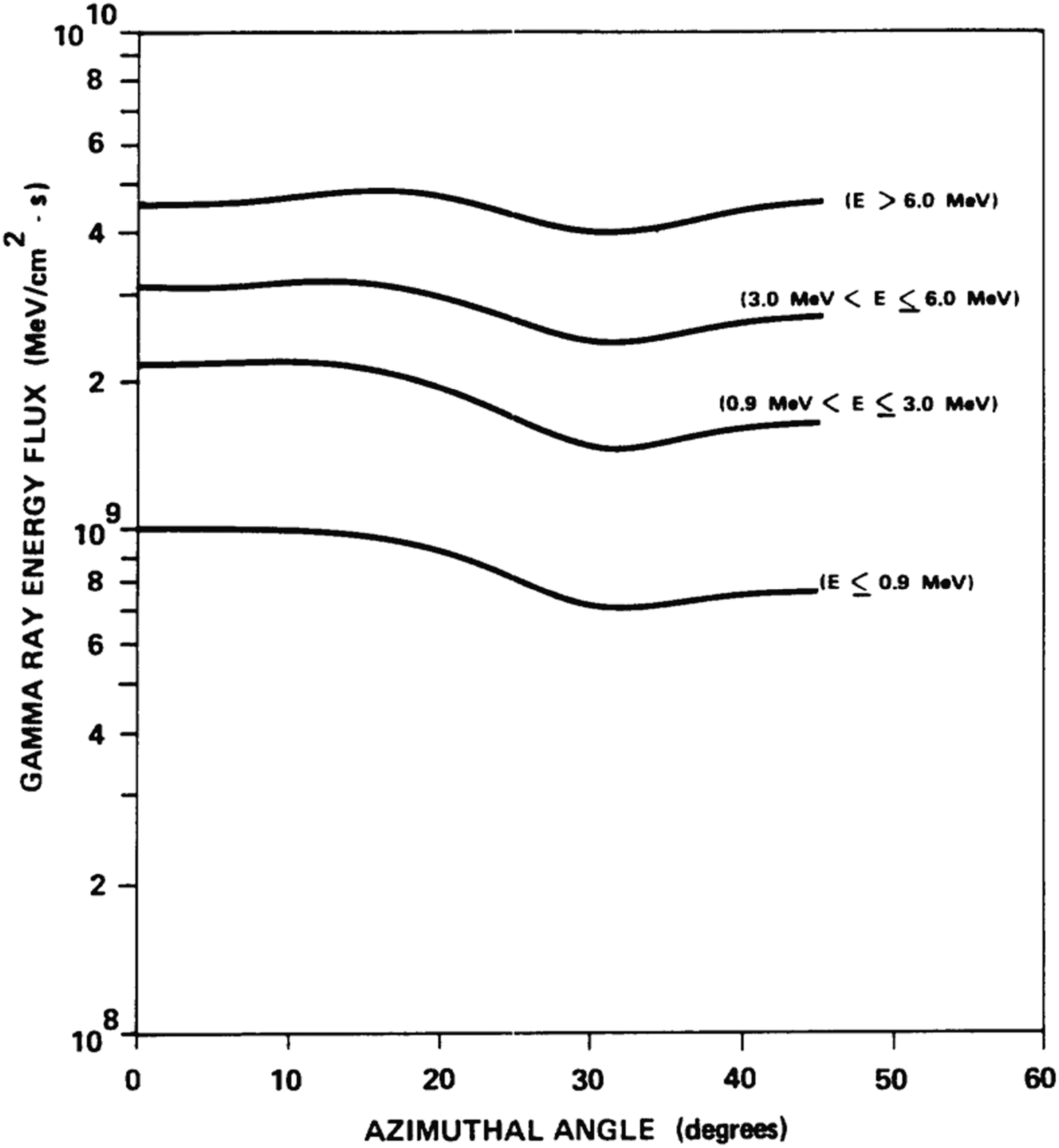
TABLE 12.2.2-2 (SHEET 2 OF 2)

	Auxiliary Building	
Nuclide	Corridor	Letdown Heat Exchanger Valve Gallery
H-3	2.48E-9	9.19E-8
N-16	-	-
Mn-54	-	-
Fe-59	-	-
Co-58	-	-
Co-60	-	-
Br-83	8.93E-13	3.25E-11
Br-84	3.59E-13	1.25E-11
Br-85	9.13E-15	2.92E-13
Kr-83m	4.49E-10	1.62E-8
Kr-85m	2.23E-9	8.18E-8
Kr-85	1.19E-10	4.41E-9
Kr-87	1.24E-9	4.47E-8
Kr-88	4.26E-9	1.55E-7
Kr-89	2.13E-11	6.83E-10
Rb-86	2.20E-16	8.17E-15
Rb-88	2.87E-13	9.76E-12
Sr-89	-	-
Sr-90	-	-
I-130	4.17E-13	1.54E-11
I-131	5.19E-11	1.93E-9
I-132	1.81E-11	6.58E-10
I-133	7.32E-11	2.71E-9
I-134	7.36E-12	2.61E-10
I-135	3.73E-11	1.37E-9
Xe-131m	3.71E-10	1.38E-8
Xe-133m	2.12E-9	7.85E-8
Xe-133	1.01E-7	3.76E-6
Xe-135m	1.61E-10	5.42E-9
Xe-135	6.73E-9	2.48E-7
Xe-137	4.42E-11	1.42E-9
Xe-138	5.40E-10	1.82E-8
Cs-134	6.44E-14	2.39E-12
Cs-136	3.47E-14	1.29E-12
Cs-137	4.71E-14	1.75E-12
Ba-137m	1.09E-16	1.82E-13



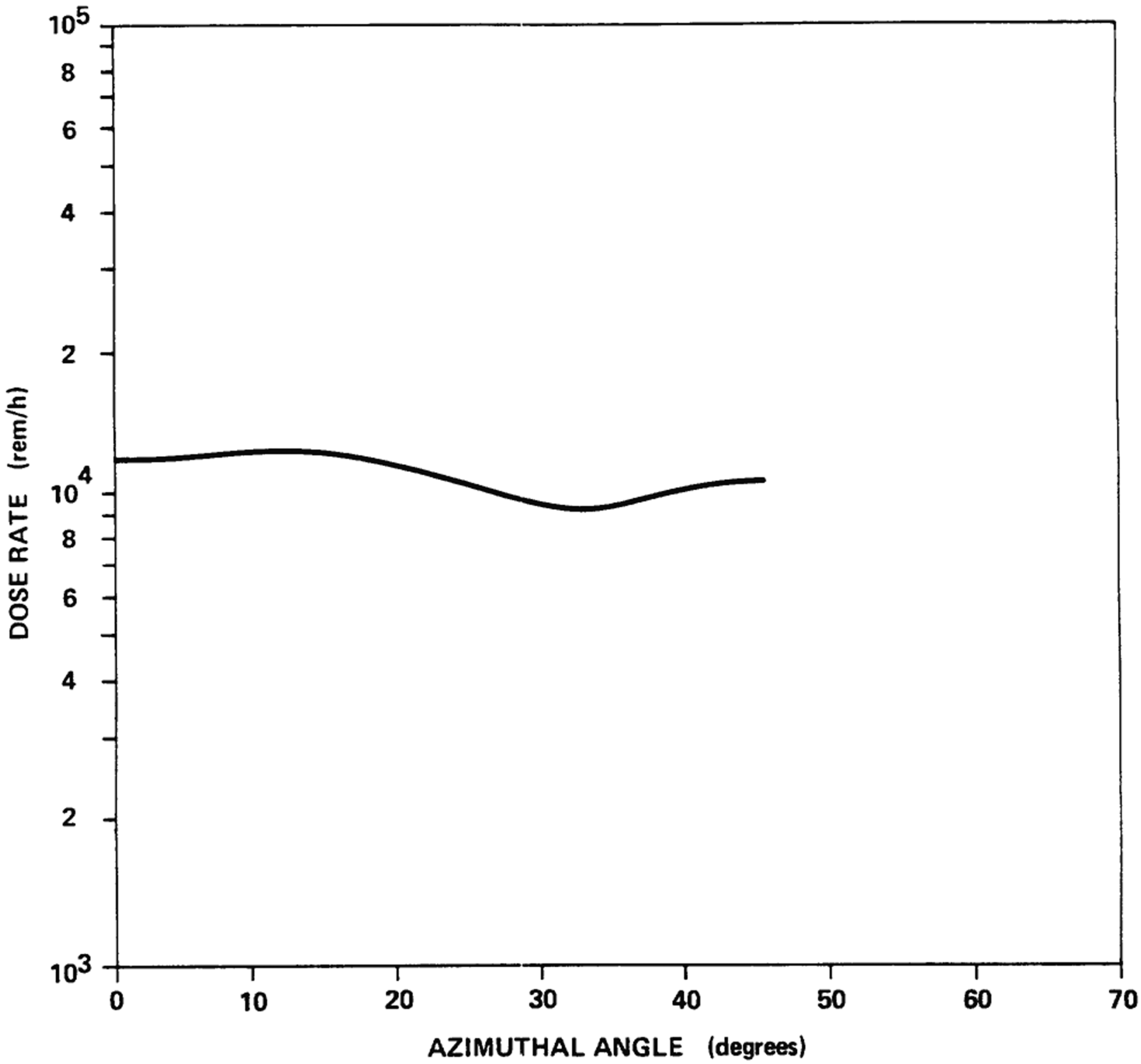
REV 14 10/07

SOUTHERN COMPANY *Energy to Serve Your World®*	VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2	AZIMUTHAL DISTRIBUTION OF NEUTRON FLUX INCIDENT ON THE PRIMARY SHIELD AT THE REACTOR CORE MIDPLANE
		FIGURE 12.2.1–1



REV 14 10/07

SOUTHERN COMPANY Energy to Serve Your World®	VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2	AZIMUTHAL DISTRIBUTION OF GAMMA RAY ENERGY FLUX INCIDENT ON THE PRIMARY SHIELD AT THE REACTOR CORE MIDPLANE FIGURE 12.2.1–2



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SOUTHERN COMPANY
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VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

AZIMUTHAL DISTRIBUTION OF GAMMA RAY DOSE RATE INCIDENT ON THE PRIMARY SHIELD AT THE REACTOR CORE MIDPLANE

FIGURE 12.2.1–3

12.3 RADIATION PROTECTION DESIGN FEATURES

12.3.1 FACILITY DESIGN FEATURES

Specific design features for maintaining personnel exposure as low as reasonably achievable (ALARA) are discussed in this subsection. The design feature recommendations given in Regulatory Guide 8.8, Paragraph C.3, are utilized to minimize exposures to personnel.

12.3.1.1 Plant Design Description for ALARA

The equipment and plant design features employed to maintain radiation exposures ALARA are based upon the design considerations of subsection 12.1.2 and are outlined in this subsection for several general classes of equipment (paragraph 12.3.1.1.1) and several typical plant layout situations (paragraph 12.3.1.1.2).

12.3.1.1.1 Common Equipment and Component Designs for ALARA

This paragraph describes the design features utilized for several general classes of equipment or components. These classes of equipment are common to many of the plant systems; thus, the features employed for each system to maintain minimum exposures are similar and are discussed by equipment class in the following paragraphs. The locations of equipment and components are shown on drawings 1X4DE312, 1X4DE313, 1X4DE314, 1X4DE315, 1X4DE317, 1X4DE318, 1X4DE320, 1X4DE321, 1X4DE322, 1X4DE323, 1X4DE324, and 1X4DE325.

12.3.1.1.1.1 Balance of Plant Equipment.

A. Filters

Filters that accumulate radioactivity are supplied with the means either to backflush the filter remotely or to perform cartridge replacement with semiremote tools. For cartridge filters, adequate space is provided to allow removing, cask loading, and transporting the cartridge to the solid radwaste area.

Backflushable filters are designed so that filter internals may be remotely removed and placed in a shielded cask for offsite shipping and disposal in the unlikely event that a filter loses its backflush capability.

Liquid systems containing radioactive cartridge filters are provided with a remote filter handling system for the removal of spent radioactive filter cartridges from their housings and for their transfer to the drumming station for packaging and shipment from the site for burial. The process is accomplished in such a manner that exposure to personnel and the possibility of inadvertent radioactive release to the environment is minimized. Each filter is contained in a shielded compartment and provided with vent and drain valving, and individual compartments are provided with compartment drainage capabilities. The filter handling system has also been designed to be simple with a minimum of components susceptible to malfunction.

B. Demineralizers

Demineralizers for highly radioactive systems are designed so that spent resins can be remotely and hydraulically transferred to spent resin tanks so that fresh resin can be loaded into the demineralizer remotely. Underdrains are designed for full system pressure drop. The demineralizers and piping are designed with provisions for being flushed with demineralized water. Strainers are installed in the vent lines to prevent entry of spent resin into the exhaust duct.

C. Evaporators (Abandoned in place)

D. Pumps

Wherever practicable, pumps are purchased with mechanical seals to reduce seal servicing time. Pumps and associated piping are arranged to provide adequate space for access to the pumps for servicing. Small pumps are installed in a manner which allows easy removal if necessary. All pumps-in radioactive waste systems are provided with flanged connections for ease of removal. Pump casings are provided with drain connections for draining the pump for maintenance.

E. Tanks

Whenever practicable, tanks are provided with sloped bottoms and bottom outlet connections. Overflow lines are directed to the waste collection system to control any contamination within plant structures. For tanks outside structures, dikes are used to contain overflows.

F. Heat Exchangers

Heat exchangers are provided with corrosion-resistant tubes of stainless steel or other suitable materials to minimize leakage. Impact baffles are provided and tube side and shell side velocities are limited to minimize erosive effects. Wherever possible, the radioactive fluid passes through the tube side of the heat exchanger.

G. Instruments

Instrument devices are located in low radiation zones and away from radiation sources whenever practical. Primary instrument devices, which for functional reasons are located in high radiation zones, are designed for easy removal to a lower radiation zone for calibration. Transmitters and readout devices are located in low radiation zones, such as corridors and the control room, for servicing.

Some instruments in high radiation zones, such as thermocouples, are provided in duplicate to reduce access and service time required. In containment, instruments are located outside the secondary shield (area of lowest radiation at power and shutdown) whenever practical.

Integral radiation check sources for response verification for airborne radiation monitors and safety-related area radiation monitors are provided.

Chemical seals are provided on the instrument sensing lines on process piping, which may contain highly radioactive solids, to reduce the servicing time required to keep the lines free of solids. Instrument and sensing line connections are located slightly above the pipe midplane wherever practical to minimize radioactive crud or gas buildup.

Process control panels are shown in the equipment location drawings.

H. Valves

To minimize personnel exposures from valve operations, motor-operated, air-operated, or other remotely actuated valves are used where justified by the activity levels and frequency of use. Valves are located in valve galleries so that they are shielded separately from the major components. Long runs of exposed piping are minimized in valve galleries. In areas where manual valves are used on frequently operated process lines, either valve stem extenders or shielding is provided such that personnel need not enter a high radiation area for valve operation.

When equipment in high radiation areas is operated infrequently, only those manual valves associated with safe operation of the equipment are provided with remote-manual operators or reach rods. All other valve operations are performed with equipment in the shutdown mode.

For valves located in radiation areas, provisions are made to drain adjacent radioactive components when maintenance is required. To the extent practicable, valves are not located at piping low points. Valves in containment expected to exhibit stem leakage are provided with leakoff connections piped to the reactor coolant drain tank (RCDT or a reduced packing configuration with the valve stem leakoff line capped). Valves for clean, nonradioactive systems are separated from radioactive sources and are located in readily accessible areas. For most larger valves (2 1/2 in. and larger) in lines carrying radioactive fluids, a double set of packing with a lantern ring is provided. A stuffing box with a leakoff connection that may be piped to a drain header is also provided. Metal diaphragm or bellows seat valves are used on those systems where essentially no leakage can be tolerated.

Manually operated valves in the filter and demineralizer valve compartments required for normal operation and shutdown are equipped with reach rods extending through or over the valve gallery wall. Personnel do not enter the valve gallery during spent resin or cartridge transfer operations. The valve gallery shield walls are designed to minimize personnel exposure during maintenance of components within or adjacent to the gallery and to protect personnel who remotely operate the valves.

Relief valves are located in an associated equipment compartment or valve gallery. Check valves are located in the equipment compartment or associated valve gallery unless they are the locking type requiring manipulation during normal operation. In this case, check valves are treated as normal manual valves.

I. Piping

The piping in pipe chases is designed for the lifetime of the unit[a]. Wherever radioactive piping is routed through areas where routine maintenance is required, pipe chases are provided to reduce the radiation contribution from these pipes to levels appropriate for the inspection or maintenance requirements. Butt welds

[a] The operating licenses for both VEGP units have been renewed, and the original licensed operating terms have been extended by 20 years. In accordance with 10 CFR Part 54, appropriate aging management programs and activities have been initiated to manage the detrimental effects of aging to maintain functionality during the period of extended operation (see chapter 19).

are used to the fullest extent possible in radwaste piping utilized for transport of spent resins or slurries. Piping containing radioactive material is routed to minimize radiation exposure to the unit personnel.

J. Floor Drains

Floor drains and properly sloped floors are provided for each room or cubicle containing serviceable components containing radioactive liquids. When practicable, shielded pipe chases are used for radioactive pipes. If a radioactive drain line must pass through a plant area requiring personnel access, shielding is provided as necessary-to ensure radiation levels consistent with the required access.

K. Lighting

Wherever practicable, multiple electric lights are provided for each cell or room containing highly radioactive components so that the burnout of a single lamp will not require entry and immediate replacement of the defective lamp since sufficient illumination will still be available. Normally, incandescent lights are provided which require less time for servicing and, hence, the personnel exposure is reduced. The fluorescent lights which are used in some areas do not require frequent service due to the increased life of the tubes. Burned out lamps can be replaced when the system is secured and flushed out.

L. Heating, Ventilation and Air-Conditioning

The heating, ventilation, and air-conditioning (HVAC) system design facilitates replacement of the filter elements.

M. Hydrogen Recombiners

Provision is made for permanent shielding of the hydrogen recombiner control console from components that are highly radioactive.

N. Sample Stations

Sample stations for routine sampling of process fluids are located as shown in the equipment location and general arrangement drawings presented in section 1.2. Proper shielding and ventilation are provided at the local sample stations to maintain radiation zoning in proximate areas and minimize personnel exposure during sampling. The counting room and laboratory facilities are described in section 12.5. The use of concrete containing fly ash is minimized in the counting room and laboratory areas.

O. Clean Services

Whenever possible, clean services and equipment such as compressed air piping, clean water piping, ventilation ducts, and cable trays are not routed through radioactive pipeways.

12.3.1.1.1.2 Nuclear Steam Supply System (NSSS) Equipment.

A. Reactor Vessel

The reactor vessel nozzle welds are designed to accommodate remote inspection with ultrasonic sensors. The nozzle area is tapered along the reinforced areas to ensure a smooth transition, and pipe branch locations are selected to ensure no interference from one branch to the next. All weld-to-pipe interfaces require a smooth, high quality finish.

B. Reactor Coolant Pumps

The reactor coolant pump design includes the use of an assembled cartridge seal for the No. 3 pump seal which reduces the time required for replacement. The cartridge seal is also expected to have a useful life double that of the old design. The reactor coolant pump design also includes a spool piece to facilitate separation and replacement of the motor from the pump.

C. Reactor Vessel Insulation

Insulation in the area of the reactor vessel nozzle welds is fabricated in sections with a thin reflective metallic sheet covering and quick disconnect clasps to facilitate removal of the insulation for inspection of the welds.

D. Steam Generators

The Model F steam generators incorporate several design features to facilitate maintenance and inspection in reduced radiation fields. The tube ends are designed to be flush with the tube sheet in the steam generator channel head to eliminate a potential crud trap. The steam generator manways (entrance to channel head) are sized to facilitate entrance and exit with protective clothing. Handholes to the secondary side are positioned to facilitate maintenance operations. Changes to increase steam generator reliability will also reduce occupational radiation exposure. Such changes include improved steam generator tube support plates (stainless steel and quatrefoil flow holes) and the use of all-volatile treatment (AVT) chemistry and the installation of a full flow condensate polishing demineralizer system on the secondary side.

E. Reactor Coolant Pipe Connections

To minimize crud buildup in branch lines, piping connections to the reactor coolant loops are located on or above the horizontal centerline of the pipe wherever possible.

F. Heat Exchangers

The heat exchangers are designed so that the shell-to-tube sheet joint need not be broken for inspection.

The shell and tube assembly can be lifted intact above the channel head to expose the tube ends for inspection and testing for leaks.

G. Valves

Valves are of the bolted body-to-bonnet forging type. This permits the use of ultrasonic testing in place of radiography for inspection and facilitates assembly and disassembly. This reduces inspection and maintenance time. Additionally, manual valves under 2 in. in diameter are designed for zero stem leakage.

H. Pumps

Pumps are designed with flanged connections to facilitate removal for maintenance. Depending on expected conditions, either canned pumps or pumps with high quality mechanical seals are used to reduce leakage and maintenance requirements.

I. Demineralizers

Demineralizer resin screens are constructed for substantially higher than normal pressure differential to accommodate higher than design flows without breakdown.

J. Filters

Filters are designed to be removable from the top with lifting bails in the middle of the head. The filter assemblies have bolt lead-ins for tool entry, and the filters are contained in a disposable cage assembly. These features facilitate remote removal, disposal, and assembly.

K. Materials

Equipment specifications for components exposed to high temperature reactor coolant contain specific limitations on the cobalt impurity content of the base metal as given in table 12.3.1-2, thereby controlling the potential for production of radioactive cobalt-60 from the base metal impurity cobalt-59. The estimated surface area of material in contact with the reactor coolant is given in table 12.3.1-3. The use of hard facing material with cobalt content such as stellite is limited to applications where its use is necessary for reliability considerations. Table 12.3.1-4 shows the estimated total surface area of stellite in the reactor coolant system (RCS). Nickel-based alloys in the RCS (cobalt-58 is produced from activation of the base metal nickel-58) are similarly used only where component reliability may be compromised by the use of other materials. The major use of nickel-based alloys in the RCS is the Inconel steam generator tubes. The surface area in contact with the RCS is given in table 12.3.1-3. From tables 12.3.1-3 and 12.3.1-4, it can be seen that the Inconel surface is the predominant area in contact with the RCS and that the stellite area is minimal.

L. Improved Head Closure System

The head closure system includes quick disconnect/connect stud tensioners which have quick-acting, hydraulically operated stud gripper devices as opposed to conventional tensioners which must be threaded onto the top of the studs. Also provided are six air motor-driven stud removal tools which can rapidly remove or insert the studs unlike the much slower manual stud removal and insertion used in the previous design. The stud tensioners are designed to operate simultaneously as are the stud removal tools.

12.3.1.1.2 Common Facility and Layout Designs for ALARA

This paragraph describes the design features utilized for standard type plant process and layout situations. These features are employed in conjunction with the general equipment described in paragraph 12.3.1.1.1 and include the features discussed in the following paragraphs.

A. Valve Galleries

Valve galleries are provided with shielded entrances for personnel protection. Floor drains are provided to control radioactive leakage. To facilitate decontamination in valve galleries, concrete surfaces are covered with a smooth surface coating which will allow easy decontamination.

B. Piping

Pipes carrying radioactive materials are routed through controlled access areas properly zoned for that level of activity. Each piping run is individually analyzed to determine the potential radioactivity level and surface dose rate. Where it is necessary that radioactive piping be routed through corridors or other low radiation zone areas, shielded pipeways are provided. Whenever practicable, valves and instruments are not placed in radioactive pipeways. Equipment compartments are used as pipeways only for those pipes associated with equipment in the compartment.

When possible and practical, radioactive and nonradioactive piping are separated to minimize personnel exposure. Should maintenance be required, provision is made to isolate and drain radioactive piping and associated equipment. Potentially radioactive piping is located in appropriately zoned and restricted areas. Process piping is monitored to ensure that access is controlled to limit exposure (section 12.5).

Piping is designed to minimize low points and dead legs. Drains are provided on piping where low points and dead legs cannot be eliminated. In radioactive systems, the use of nonremovable backing rings in the piping joints is prohibited. Whenever possible, branch lines having little or no flow during normal operation are connected above the horizontal midplane of the main pipe.

Piping which carries resin slurries or evaporator bottoms is run vertically as much as possible. Horizontal runs carrying spent resin are sloped toward the spent resin tanks. Large radius bends are utilized instead of elbows. Where sloped lines or large radius bends are impractical, adequate flush and drain capability is provided to prevent flow blockage and minimize crud traps.

C. Penetrations

To minimize radiation streaming through penetrations, as many penetrations as practicable are located with an offset between the source and the accessible areas. If offsets are not practicable, penetrations are located as far as possible above the floor elevation to reduce the exposure to personnel. If these two methods are not used, alternate means are employed, such as baffle shield walls or grouting the area around the penetration. As a result of radiation streaming into the electrical penetration areas, the radiation doses to personnel and equipment in these areas is higher than the balance of the control building. Post-accident radiation zones for the control building are shown on drawings 1X6DD100, 1X6DD101, AX6DD100, AX6DD101, AX6DD102, AX6DD103, AX6DD104, AX6DD105, AX6DD106, AX6DD107, AX6DD108, AX6DD109, AX6DD110, AX6DD111, AX6DD112, AX6DD113, AX6DD114, AX6DD115, AX6DD116, AX6DD117, AX6DD118, AX6DD119, and AX6DD120.

D. Contamination Control

Access control and traffic patterns are considered in the basic plant layout to minimize the spread of contamination. Equipment vents and drains from highly radioactive systems are piped directly to the collection system instead of allowing any contaminated fluid to flow across to the floor drain. All-welded piping systems are employed on contaminated systems to the maximum extent practicable to reduce system leakage and crud buildup at joints. The valves in some radioactive systems are provided with leakoff connections piped directly to the collection system.

Decontamination of potentially contaminated areas and equipment within the plant is facilitated by the application of decontaminable paints and suitable smooth-surface coatings to the concrete floors and walls. Floor drains with properly sloping floors are provided in all potentially contaminated areas of the plant. In addition, radioactive and potentially radioactive drains are separated from nonradioactive drains.

In controlled access areas where contamination is expected, radiation monitoring equipment is provided (subsection 12.3.4). Those systems that become highly radioactive, such as the spent resin lines in the radwaste system, are provided with flush and drain connections.

The role of the ventilation systems in minimizing the spread of airborne contamination is discussed in subsection 12.3.3.

E. Equipment Layout

In those systems where process equipment is a major radiation source (such as fuel pool cleanup, coolant and boric acid recycle, chemical waste, and miscellaneous waste) pumps, valves, and instruments are separated from the process component. This allows servicing and maintenance of these items in reduced radiation zones. In general, control panels are located in low radiation zones.

Major components such as tanks, demineralizers, and filters in radioactive systems are isolated in individual shielded compartments insofar as practical. Labyrinth entranceway shields or shielding doors are provided for each compartment from which radiation could stream or scatter to access areas and exceed the radiation zone dose limits for those areas. For potentially high radiation components (such as ion exchangers or heat exchangers and tanks in the primary coolant system) completely enclosed shielded compartments with hatch openings or removable concrete block walls are used. Provision is made on some major plant components for removal of these components to lower radiation zones for maintenance.

Equipment in nonradioactive systems which requires lubrication is located in lower radiation areas. Wherever practicable, lubrication of equipment in radiation areas is achieved with the use of tube-type extensions to reduce exposure during maintenance.

Exposure from routine in-plant inspection is controlled by locating, whenever possible, inspection points in properly shielded low-background radiation areas. Radioactive and nonradioactive systems are separated as far as practicable to limit radiation exposure from routine inspection of nonradioactive systems. For radioactive systems, emphasis is placed on adequate space and ease of motion in a properly shielded inspection area. Where longer times for routine inspection are required and permanent shielding is not feasible, sufficient space for portable shielding is provided. For example, a remotely operated device is provided for inservice inspections of the reactor vessel. When this is not practicable, written procedures are used which reduce the total time personnel are exposed to the radiation field. Also, access to high radiation areas is under the direct supervision of the unit health physics personnel.

F. Field Run Piping

Radioactive process piping is routed dimensionally on orthographic drawings by the plant architect/engineer. Fabrication isometrics of radioactive process piping are reviewed by the plant architect/engineer to ensure that adequate shielding is provided.

12.3.1.2 Radiation Zoning and Access Control

Access to areas inside the plant structures and plant yard area is regulated and controlled by radiation zoning and access control (section 12.5). Each radiation zone defines the radiation level range to which the aggregate of all contributing sources must be attenuated by shielding. During plant operation, personnel normally gain access to radiation controlled areas through designated access control points determined by health physics.

All plant areas are categorized into radiation zones according to expected radiation levels and anticipated personnel occupancy with consideration given toward maintaining personnel exposures ALARA and within the standards of 10 CFR 20. Each room, corridor, and pipeway of every plant building is evaluated for potential radiation sources during normal, shutdown, spent resin transfer, and emergency operations; for maintenance occupancy requirements; for general access requirements; and for material exposure limits to determine appropriate zoning. The radiation zone categories employed and their descriptions are given in table 12.3.1-1. The zoning for each plant area under normal conditions is shown on drawings 1X6DD001, 1X6DD002, 1X6DD003, 1X6DD004, 2X6DD001, 2X6DD002, 2X6DD003, AX6DD001, AX6DD002, AX6DD003, AX6DD004, AX6DD005, AX6DD006, AX6DD007, AX6DD008, AX6DD009, AX6DD010, AX6DD011, AX6DD012, AX6DD013, AX6DD014, AX6DD015, AX6DD016, AX6DD017, AX6DD018, AX6DD019, AX6DD020, AX6DD021, AX6DD022, AX6DD023, AX6DD024, AX6DD025, AX6DD026, AX6DD027, AX6DD028, AX6DD029, AX6DD030, AX6DD031, and AX6DD032. The zoning for each plant under accident conditions is shown on drawings 1X6DD100, 1X6DD101, AX6DD100, AX6DD101, AX6DD102, AX6DD103, AX6DD104, AX6DD105, AX6DD106, AX6DD107, AX6DD108, AX6DD109, AX6DD110, AX6DD111, AX6DD112, AX6DD113, AX6DD114, AX6DD115, AX6DD116, AX6DD117, AX6DD118, AX6DD119, and AX6DD120. Radiation zones shown in the figures are based upon conservative design data. Actual in-plant zones and control of personnel access will be based upon surveys conducted by health physics, as described in section 12.5.

In accordance with Section II.B.2 of NUREG-O737, a radiation and shielding design review was performed to identify vital areas and equipment. Areas which may require occupancy to permit an operator to aid in the long-term recovery from an accident are considered as vital. Vital areas include the control room, technical support center, safety-related motor control centers and switchgear in the control building, auxiliary building, diesel generator building, auxiliary feedwater pumphouse, radiochemistry laboratory, and the remote shutdown panels. Projected dose rates for these vital areas at various times after an accident are given in table 12.3.1-5. VEGP is designed to ensure the capability to achieve cold shutdown without subjecting personnel to excessive radiation exposure. This capability is further described in section 7.4. Radiation protection design features and access controls are described in sections 12.3 and 12.5. In the event that entry is desired into areas where excessive radiation exposures may occur, due consideration is given to the dose rates defined on drawings 1X6DD100, 1X6DD101, AX6DD100, AX6DD101, AX6DD102, AX6DD103, AX6DD104, AX6DD105, AX6DD106, AX6DD107, AX6DD108, AX6DD109, AX6DD110, AX6DD111, AX6DD112, AX6DD113, AX6DD114, AX6DD115, AX6DD116, AX6DD117, AX6DD118, AX6DD119, AX6DD120, and table 12.3.1-5, and appropriate time limits for presence in the area are imposed.

Ingress or egress of plant operating personnel to radiation controlled areas is controlled by the plant health physics staff to ensure that radiation levels and exposures are within the limits prescribed in 10 CFR 20. Any area having a radiation level that could result in an individual receiving a dose equivalent in excess of 5 mrem in 1 hour at 30 cm (12 in.) from the radiation source or from any surface which the radiation penetrates will be posted "Caution, Radiation Area." Any area having a radiation level that could result in an individual receiving a dose equivalent in excess of 100 mrem in 1 hour at 30 cm (12 in.) from the radiation source or from any surface which the radiation penetrates will be barricaded and posted "Caution, High Radiation Area" or "Danger, High Radiation Area." High radiation areas with radiation levels greater than 1000 mrem/h at 30 cm (12 in.) from the radiation source or from any surface which the radiation penetrates are provided with locked or alarmed barriers. For individual high radiation areas accessible to personnel with radiation levels of greater than 1000 mrem/h at 30 cm (12 in.) that are located within large areas where no enclosure exists for purposes of locking, and where no enclosure can be reasonably constructed around the individual area, that individual area shall be barricaded, conspicuously posted, and a flashing light shall be activated as a warning device. During periods when access to a high radiation area is required, positive control is exercised over each individual entry. Any area having a radiation level that could result in an individual receiving an absorbed dose in excess of 500 rads in 1 hour at 1 meter from the radiation source or from any surface which the radiation penetrates will be posted "Grave Danger, Very High Radiation Area." Measures taken to control access to very high radiation areas will meet the intent of Draft Regulatory Guide 8.38. To the extent practicable, the measured radiation level and the location of the source is posted at the entry to any radiation area or high radiation area.

Posting of radiation signs, control of personnel access, and use of alarms and locks are in compliance with requirements of 10 CFR 20.1601 and 1902. The flow of personnel is shown on drawings 1X6DD200, 1X6DD201, 1X6DD202, 1X6DD203, 2X6DD200, 2X6DD201, 2X6DD202, AX6DD200, AX6DD201, AX6DD202, AX6DD203, AX6DD204, AX6DD205, AX6DD206, AX6DD207, AX6DD208, AX6DD209, AX6DD210, AX6DD211, AX6DD212, AX6DD213, AX6DD214, AX6DD215, AX6DD216, AX6DD217, AX6DD218, AX6DD219, AX6DD220, AX6DD221, AX6DD222, AX6DD223, AX6DD224, AX6DD225, and AX6DD226.

Entry into high radiation areas is controlled by the Technical Specifications.

12.3.2 SHIELDING

The bases for the nuclear radiation shielding and the shielding configurations are discussed in this subsection.

12.3.2.1 <u>Design Objectives</u>

The objective of the plant radiation shielding is to reduce personnel and population exposures, in conjunction with a program of controlled personnel access to and occupancy of radiation areas, to levels that are within the dose standards of 10 CFR 50 and are as low as reasonably achievable (ALARA) within the dose standards of 10 CFR 20. Shielding and equipment layout and design are considered in ensuring that exposures are kept ALARA during anticipated personnel activities in areas of the plant containing radioactive materials, utilizing the design recommendations given in Regulatory Guide 8.8, paragraph C.3, where practicable.

Four plant conditions are considered in the nuclear radiation shielding design:

A. Normal, full-power operation.

B. Shutdown operation.

C. Spent resin transfer (radwaste areas and radwaste portion of the auxiliary building only).

D. Emergency operations (for required access to safety-related equipment).

The shielding design objectives for the plant during normal operation (including anticipated operational occurrences), for shutdown operations, and for emergency operations are listed below:

A. To ensure that radiation exposure to plant operating personnel, contractors, administrators, visitors, and approximate site boundary occupants are ALARA and within the limits of 10 CFR 20.

B. To ensure sufficient personnel access and occupancy time to allow normal anticipated maintenance, inspection, and safety-related operations required for each plant equipment and instrumentation area.

C. To reduce potential equipment neutron activation and to mitigate the possibility of radiation damage to materials.

D. To provide sufficient shielding for the control room so that for design basis accidents (DBAs) the direct dose plus the inhalation dose (calculated in chapter 15) will not exceed the limits of 10 CFR 50, Appendix A, General Design Criterion 19.

12.3.2.2 General Shielding Design

Shielding is provided to attenuate direct radiation through walls and penetrations and scattered radiation to less than the upper limit of the radiation zone for each area shown on drawings 1X6DD001, 1X6DD002, 1X6DD003, 1X6DD004, 2X6DD001, 2X6DD002, 2X6DD003, AX6DD001, AX6DD002, AX6DD003, AX6DD004, AX6DD005, AX6DD006, AX6DD007, AX6DD008, AX6DD009, AX6DD010, AX6DD011, AX6DD012, AX6DD013, AX6DD014, AX6DD015, AX6DD016, AX6DD017, AX6DD018, AX6DD019, AX6DD020, AX6DD021, AX6DD022, AX6DD023, AX6DD024, AX6DD025, AX6DD026, AX6DD027, AX6DD028, AX6DD029, AX6DD030, AX6DD031, and AX6DD032. General locations of the plant areas and equipment discussed in this subsection are shown in the general arrangement drawings of section 1.2. Design criteria for penetrations are consistent with the recommendations of Regulatory Guide 8.8 and are discussed in paragraph 12.3.1.1.2.

The material used for most of the plant shielding is ordinary concrete with a bulk density of approximately 137 lb/ft^3. Whenever poured-in-place concrete has been replaced by concrete blocks, design ensures protection on an equivalent shielding basis as determined by the density of the concrete block selected. Water is used as the primary shield material for areas above the spent fuel storage area.

12.3.2.2.1 Containment Shielding Design

During reactor operation, the containment protects personnel occupying adjacent plant structures and yard areas from radiation originating in the reactor vessel and primary loop components. The concrete containment wall and the reactor vessel and steam generator compartment shield walls reduce radiation levels outside the containment to less than 0.25 mrem/h from sources inside containment. The containment wall is a reinforced, prestressed concrete structure completely surrounding the nuclear steam supply system. The wall and dome are of a minimum 3 ft 9 in. thickness.

For DBAs, the containment shield and the control room shielding reduce the plant radiation intensities from fission products inside the containment to acceptable emergency levels, as defined by 10 CFR 50, Appendix A, General Design Criterion 19, for the control room. (See paragraph 12.3.2.2.7.)

Where personnel locks and equipment hatches or penetrations pass through the containment wall, additional shielding is provided to attenuate radiation to the required level defined by the outside radiation zone during normal operation and shutdown and to acceptable emergency levels as defined by 10 CFR 50, Appendix A, General Design Criterion 19, during DBAs.

12.3.2.2.2 Containment Interior Shielding Design

During reactor operation, many areas inside the containment are Zone V and normally inaccessible. However, shielding is provided to reduce dose rates to approximately 100 mrem/h or less in areas of the containment that potentially require access at power. These are the Zone IV or lower areas shown on drawings 1X6DD001, 1X6DD002, 1X6DD003, 1X6DD004, 2X6DD001, 2X6DD002, 2X6DD003, AX6DD001, AX6DD002, AX6DD003, AX6DD004, AX6DD005, AX6DD006, AX6DD007, AX6DD008, AX6DD009, AX6DD010, AX6DD011, AX6DD012, AX6DD013, AX6DD014, AX6DD015, AX6DD016, AX6DD017, AX6DD018, AX6DD019, AX6DD020, AX6DD021, AX6DD022, AX6DD023, AX6DD024, AX6DD025, AX6DD026, AX6DD027, AX6DD028, AX6DD029, AX6DD030, and AX6DD031.

The main sources of radiation are the reactor vessel and the primary loop components, consisting of the steam generators, pressurizer, reactor coolant pumps, and associated piping. The reactor vessel is shielded by the concrete primary shield and by the concrete secondary shield which also surrounds other primary loop components. Air cooling is provided to prevent overheating, dehydration, and degradation of the shielding and structural properties of the primary shield.

The primary shield is a large mass of reinforced concrete surrounding the reactor vessel and extending upward from the containment floor to form a portion of the walls of the fuel transfer canal. The minimum concrete thickness of the primary shield is 8 ft. The primary shield meets the following objectives:

A. In conjunction with the secondary shield, to reduce the radiation level from sources within the reactor vessel and reactor coolant system (RCS) to allow limited access to the containment during normal, full-power operation.

B. After shutdown, to limit the radiation level from sources within the reactor vessel, to permit limited access to the reactor vessel, and to permit limited access to the RCS equipment.

C. To limit neutron activation of component and structural materials.

The secondary shield is a reinforced concrete structure surrounding the RCS equipment, including piping, pumps, and steam generators. This shield protects personnel from the direct gamma radiation resulting from reactor coolant activation products and fission products carried away from the core by the reactor coolant. In addition, the secondary shield supplements the primary shield by attenuating neutron and gamma radiation escaping from the primary shield. The secondary shield is sized to allow limited access to the containment during full-power operation. The minimum thickness of secondary shield walls is 3 ft.

Components of the letdown portion of the chemical and volume control system (CVCS) in the containment are located in shielded compartments that are normally Zone V, restricted access areas. Shielding is provided for each piece of equipment in the letdown system consistent with its postulated maximum activity (subsection 12.2.1) and with the access and zoning requirements of adjacent areas. This equipment includes the regenerative heat exchanger, the excess letdown heat exchanger, and the letdown lines.

After shutdown, the containment is accessible for limited periods of time and all access is controlled. Areas are surveyed to establish allowable working periods. Dose rates are expected to range from 0.5 to 1000 mrem/h, depending on the location inside the containment (excluding reactor cavity). These dose rates result from residual fission products and neutron activation products (components and corrosion products) in the RCS.

Spent fuel is the primary source of radiation during refueling. Because of the high activity of the fission products contained in the spent fuel elements, extensive shielding is provided for areas surrounding the refueling pool and the fuel transfer canal to ensure that radiation levels remain below zone levels specified for adjacent areas. Water provides the shielding over the spent fuel assemblies during fuel handling.

12.3.2.2.3 Auxiliary Building Shielding

During normal operations, the major components in the auxiliary building with potentially high radioactivity are those in the CVCS, steam generator blowdown, boron recycle, liquid radwaste, gaseous radwaste, and spent resin handling systems. Shielding is provided for each piece of equipment consistent with its postulated maximum activity (sections 11.1, 11.2, 11.3, and 12.2) and with the access and zoning requirements of adjacent areas. (See drawings 1X6DD001, 1X6DD002, 1X6DD003, 1X6DD004, 2X6DD001, 2X6DD002, 2X6DD003, AX6DD001, AX6DD002, AX6DD003, AX6DD004, AX6DD005, AX6DD006, AX6DD007, AX6DD008, AX6DD009, AX6DD010, AX6DD011, AX6DD012, AX6DD013, AX6DD014, AX6DD015, AX6DD016, AX6DD017, AX6DD018, AX6DD019, AX6DD020, AX6DD021, AX6DD022, AX6DD023, AX6DD024, AX6DD025, AX6DD026, AX6DD027, AX6DD028, AX6DD029, AX6DD030, and AX6DD031.)

Depending on the equipment in the compartments, the radiation zones vary from Zone II through V. Corridors are generally shielded to allow Zone II access, and operator areas for valve compartments are generally Zone II or III for access.

Removable sections of block shield walls and concrete plugs are utilized as necessary for equipment maintenance and spent filter cartridge replacement. Permanent or temporary shielding is used between equipment in compartments with more than one piece of equipment to permit access for maintenance. Where necessary, labyrinth entrances with provisions for adequate ingress and egress for equipment maintenance and inspection are provided and are designed to be consistent with the access and zoning requirements of adjacent areas.

Following reactor shutdown, the residual heat removal (RHR) system pumps and heat exchangers are in operation to remove heat from the RCS. The radiation levels in the vicinity of this equipment will temporarily reach Zone V levels due to corrosion and fission products in the reactor water. Shielding is provided to attenuate radiation from RHR equipment during shutdown cooling operations to levels consistent with the radiation zoning requirements of adjacent areas. The shielding around the RHR equipment is a minimum of 2.5 ft of concrete.

12.3.2.2.4 Fuel Handling Building Shielding Design

The concrete shield walls surrounding the spent fuel cask loading and storage area, and the shield walls surrounding the fuel transfer and storage areas, are sufficiently thick to limit radiation levels outside the shield walls in all accessible areas to Zone II. The building external walls are sufficient to shield external plant areas to Zone I.

All spent fuel removal and transfer operations are performed under borated water to provide radiation protection and maintain subcriticality. Nominal water depths above the active fuel during fuel handling are 10 ft in the reactor cavity and 10 ft in the fuel transfer canal and spent fuel pool. This limits the dose at the water surface to less than 2.5 mrem/h for an assembly in a vertical position. The minimum water depth in the spent fuel pool is a nominal 13 ft above the top of the active fuel in the storage racks; for this depth the dose rate at the water surface is less than 15 mrem/h. Normal water depth above the stored assemblies is about 26 ft, and for this depth the dose rate at the pool surface is significantly less than 2.5 mrem/h. The minimum 4 ft thick concrete walls of the fuel transfer canal and spent fuel pool walls supplement the water shielding and limit the maximum radiation dose levels in working areas to less than 2.5 mrem/h. In the event that fuel is damaged and fuel fragments or pellets are collected on filters or other devices, the shielding requirements for the movement of the filters govern rather than the requirements on the submergence of spent fuel.

The spent fuel cooling and purification (SFCP) system shielding (section 9.1) is based on the maximum activity discussed in subsection 12.2.1 and the access and zoning requirements of adjacent areas. Equipment in the SFCP system to be shielded includes the SFCP heat exchangers, pumps, and piping.

12.3.2.2.5 Turbine Building Shielding Design

Radiation shielding is not required for any process equipment located in the turbine building. Space has been provided so that shielding may be added around the condensate polishing demineralizers if they become radioactive.

12.3.2.2.6 Control Room Shielding Design

The design basis loss-of-coolant accident (LOCA) dictates the shielding requirements for the control room. Consideration is given to shielding provided by the containment structure. Shielding combined with other engineered safety features is provided to permit access and occupancy of the control room following a postulated LOCA, so that radiation doses are limited to 5 rem whole body from contributing modes of exposure for the duration of the accident, in accordance with 10 CFR 50, Appendix A, General Design Criterion 19.

The design basis LOCA is described in subsection 15.6.5. The contribution from direct radiation from airborne fission products inside the containment to personnel doses inside the control room following a postulated LOCA is discussed in section 15A.3. The shielding of the control room ensures compliance with 10 CFR 50, Appendix A, General Design Criterion 19.

The parameters used in the demonstration of control room habitability, in addition to those contained in Regulatory Guide 1.4, are listed in chapter 15. Control room ventilation system parameters are provided in section 6.4. Drawing AX6DD404 provides an isometric view of the control room shielding.

12.3.2.2.7 Miscellaneous Plant Areas and Plant Yard Areas

Sufficient shielding is provided for all plant buildings containing radiation sources so that radiation levels at the outside surfaces of the buildings are maintained below Zone I levels. Plant yard areas that are frequently occupied by plant personnel are fully accessible during normal operation and shutdown. These areas are surrounded by a security fence and closed off from areas accessible to the general public. Access to outside storage tanks that have a contact dose rate greater than 0.25 mrem/h is restricted by a concrete shield wall sufficiently high so that dose rates to personnel in plant yard areas are limited to less than 0.25 mrem/h. Access to the low-level waste storage facility is restricted by a fence and locked gate with shielding as necessary to assure that dose rates to personnel in plant yard areas surrounding the low-level waste storage facility are limited to less than 0.25 mrem/h.

12.3.2.2.8 Spent Fuel Transfer Tube Shielding

The spent fuel transfer tube (drawings 1X2D48J006, 1X2D48A050, and AX2D09A017) is shielded to within local radiation zone limits. This is primarily achieved through the use of permanent shielding. The only removable shielding consists of concrete or steel hatches which reduce radiation in accessible areas to within those levels prescribed in the normal operation radiation zone maps (drawings 1X6DD001, 1X6DD002, 1X6DD003, 1X6DD004, 2X6DD001, 2X6DD002, 2X6DD003, AX6DD001, AX6DD002, AX6DD003, AX6DD004, AX6DD005, AX6DD006, AX6DD007, AX6DD008, AX6DD009, AX6DD010, AX6DD011, AX6DD012, AX6DD013, AX6DD014, AX6DD015, AX6DD016, AX6DD017, AX6DD018, AX6DD019, AX6DD020, AX6DD021, AX6DD022, AX6DD023, AX6DD024, AX6DD025, AX6DD026, AX6DD027, AX6DD028, AX6DD029, AX6DD030, and AX6DD031.)

The removable hatch in the fuel handling building consists of two concrete blocks and provides access to the bellows inspection room outside containment. The hatch inside containment consists of seven laminated steel plates and provides access to the bellows inspection room inside containment. The opening of these hatches is administratively controlled; these hatches are in place during the fuel transfer operation. This will ensure that there will be no access to the spent fuel transfer tube during refueling. With the hatches in place, all accessible areas will be as indicated on the radiation zone maps.

The spent fuel transfer tube in the seismic gap between the containment wall and the internal containment structure and in the seismic gap between the containment wall and the fuel handling building is shielded by a combination of concrete, steel, and permanently installed lead-loaded silicone foam rubber to maintain radiation zone limits for normal operation. Therefore, there is no unshielded portion of the spent fuel transfer tube during the refueling operation.

12.3.2.3 Shielding Calculational Methods

The shielding thicknesses provided to ensure compliance with plant radiation zoning and to minimize plant personnel exposure are based on maximum equipment activities under the plant operating conditions described in chapter 11 and section 12.2.

The thickness of each shield wall surrounding radioactive equipment is determined by approximating as closely as possible the actual geometry and physical condition of the source or sources. The isotopic concentrations are converted to energy group sources using data from standard references.(1,5)

The geometric model assumed for shielding evaluation of tanks, heat exchangers, filters, ion exchangers, evaporators, and the containment is a finite cylindrical volume source. For shielding evaluation of piping, the geometric model is a finite shielded cylinder. In cases where radioactive materials are deposited on surfaces such as pipe, the latter is treated as an annular cylindrical surface source.

The methods and equations of the Reactor Shielding Design Manual (6) and the Engineering Compendium on Radiation Shielding (8) are used to calculate dose rates. Buildup is calculated using Taylor coefficients presented in ORNL-RSIC-10; (7).

In addition, two industry-accepted computer codes--ANISN(9) and QAD(10)--are used for shielding analysis. ANISN is a multigroup one-dimensional discrete ordinates transport code that solves the one-dimensional Boltzmann transport equation for neutrons and gamma rays in slab, sphere, or cylinder geometry. Using a finite-difference technique, ANISN allows general anisotropic scattering; i.e., an Lth order Legendre expansion of the scattering cross-sections. Monte Carlo techniques as described below may be used for more complicated geometries such as penetrations. QAD is a point-kernel general purpose code for estimating the penetration of gamma rays and neutrons that originate in a volume-distributed source. ANISN is used primarily for primary shield design, and QAD is used for configurations not conveniently modeled in one-dimensional geometrics.

For final design, a three-dimensional model is used to simulate radiation streaming from the reactor vessel surface to the containment using the MORSE Monte Carlo program. (11) The source terms used for the MORSE code are generated by the computer code DOT.(12) The source terms are divided into 13 neutron energy groups and 14 gamma energy groups.

The shielding thicknesses are selected to reduce the aggregate computed radiation level from all contributing sources below the upper limit of the radiation zone specified for each plant area.

Shielding requirements in each plant area are evaluated at the point of maximum radiation dose through any wall. In addition, for shielding design purposes the concrete density of 135 lb/ft^3 was assumed. Therefore, the actual anticipated radiation level in each plant area is less than this maximum dose and consequently less than the radiation zone upper limit.

Where shielded entryways to compartments containing high-radiation sources are necessary, labyrinths are designed using methods summarized in ORNL-RSIC-21.(13) The labyrinths are constructed so that the scattered dose rate, plus the transmitted dose rate through the shield wall from all contributing sources, is below the upper limit of the radiation zone specified for each plant area.

12.3.2.4 References

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12.3.3 VENTILATION

The plant heating, ventilating, and air-conditioning (HVAC) systems are designed to provide a suitable environment for personnel and equipment during normal operation, during anticipated events of moderate frequency, and during certain infrequent events. Parts of the plant HVAC systems perform safety-related functions.

12.3.3.1 Design Objectives

The plant HVAC systems for normal plant operation, anticipated events of moderate frequency, and certain infrequent events are designed to meet the requirements of 10 CFR 20, Standards for Protection Against Radiation, and 10 CFR 50, Licensing of Production and Utilization Facilities.

12.3.3.2 Design Criteria

Design criteria for the plant HVAC systems include the following.

A. During normal operation, anticipated events of moderate frequency, and certain infrequent events, the average and maximum airborne radioactivity levels to which plant personnel are exposed in radiation controlled areas of the plant are as low as reasonably achievable (ALARA) and within the limits specified in 10 CFR 20. The average and maximum airborne radioactivity levels outside radiation controlled areas of the plant during normal operation, events of moderate frequency, and certain infrequent events are ALARA and within the limits of Appendix B, Table II of 10 CFR 20.1 - 20.601.

B. During normal operations, anticipated events of moderate frequency, and certain infrequent events, the dose from concentrations of airborne radioactive material in unrestricted areas beyond the site boundary is ALARA and within the limits specified in 10 CFR 20.1 - 20.601 and 10 CFR 50, Appendix I.

C. The plant siting dose guidelines of 10 CFR 100 will be satisfied following those hypothetical accidents described in chapter 15.

D. The dose to control room personnel shall not exceed the limits specified in General Design Criterion 19 of Appendix A to 10 CFR 50 following those hypothetical accidents described in chapter 15.

12.3.3.3 Design Guidelines

To accomplish the design objectives and to conform to the design criteria, the following design guidelines are employed wherever practicable.

12.3.3.3.1 Guidelines to Minimize Airborne Radioactivity

A. Access control and traffic patterns are considered in the basic plant layout to minimize the spread of contamination.

B. Equipment vents and drains are piped directly to a collection device connected to the collection system. This is to prevent any contaminated fluid from flowing across the floor to a floor drain.

C. All welded piping systems are employed on systems containing radioactive fluids to the maximum extent practicable. If welded piping systems are not employed, drip trays are provided at the points of potential leakage. Drains from drip trays are piped directly to the collection system.

D. Suitable coatings are applied to the concrete floors and walls of potentially contaminated areas to facilitate decontamination.

E. To minimize the amount of airborne radioactivity as a result of valve leakage, 2 1/2 in. and larger valves in radioactive systems are provided with a leakage control system consisting of double sets of packing with lantern rings and with leakoff connections that are piped to drain headers or a reduced packing configuration and the leakoff connection capped. Diaphragm or bellows seal valves are used on those systems where essentially no leakage can be tolerated.

F. Design of equipment incorporates features that minimize the spread of radioactivity during maintenance operations. These features include, but are not limited to, flush connections on pump casings for draining and flushing the pump prior to maintenance and flush connections on piping systems that could become highly radioactive.

12.3.3.3.2 Guidelines to Control Airborne Radioactivity

A. The airflow is directed from areas with lesser potential for contamination to areas with greater potential for contamination.

B. In building compartments with a potential for contamination, the exhaust is designed for greater volumetric flow than is supplied to that area. This minimizes the amount of uncontrolled exfiltration from the area.

C. Consideration is given to the possible disruption of normal airflow patterns by maintenance operations, and provisions are made in the design to prevent adverse airflow direction.

D. The air cleaning system's design, maintenance, and testing criteria are discussed in detail (in response to Regulatory Guides 1.52 and 1.140) in sections 1.9, 6.4, and 9.4 and in subsections 6.2.5 and 6.5.1. An illustrative example of an air cleaning system design is given in paragraph 12.3.3.5.

E. Air being discharged from potentially contaminated areas in the containment, fuel handling building, and the auxiliary building is passed through high efficiency particulate air filters and charcoal adsorbers to remove particulates and halogens. Means are provided to isolate the affected areas in the containment, fuel handling building, and auxiliary building upon indication of contamination. This minimizes the discharge of contaminants to the environment and in-plant exposures.

F. Means are provided to isolate the control room to minimize inleakage of contaminated air to protect personnel.

G. Suitable containment isolation valves are installed in accordance with General Design Criteria 54 and 56, including valve controls, to ensure that containment integrity is maintained. See additional discussion in subsection 6.2.4.

H. Redundant Seismic Category 1 systems and/or components are provided for portions of the ventilation system that serve areas required for safe shutdown of the reactor plant. Included herein are the plant control room and selected engineered safety feature equipment rooms.

I. Atmospheric tanks which contain radioactive materials are vented to the respective building ventilation system for filtration prior to release.

12.3.3.3.3 Guidelines to Minimize Personnel Exposure from HVAC Equipment

A. The guidelines of Regulatory Guide 8.8 have been utilized, as practicable, in the design of the plant ventilation systems.

B. Ventilation fans and filters are provided with adequate access space to permit servicing with minimum personnel radiation exposure. The HVAC system is designed to allow rapid replacement of components. Filter-adsorber unit conformance with the recommendations of Regulatory Guides 1.52 and 1.140 for access and service requirements is summarized in section 1.9.

C. Ventilation ducts are designed to minimize the buildup of radioactive contamination within the ducts.

D. Ventilating air is recirculated in clean areas only. Exhaust from potentially contaminated areas is filtered and then discharged.

E. Access and service of ventilation systems in potentially radioactive areas is provided by component location to minimize operator exposure during maintenance, inspection, and testing as follows.

1. The outside air supply units and building exhaust system components are enclosed in ventilation equipment rooms. These equipment rooms are accessible to the operators. Work space is provided around each unit for anticipated maintenance, testing, and inspection.
2. Local cooling equipment servicing the normal building requirements is located in areas of low contamination potential. (See drawings 1X6DD001, 1X6DD002, 1X6DD003, 1X6DD004, 2X6DD001, 2X6DD002, 2X6DD003, AX6DD001, AX6DD002, AX6DD003, AX6DD004, AX6DD005, AX6DD006, AX6DD007, AX6DD008, AX6DD009, AX6DD010, AX6DD011, AX6DD012, AX6DD013, AX6DD014, AX6DD015, AX6DD016, AX6DD017, AX6DD018, AX6DD019, AX6DD020, AX6DD021, AX6DD022, AX6DD023, AX6DD024, AX6DD025, AX6DD026, AX6DD027, AX6DD028, AX6DD029, AX6DD030, and AX6DD031.)

12.3.3.4 Design Description

The ventilation systems serving the following structures are considered to be potentially radioactive and are discussed in detail in subsections 6.2.2, 6.2.5, and 6.5.1 and in section 9.4.

- Containment building. (See subsections 6.2.2, 6.2.5, and 6.5.1.)
- Auxiliary building. (See subsection 9.4.3.)
- Fuel handling building. (See subsection 9.4.2.)

Although the control room is considered to be a nonradioactive area, radiation protection is provided to ensure habitability. (See section 6.4 and subsection 9.4.1.)

Other structures; e.g., pump intake structures, the administrative building, etc., contain no potential source of radioactivity and are not addressed in this chapter.

12.3.3.5 Air Cleaning Units

The guidance and recommendations of Regulatory Guides 1.52 and 1.140 concerning maintenance and inplace testing provisions for atmospheric cleanup systems, air filtration, and adsorption units have been used as a reference in the design of the various ventilation systems. The extent to which Regulatory Guide 1.52 and 1.140 have been followed is discussed in section 1.9. Drawing AX4DJ3101 shows the typical layout of an air cleaning unit.

Provisions specifically included to minimize personnel exposures and to facilitate maintenance or inplace testing operations are as follows.

A. The loading of the filters and adsorbers with radioactive material during normal plant operation is a slow process. Therefore, in addition to monitoring for pressure drop, the filters are checked for radioactivity on a scheduled maintenance basis with portable equipment. The filter elements are replaced before the radioactivity level is of sufficient magnitude to create a personnel hazard. No shielding is provided since it is not required for the level of radioactivity developed during normal operation. In case of excessive radioactivity caused by a postulated accident, the whole filter is replaced before normal personnel access is resumed. It will not be necessary for workers to handle filter units immediately after a design basis accident, so exposures can be minimized by allowing the short-lived isotopes to decay before changing the filter.

B. Active components of the atmospheric cleanup systems are designed to permit ready removal.

C. Access to active components is direct from working platforms to simplify element handling. Ample space is provided on the platforms for accommodating safe personnel movement during replacement of components, including the use of necessary material handling facilities and inplace testing operations.

D. HEPA filter banks which are more than three filter units high, where each filter is 2 ft by 2 ft, have a platform to facilitate access to the upper filters.

E. The clear space for doors is a minimum of 20 in. by 50 in.

F. The filters are designed with replaceable units that are clamped in place against compression seals. The filter housing is designed, tested, and proven to be airtight with bulkhead type doors that are closed against compression seals.

12.3.4 AREA RADIATION AND AIRBORNE RADIOACTIVITY MONITORING INSTRUMENTATION

The radiation monitoring system consists of the following:

A. Area radiation monitoring system (ARMS).

B. Process and effluent radiation monitoring system PERMS).

C. Sampling system.

D. Post-accident monitoring systems (PAMS) radiation monitors.

The PERMS, sampling systems, and PAMS radiation monitors are described in section 11.5.

12.3.4.1 <u>Area Radiation Monitoring</u>

The ARMS is provided to supplement the personnel and area radiation survey provisions of the plant health physics program described in section 12.5 and to ensure compliance with the personnel radiation protection guidelines of 10 CFR 20, 10 CFR 50, 10 CFR 70, and Regulatory Guides 1.97, 8.2, 8.8, and 8.12.

The design of the fuel pool racks precludes criticality under all postulated normal and accident conditions. Therefore, criticality monitors, as stated in 10 CFR 70.24 and Regulatory Guide 8.12, are not needed.

The area radiation monitors (ARM) installed in the alternate radwaste building (ARB) monitor radiation levels in and around the ARB. The design objectives of these ARMs are described in paragraph 12.3.4.1.10.

The ARM, which are installed in the radwaste processing facility, monitor radiation levels in the radwaste processing facility process and dressout areas during liquid radwaste processing. The design objectives of the ARM are described in paragraph 12.3.4.1.11.

12.3.4.1.1 Design Objectives

The design objectives of the ARMS during normal operating plant conditions and anticipated operational occurrences are:

A. To furnish records of radiation levels in specific areas of the plant.

B. To warn of uncontrolled or inadvertent movement of radioactive material in the plant.

C. To provide local and remote indication of ambient gamma radiation and local and remote alarms at key points where substantial change in radiation levels might be of immediate importance to personnel frequenting the area.

D. To annunciate and warn of possible equipment malfunctions and leaks in specific areas of the plant.

E. To furnish information for making radiation surveys.

By meeting the above objectives, the area radiation monitoring system aids health physics personnel in keeping radiation exposures as low as reasonably achievable (ALARA).

The design objectives of the ARMS during postulated accidents are:

A. To provide the capability to alarm and initiate a containment ventilation isolation signal in the event of a loss-of-coolant accident (LOCA), fuel handling accident inside containment, or abnormally high radiation inside the containment (monitors RE-0002, RE-0003). In modes 1 through 4, two channels of radiation monitors are required to be operable to ensure radiation monitoring instrumentation necessary to initiate containment ventilation isolation. In mode 6 during core alterations or movement of irradiated fuel assemblies in containment, automatic or system level manual CVI capability is not required and the required channels provide input to control room alarms to ensure prompt operator action to manually close the containment purge and exhaust valves. (See subsection 6.2.4.)

B. To provide long-term post-accident monitoring of conditions at strategic locations. (See subsection 11.5.5.)

12.3.4.1.2 Criteria for Location of Area Monitors

The locations of the area monitors are shown in the flow of personnel drawings (drawings 1X6DD200, 1X6DD201, 1X6DD202, 1X6DD203, 2X6DD200, 2X6DD201, 2X6DD202, AX6DD200, AX6DD201, AX6DD202, AX6DD203, AX6DD204, AX6DD205, AX6DD206, AX6DD207, AX6DD208, AX6DD209, AX6DD210, AX6DD211, AX6DD212, AX6DD213, AX6DD214, AX6DD215, AX6DD216, AX6DD217, AX6DD218, AX6DD219, AX6DD220, AX6DD221, AX6DD222, AX6DD223, AX6DD224, AX6DD225, and AX6DD226).

Considerations for area monitor locations are based on the following:

A. Frequency and length of personnel occupancy of a specific area.

B. Potential for personnel to unknowingly receive high radiation doses.

C. Potential for equipment malfunction.

D. Access areas where, during normal plant operation (including refueling), radiation exposures could exceed the radiation limits due to system failure or personnel error.

E. Access areas where new and spent fuel is received and stored.

F. Containment area for indicating the level of radioactivity and detecting the presence of fission products due to a reactor coolant pressure boundary (RCPB) leak, or fuel handling accident.

G. Normally or potentially radioactive areas.

12.3.4.1.3 General System Description

The area radiation monitors are located at selected locations throughout the plant to detect, indicate, and store information through their associated data processing module on the radiation levels and, if necessary, annunciate abnormal radiation conditions. The ARMS monitors are an integral part of the PERMS, which is described in detail in section 11.5.

The ARMS consists of individual, locally mounted area monitors. Each monitor is composed of the requisite number of channels, with a channel consisting of a radiation detector and check source, except for monitors RE-0005 and RE-0006 which have "keep-alive sources" instead of check sources, and the radwaste processing monitors ARE-16971, ARE-16972, and ARE-16973 which have “no pulse” detection. The detectors for all area monitors are either gamma-sensitive Geiger-Mueller counter tubes or ionization chambers. If exposed to radiation in excess of full-scale indication, the area monitors indicate that the full-scale reading has been exceeded and remain at the full-scale value. If the radiation field causing the overload condition is removed, the system returns to its normal operating condition unless the detector has failed. An administrative procedure (positioning the check source) is initiated to ensure that radiation monitoring equipment has not been damaged. All channels, except for the alternate radwaste building monitors and radwaste processing facility monitors, associated with a monitor are served by a local dedicated data processing module. All channels are indicated and annunciated in the control room and, with the exception of containment area monitors, indicated and alarmed near the detector location (normally at the data processing module) and the alternate radwaste building monitors which alarm in the alternate radwaste building control room. Monitors RE-0002, RE-0003, RE-0005, and RE-0006, which are safety-related, Class 1E, are also indicated and annunciated at the safety-related display console. The radwaste processing facility monitors are indicated and alarmed locally near the detector in the radwaste processing facility local control room, and in the Unit 1 main control room.

12.3.4.1.4 Data Processing Module and Display Console

A description of these components is given in section 11.5.

12.3.4.1.5 Local Annunciation

All area monitors, except those in containment, have local annunciation consisting of an audible alarm rated at 80 dB at 10 ft and a warning light at the local readout.

12.3.4.1.6 Power Supplies

Each channel is provided with an independent power supply, designed such that a failure in that channel does not affect any other channel, except for radwaste processing facility monitors which share a common microprocessor assembly. Monitors that are identified as safety related are redundant and are supplied power from the plant 120-V safety-related buses. Power to the channels that monitor only normal operations is supplied from the regulated 120-V instrumentation bus that is backed by the diesel generator.

12.3.4.1.7 Redundancy, Diversity, and Independence

Monitors designated safety related are part of the safety-related portion of the PERMS and are designed for redundancy, diversity, and independence in accordance with Institute of Electrical and Electronic Engineers (IEEE) 344-1975, IEEE 336-1971, IEEE 279-1971, IEEE 308-1974, IEEE 323-1974, and IEEE 384-1974. All monitors which are Seismic Category 1 are also manufactured and rated to the above standards.

12.3.4.1.8 Area Monitor Description

Table 12.3.4-1 gives the conditions of service for the area monitors. A brief description of each area monitor's function is given below.

A. Control Room Area Monitor RE-0001

   To continuously indicate the radiation levels in the control room. A high alarm signal warns control room personnel of a deteriorated radiological condition inside the control room.

B. Containment Low-Range Area Monitors RE-0002 and RE-0003

   To continuously indicate the radiation levels inside the containment building at the operating deck. During refueling operations a high radiation alarm indicates a fuel drop accident and isolates the containment ventilation system. During power operations, a high radiation alarm indicates a possible LOCA and isolates the containment ventilation system.

C. Fuel Handling Building Area Monitor RE-0008

   To continuously indicate the radiation levels inside the fuel handling building. A high radiation alarm signal warns the occupants of the fuel handling building of a deteriorated radiological condition.

D. Sampling Room Area Monitor ARE-0007B

   To continuously indicate the radiation levels in the sampling room. A high radiation alarm signal warns the occupants of the sampling room of a deteriorated radiological condition.

E. Seal Table Room Area Monitor RE-0011

   To continuously indicate the radiation levels in the seal table room and establish radiological habitability prior to entry. A high radiation alarm signal warns occupants of the seal table room of a deteriorated radiological condition.

F. Containment Access Hatch Area Monitor RE-0004

To continuously indicate the radiation levels in the containment access hatch and establish radiological habitability prior to entry.

G. Containment High Range Area Monitors RE-0005 and RE-0006

To indicate, along with RE-0002 and RE-0003, the radiation levels inside the containment building at the operating deck following a design basis accident.

H. Technical Support Center Area Monitor

Upon activating the Technical Support Center, a portable area radiation monitor is used to continuously indicate the radiological condition of the area. A high alarm signal provides warning to personnel of a deteriorating radiological condition. The monitor is placed in a manned area which is closest to the principal filtration units adjacent to the TSC. The portable unit provides an alarm point of 5 millirem per hour and dose rate indication up to 100 millirem per hour.

I. Radwaste Processing Facility Area Monitor ARE-16971 (North Wall Near the High Integrity Container (HIC) Storage Area)

To continuously indicate the radiation levels at the north wall of the radwaste processing facility process area near the subterranean HIC storage area. A high alarm alerts personnel in the area of a deteriorating radiological condition.

J. Radwaste Processing Facility Area Monitor ARE-16972 (East Wall Near Deminerilizer Subterranean Vault Area)

To continuously indicate the radiation levels at the east wall of the radwaste processing facility process area near the subterranean demineralizer vault area. A high alarm alerts personnel in the area of a deteriorating radiological condition.

K. Radwaste Processing Facility Area Monitor ARE-16973 (Dressout Area)

To continuously indicate the radiation levels in the dressout area of the radwaste processing facility. A high alarm alerts personnel in the area of a deteriorating radiological condition.

12.3.4.1.9 Range and Alarm Setpoints

The range and control function of the PERMS area monitors are given in table 12.3.4-2.

Alarm setpoints are controlled by plant procedures and the Offsite Dose Calculation Manual (ODCM) where appropriate.

Radiation zones for VEGP are described in table 12.3.1-1.

With the exception of area monitors RE-0002, RE-0003, RE-0004, RE-0005, RE-0006, and RE-0011, all of the area monitors are located in radiation Zones I or II.

The control room monitor RE-0001 has a greater sensitivity than the other area monitors, since it is located in a Zone I radiation area; monitors RE-0002, RE-0003, RE-0005, and RE-0006 cover a wide range of radiation levels. During plant shutdown including refueling operations, the radiation level on and above the operating deck should be less than 2.5 mR/h. The high end of the range is dictated by the design basis accident, a LOCA.

Each area monitor has two alarm setpoints, intermediate and high. If a monitor has a control function; i.e., RE-0002 and RE-0003, the control function is triggered coincidentally with the high alarm setpoint. An intermediate alarm gives a visual indication in the control room and near the detector, except for containment area monitors which have no local alarm or indicator, that the radiation level has reached the intermediate setpoint. A high alarm gives both a visual and audible indication in the control room and near the detector, except for containment area monitors which have no local alarm indicators, that the high alarm setpoint has been reached. The radwaste processing facility monitors' alarm gives visual and audible indication near the detector in the radwaste processing facility local control room, and in the Unit 1 main control room.

For testing, each area monitor has a check source assembly (except for RE-0005 and RE-0006, which have keep-alive sources) which is operated from the safety-related display console for the safety-related monitors and from the communications display computer and PERMS display computer for the nonsafety-related monitors, and uses a sealed source. Inservice inspection, calibration, and maintenance of the ARMS monitors is discussed in paragraph 11.5.2.5.

12.3.4.1.10 Design Objectives for the Radwaste Processing Facility Area Radiation Monitors

The design objectives of the radwaste processing facility ARM during normal liquid radwaste processing and anticipated operational occurrences are:

A. To warn of uncontrolled or inadvertent movement of radioactive material in the radwaste processing facility.

B. To provide local and remote (in the radwaste processing facility control room and Unit 1 main control room) indication (audible and visual) of ambient gamma radiation levels in the radwaste processing facility.

C. To annunciate and warn of possible equipment malfunctions and radioactive leaks in the radwaste processing facility.

D. To furnish information for making radiation surveys.

By meeting the above objectives, the ARM aid health physics personnel in keeping radiation exposure as low as reasonably achievable (ALARA).

12.3.4.2 Standard Review Plan Evaluation

The VEGP did not utilize American National Standards Institute/ American Nuclear Society (ANSI/ANS) HPSSC-6.8.1-1981. Design of the VEGP ARMS began prior to the issuance in 1981 of ANSI/ANS HPSSC-6.8.1, Location and Design Criteria for Area Radiation Monitoring Systems for Light-Water Nuclear Reactors, and therefore the standard was not specifically addressed in the initial design stages. However, the ARMS is in conformance with other applicable regulations and guides. (See paragraph 12.3.4.1.)

Criticality monitors as defined in the Standard Review Plan are not included in the VEGP design. Paragraph 12.3.4.1 states that the design of the fuel pool racks precludes criticality under all postulated normal and accident conditions and that therefore criticality monitors are not needed. Supporting information of this subject is given in subsections 9.1.1, New Fuel Storage, and 9.1.2, Spent Fuel Storage, and paragraph 4.3.2.6, Criticality of the Reactor During Refueling.

TABLE 12.3.1-1

RADIATION ZONES

Zone	Maximum Dose Rate (mrem/h)	Description[a]
I	≤ 0.25	Controlled access, unlimited occupancy.
II	≤ 2.5	Controlled access, limited occupancy.
III	≤ 15	Controlled access, limited occupancy.
IV	≤ 100	Controlled access, limited occupancy.
V	> 100	Restricted access, limited occupancy for very short periods. Access controlled as stated in the Technical Specifications.

a. Controlled access, unlimited occupancy areas: where entry and exit by plant employees and visitors are not under the direct supervision of the plant health physics staff. These areas can be occupied by plant personnel or visitors on an unlimited time basis with a minimum probability of health hazard from radiation exposure.

Controlled access, limited occupancy areas: Where higher radiation levels arid/or radioactive contamination, which have a greater probability of radiation health hazard to individuals, can be expected. Only individuals directly involved in the operation of the plant will, in general, be allowed to enter these areas. Entry and exit are authorized and supervised by the plant health physics staff.

Occupancy: The time spent by an individual in a particular area. Occupancy is to be determined on an area by area and individual by individual basis.

Restricted access, limited occupancy areas: Where extremely high radiation levels and/or radioactive contamination is expected. Only individuals directly involved in the operation of the plant will, in general, be allowed to enter these areas. These areas are normally inaccessible with locked doors and positive control of access. Entry and exit are under the supervision of the plant health physics staff.

TABLE 12.3.1-2

EQUIPMENT SPECIFICATION LIMITS FOR
COBALT IMPURITY LEVELS

Component	Material	Maximum Weight Percent Cobalt
Reactor internals (nonactive region)	SS	0.20
Reactor internals (active region)	SS	0.12
Reactor vessel clad	SS	0.20
Reactor coolant piping	SS	0.20
Reactor internal bolting material	SS	0.25
Reactor coolant pumps	SS	0.20
Pressurizer	SS	0.20
Steam generators	Inconel	0.10
Fuel (nonactive region)	SS	0.12
Fuel (active region)	SS	0.08
Fuel	Inconel	0.10
Fuel	Zircaloy/ ZIRLO®/ Optimized ZIRLO™	0.002

TABLE 12.3.1-3

APPROXIMATE REACTOR COOLANT SYSTEM
WETTED SURFACE AREAS

Component	Material	Surface Area (ft^2)
Reactor internals	SS	4236
Reactor vessel clad	SS	2190
Reactor coolant piping	SS	2758
Reactor internal bolting material	SS	Negligible
Reactor coolant pumps	SS	Negligible
Steam generators	Inconel	1.90×10^5
Fuel (nonactive region)	SS	2000
Fuel (active region)	SS	3600
Fuel	Inconel	7.80×10^3
Fuel	Zircaloy/ ZIRLO®/ Optimized ZIRLO™	7.78×10^4

TABLE 12.3.1-4

APPROXIMATE REACTOR COOLANT SYSTEM WETTED
SURFACE AREAS OF STELLITE

Component	Surface Area (ft^2)
Reactor internals	3.2
Reactor coolant pump journals	17.2
Control rod drive mechanisms	10.8
RCS valves	2.6

TABLE 12.3.1-5

POST-ACCIDENT RADIATION LEVELS IN VITAL AREAS (NUREG-0737, 11.B.2 SHIELDING REVIEW)

Building Room Description		Projected Dose Rate (rem/hr) 0 h	1 h	24 h	168 h	720 h
Auxiliary building						
D105	Switchgear train A	2.10E1	1.11E1	2.56E-1	5.82E-2	1.78E-2
C109	MCCIABD	3.80E2	1.00E2	2.45	5.24E-1	1.64E-1
B16	MCC1BBC Unit 1[(a)]	1.10	5.85E-1	1.35E-2	3.07E-3	9.40E-4
B122	MCC2BBC Unit 2	1.40E1	7.38	1.71E-1	3.89E-2	1.18E-2
116	1BBB motor control center train B	1.00E-1	5.27E-2	1.22E-3	2.77E-4	8.47E-5
118	1ABB motor control center train A	1.50E1	3.96	9.69E-2	2.07E-2	6.48E-3
207	Switchgear room	<4.4E-4	Nil	Nil	Nil	Nil
Control building						
B47	Train B channel 2	3.08E-3	1.62E-3	3.75E-5	8.53E-6	2.61E-6
B48	Train D channel 4	3.08E-2	1.62E-2	3.75E-4	8.53E-5	2.61E-5
B52	Train A channel 1	3.08E-2	1.62E-2	3.75E-4	8.53E-5	2.61E-5
B55	Train C channel 3	3.08E-3	1.62E-3	3.75E-5	8.53E-6	2.61E-6
B61	Switchgear train B	1.37E2	7.22E1	1.67	3.79E-1	1.16E-1
B76	Switchgear train A	3.47	1.83	4.23E-2	9.61E-3	2.94E-3
B79	Motor control center room	1.40E2	7.38E1	1.71	3.88E-1	1.19E-1
A43	Shutdown room train B	9.87	5.20	1.20E-1	2.73E-2	8.36E-3
A48	4.16-kV switchgear train A	6.00E-1	3.16E-1	7.32E-3	1.66E-3	5.08E-4
A50	4.16-kV switchgear train B	1.52E2	8.01E1	1.85	4.21E-1	1.29E-1
A54	13.8-kV switchgear	3.62E2	1.91E2	4.42	1.00	3.07E-1
A64	13.8-kV switchgear	4.56E2	2.40E2	5.56	1.26	3.86E-1
A75	Shutdown room train A	2.58E-2	1.36E-2	3.15E-4	7.15E-5	2.19E-5
A77	Motor control center room	2.42E1	1.28E1	2.95E-1	6.7OE-2	2.05E-2
Diesel generator building		1.09	5.75E-1	1.33E-2	3.02E-3	9.23E-4
Auxiliary feedwater building		2.65E-1	1.40E-1	3.23E-3	7.34E-4	2.24E-4
Control room, from containment[(b)]		3.8E-5	8.7E-5	5.6E-5	2.0E-5	6.1E-6
Technical support center, from containment and TSC filter units		2.44E-1	1.74E-1	1.27E-2	4.84E-3	4.68E-4

a. Doses to the equivalent Unit 2 room are similar unless noted otherwise.

b. Doses to the radiochemistry laboratory from direct radiation are similar to that seen in the control room.

TABLE 12.3.4-1 (SHEET 1 OF 2)

CONDITIONS OF SERVICE FOR AREA RADIATION MONITORS

Area Monitor	Detectors Per Unit	Operating Temperature (°F)	Pressure	Relative Humidity (%)	Radiation Zone	Safety Classification	Location/ Elevation (ft)
RE-0001 control room	1	65-85	-1/8 in. to +1/2 in. WG	50 (max)	I	NNS	Control room at 224.5
RE-0002, RE-0003 containment low range	2	60-120	-1.5 in. WG to +3 psig	17.7 to 50	V	SC-3/1E	Containment at 224.5
RE-0008 fuel handling building	1	40-104	-1/4 in. to 0 in. WG	20 to 95	II	NNS	Fuel handling building at 224.5
ARE-0007B sampling room	1 (shared)	65-85	0 psig	40 to 60	II	NNS	Sample room at 224.5
RE-0011 seal table instrumentation	1	60-120	-1.5 in. WG to +3 psig	17.7 to 50	IV	NNS	Seal table room at 197
RE-0004 containment access hatch	1	60-120	-1.5 in. WG to +3 psig	17.7 to 50	III	NNS	Inside containment personnel airlock at 227
RE-0005, RE-0006 containment high range[a]	2	60-120	-1.5 in. WG to +3 psig	17.7 to 50	V	SC-3/1E	Outside surface of shield wall at 224.5. 2-RE-0006 is located near the personnel airlock.
ARE-16971 Radwaste processing facility HIC storage (north wall)	1 (shared)	50-100	Slightly negative to 0 psig	100(max)	V	NNS	Radwaste processing facility north wall near subterranean HIC storage
ARE-16972 Radwaste processing facility demin vault (east wall)	1 (shared)	50-100	Slightly negative to 0 psig	100(max)	V	NNS	Radwaste processing facility east wall near subterranean demin vault
ARE-16973 Radwaste processing facility dressout area	1 (shared)	60-80	0 psig to slightly positive	20 to 95	II	NNS	Radwaste processing facility dressout area

TABLE 12.3.4-1 (SHEET 2 OF 2)

Area Monitor	Detectors Per Unit	Operating Temperature (°F)	Pressure	Relative Humidity (%)	Radiation Zone	Safety Classification	Location/ Elevation (ft)
ARE-16851 ARB west wall	1 (shared)	17-104	0 psig to-3 psig	100 (max)	I	NNS	Outside, west wall of the alternate radwaste building (ARB)
ARE-16852 ARB east wall	1 (shared)	17-104	0 psig to-3 psig	100 (max)	I	NNS	Outside surface of the shield wall on the east side of the ARB
ARE-16853 ARB interior	1 (shared)	50-120	Slightly negative to 0 psig	100 (max)	IV	NNS	South wall of the demineralizer vault in the ARB
ARE-16854 ARB dressout area	1 (shared)	60-80	0 psig to slightly positive	100 (max)	II	NNS	Dressout area for the ARB

a. These monitors are qualified for post-LOCA environment.

TABLE 12.3.4-2 (SHEET 1 OF 2)

RANGE AND CONTROL FUNCTIONS FOR AREA RADIATION MONITORS

Monitor	Range (mR/h)	Sensitivity (mR/h)	Control Function	Accuracy
RE-0001 control room	10^{-2} to 10^{3}	10^{-2}	No	± 20 percent of actual radiation field
RE-0002, RE-0003 containment low range	$4.5x10^{-1}$ to $5.4x10^{3}$ (both)	10^{-1}	Yes, isolates containment ventilation	± 20 percent of actual radiation field system
RE-0008 fuel handling building	10^{-1} to 10^{4}	10^{-1}	No	± 20 percent of actual radiation field
ARE-0007B sampling room	10^{-1} to 10^{4}	10^{-1}	No	± 20 percent of actual radiation field
RE-0011 seal table room	10^{-1} to 10^{4}	10^{-1}	No	± 20 percent of actual radiation field
RE-0004 containment access hatch	10^{-1} to 10^{4}	10^{-1}	No	± 20 percent of actual radiation field
RE-0005, RE-0006 (A and B) containment high range	10^{3} to 10^{11} (both)	10^{3}	No	± 20 percent of actual radiation field

TABLE 12.3.4-2 (SHEET 2 OF 2)

Monitor	Range (mR/h)	Sensitivity (mR/h)	Control Function	Accuracy
ARE-16971 Radwaste processing facility HIC storage (north wall)	10^{-3} to 10^{4}	10^{-3}	No	± 23 percent of actual radiation field
ARE-16972 Radwaste processing facility demin vault (east wall)	10^{-3} to 10^{4}	10^{-3}	No	± 23 percent of actual radiation field
ARE-16973 Radwaste processing facility dressout area	10^{-3} to 10^{4}	10^{-3}	No	± 23 percent of actual radiation field

12.4 DOSE ASSESSMENT

Radiation exposures in the plant are primarily due to direct radiation from components and equipment containing radioactive fluids. In addition, in some plant radiation areas there can be radiation exposure to personnel due to the presence of airborne radionuclides. Inplant radiation exposures during normal operation and anticipated operational occurrences are discussed in subsection 12.4.1.

A summary paragraph on radiation exposures due to direct radiation and airborne radioactivity at locations outside the plant structures is provided in subsection 12.4.2.

Radiation exposures to operating personnel will be within 10 CFR 20 limits. Radiation protection design features described in section 12.3 and the health physics program outlined in section 12.5 will ensure that the occupational radiation exposures (ORE) to operating personnel during operation and anticipated operational concurrences will be as low as reasonably achievable (ALARA).

Regulatory Guide 8.19 establishes a procedure for providing an estimate of annual dose to personnel. This procedure provides an assessment by specifying all actions and/or operations which could potentially lead to significant levels of exposure and estimating the frequency and anticipated dose levels for said actions. These data are then combined to formulate an annual plant dose.

While the method employed in the dose assessment of the VEGP provides the same information, it is assessed in a more generalized manner. For the initial dose estimates, rather than performing a breakdown of all operations concerned (routine maintenance operations, refueling, etc.), an estimate of the manhours required annually was performed for each type operation. These manhours were broken down by the percent of time spent in the five radiation zones, and a total occupancy time for each radiation zone was determined. With the expected average dose for each of the zones, an estimated annual exposure was obtained for each zone. Further discussion of this method is provided in subsection 12.4.1. After the plants are in operation, annual dose estimates will be based on previous VEGP history.

In addition, Regulatory Guide 8.19 establishes a means of verifying that the designs of specific areas and systems are consistent with the ALARA philosophy. However, rather than relying upon more limited information upon which the specific manpower and dose breakdown as recommended in Regulatory Guide 8.19 are based, VEGP utilizes the concept of the ALARA review. This review is a systematic review performed to ensure that the ALARA considerations of Regulatory Guide 8.8 are followed and provides the same design control as the method proposed in Regulatory Guide 8.19. For further discussion of the ALARA review, see paragraph 12.1.1.1.

12.4.1 EXPOSURES WITHIN THE PLANT STRUCTURES

12.4.1.1 Direct Radiation Dose Estimates

Annual man-rem doses from direct radiation during the performance of the routine functions, such as operation and surveillance, normal maintenance, radwaste handling, refueling, and inservice inspection, have been estimated, using the following bases:

A. Radiation exposure data from operating pressurized- water reactors (PWRs) are given in tables 12.4.1-1 through 12.4.1-11.[(1)(2)(3)]

B. Expected average dose rates in plant radiation areas are discussed in this section.

C. Expected occupancy times for various work function personnel in the different plant radiation areas are listed in table 12.4.1-12.

D. Anticipated manpower requirements for each unit are given in table 12.4.1-12.

E. Table 12.4.1-13 provides an estimate of the distribution of the annual man-rem according to routine functions.

The maximum and expected average dose rates in the plant radiation areas are given below:

Zone	Maximum Dose Rate (mrem/h)	Expected Average Dose (mrem/h)
I	0.25	0.06
II	2.5	1.0
III	15.0	6.0
IV	100.0	40.0
V	>100.0	100.0

The maximum dose rates are determined by shielding calculations based on conservative assumptions regarding sources, self-shielding, locations, etc. The expected dose rates are estimated by assuming a failed fuel percentage of 0.12, by assuming that stringent water chemistry control and improved design will minimize crud buildup and hence the expected dose rates in various radiation zones, and by the recognition that real maximum doses in a given zone are localized effects. The expected average doses given above are used in computing the doses for personnel involved in all operations, except inservice inspection and special maintenance.

The direct radiation dose estimates have been developed from exposure models for each of the major job categories within routine functions. Each exposure model has been developed by breaking the job into individual packages and identifying expected radiation fields, time spent in each radiation field, and the number of men required to carry out each package. Engineering judgment and feedback from operating plant experience have been used to define typical values for each parameter in the exposure model. As such, the resultant exposure estimates should be used as typical values, keeping in mind the variability of the input data from which the estimates were developed.

Initial estimated exposure to plant personnel from direct gamma radiation during the performance of routine functions is estimated to be approximately 418 man-rem/year/unit. Details of these initial man-rem estimates are given in table 12.4.1-13.

12.4.1.2 Airborne Radioactivity Dose Estimates

Due to leakages of radioactive fluids into the auxiliary, containment, radwaste, fuel handling, and turbine buildings, plant personnel are exposed to radionuclides released into the atmosphere of these buildings by the leaked fluids. These atmospheric contaminants contribute to the total body, thyroid, and lung doses.

The peak airborne concentrations for most areas in the plant are within the limits specified in 10 CFR 20. By use of appropriate respiratory equipment and/or limitation of occupancy time, personnel are allowed to enter areas where the airborne activity levels exceed 10 CFR 20 limits.

The expected annual doses to plant personnel from airborne radioactivity for each building in the plant are presented in table 12.4.1-14. The assumptions used to determine airborne radioactivity in each building, along with the airborne concentrations for all areas, are presented in subsection 12.2.2 and tables 12.2.2-1 and 12.2.2-2.

Doses resulting from airborne radioactivity are calculated by the methods discussed below using appropriate portions of 10 CFR 20.1 - 20.601; TID-14844[(4)]; recommendations of Regulatory Guides 1.4, 1.5, 1.21, 1.24, and 1.25; and the assumptions of ICRP-2[(5)]. Accident doses are discussed in chapter 15.

12.4.1.2.1 Method for Calculating the Total Body Gamma Dose

The total body gamma dose for isotopes other than tritium can be calculated using the following formulas:

$$TBD = \sum_i (C_i)\,(DCF_i)\,(t)\,(10^9)$$

where:

TBD = total body dose (rem/year).

C_i = equilibrium airborne concentration of each radioisotope (μCi/cm^3).

DCF_i = dose conversion factor for each isotope (mrem-m^3/pCi-h).

t = total exposure time in hours over 1 year.

10^9 = conversion factor (pCi-cm^3-rem/μCi-m^{3}mrem).

The dose conversion factors are tabulated in table 12.4.1-15.

12.4.1.2.2 Method for Calculating the Lung Dose

The lung dose for isotopes other than tritium can be calculated using the following formula:

$$D_{lung} = \sum_i \frac{(C_i)\ (MPD)_{lung}}{(MPC)_{a_{lung_i}}}\left(\frac{t}{250}\right)$$

where:

D_{lung} = lung dose (rem/year).

C_i = equilibrium airborne concentration of radioisotope i (μCi/cm^3).

$(MPC)_{a_{lung_i}}$ = maximum permissible concentration in air resulting in the MPD to the lung for radioisotope i (μCi/cm^3).

$(MPD)_{lung}$ = maximum permissible dose to lung (rem/year).

t = total exposure time in days (8-h basis).

The maximum permissible concentration in air and maximum permissible dose to the lungs are tabulated in table 12.4.1-16.

12.4.1.2.3 Method for Calculating the Inhalation Thyroid Dose

The thyroid dose can be calculated using the following formula:

$$D_{thyroid} = \sum_i (C_i)(t)(BR)(DCF)_i$$

where:

$D_{thyroid}$ = inhalation thyroid dose (rem/year).

C_i = equilibrium concentration of each radio-isotope (μCi/cm^3).

t = exposure time in days over 1 year.

BR = breathing rate (cm^3/day).

$(DCF)_i$ = dose conversion factors for each isotope (rem/μCi).

The breathing rate and dose conversion factors are presented in table 12.4.1-16.

12.4.1.2.4 Method for Calculating the Tritium Dose

The dose for tritium can be calculated using the following formula:

$$D_{tritium} = (4.12 \times 10^3)(C)(t)$$

where:

$D_{tritium}$ = dose caused by absorption of tritium by lungs and skin (rem/year).

C = equilibrium airborne concentration of tritium (μCi/cm^3).

t = exposure time in days over 1 year (8-h basis).

The factor 4.12×10^3 is derived from the breathing rate, tritium uptake coefficient, and other factors given in table 12.4.1-16. The model assumes that the dose from absorption through the skin is equal to the dose absorbed by the lungs.

12.4.1.2.5 In-Plant Radiation Monitoring Program

A program shall be maintained, the in-plant radiation monitoring program, which ensures the capability to accurately determine the airborne iodine concentration in vital areas under accident conditions. This program shall include the following:

A. Training of personnel.

B. Procedures for monitoring, and

C. Provisions for maintenance of sampling and analysis equipment.

12.4.1.3 Standard Review Plan Evaluation

Dose assessments should be made in accordance with the procedures stated in Regulatory Guide 8.19.

An evaluation of the VEGP dose assessment methodology is provided in the introductory paragraphs of section 12.4.

12.4.1.4 References

1. U.S. Nuclear Regulatory Commission, "Occupational Radiation Exposure at Light Water Cooled Power Reactors," 1969-1975, NUREG-0109, 1976.
2. U.S. Nuclear Regulatory Commission, "Occupational Radiation Exposure at Light Water Cooled Power Reactors," 1969-1974, NUREG-75/032, 1975.
3. U.S. Nuclear Regulatory Commission, "Occupational Radiation Exposure at Commercial Nuclear Power Reactors," 1981, NUREG-0713, vol 3, 1982.
4. DiNunno, J. J., et al, "Calculations of Distance Factors for Power and Test Reactor Sites," TID-14844, March 1962.
5. "Report of Committee II on Permissible Dose for Internal Radiation," ICRP Publication 2, Pergamon Press, New York, 1959.
6. U.S. Nuclear Regulatory Commission letter, "Request for Exemption from Portion of General Design Criteria 4 of Appendix A to 10 CFR 50 (Units 1 and 2)," (H. R. Denton to D. O. Foster), dated February 5, 1985.
7. Georgia Power Company letter, "Alternative Pipe Break Design Considerations," (D. O. Foster to H. R. Denton), dated April 2, 1984.

12.4.2 RADIATION EXPOSURE OUTSIDE THE PLANT STRUCTURES

12.4.2.1 Direct Radiation Dose Estimates

The direct radiation from the containment, the low-level waste storage facility, and the auxiliary and turbine buildings is negligible compared to that from the radwaste processing facility and outside storage tanks. The principal sources of radioactivity not stored in the plant structures are the radwaste processing facility, the reactor makeup water storage tanks, and the refueling water storage tanks. The tanks are surrounded by concrete walls that shield the yard area to Zone I limits (0.25 mrem/h). The radwaste processing facility utilizes interior and exterior shield walls to maintain yard levels to Zone I. The annual dose at the nearest site boundary, based on 8760-h occupancy, is less than 1 mrem.

12.4.2.2 Exposures Due to Airborne Radioactivity

Doses at the site boundary due to released activity are given in subsection 11.3.3.

12.4.3 EXPOSURES TO CONSTRUCTION WORKERS

12.4.3.1 Direct Radiation Dose Estimates

For the purpose of estimating the direct radiation exposure to construction workers in Unit 2 resulting from the operation of Unit 1, it is assumed:

A. The radiation field at the interface between Units 1 and 2 is 0.25 mrem/h, ie., the interface between the two units is Zone 1. Because there are no contained sources in the Unit 2 construction areas, this radiation level is reduced for most of the construction area due to distance from the interface and shielding provided by structures.

B. Construction manpower levels are as given in table 12.4.3-1 for the time period between the scheduled fuel loading of Unit 1 (September 1986) and the completion of Unit 2 construction (March 1988).

   1. The construction workers are onsite 60 h/week for 50 weeks/year.

   2. The construction workers spend 15 percent of their time in close proximity to the interface with Unit 1. The exposure rate for this time is 0.25 mrem/h (the full radiation field at the interface).

   3. The construction workers spend 85 percent of their time in areas where the radiation field is reduced by distance and/or shielding. A dose reduction factor of 5 is assumed, resulting in an exposure rate of 0.05 mrem/h.

Based on these assumptions, the anticipated annual exposures to Unit 2 construction workers are given in table 12.4.3-2.

The average annual exposure to an individual is 0.26 rem/year; however, as can be seen from table 12.4.3-1, the majority of the workers will be on the site for less than a year. Assuming that no new workers are brought onto the site, thus maximizing individual exposure estimates, results in an anticipated individual cumulative exposure of 0.36 rem over 18 months.

12.4.3.2 Exposures Due to Airborne Radioactivity

The construction worker doses (table 12.4.3-1) due to airborne radioactivity were calculated using the individual dose option in the GASPAR code, in conjunction with the near field dispersion and deposition data for each release point (table 12.4.3-3) and the projected craft requirements for future periods at Unit 2 construction activities.

12.4.4 DECOMMISSIONING DOSE ESTIMATES

The radiation protection design features (section 12.3) established for station operation will aid in maintaining occupational radiation exposure as low as reasonably achievable during decommissioning. The shielding design allows for efficient mothballing and entombment. Decommissioning by removal of all contaminated and activated equipment will be aided by remote handling of equipment, equipment layout (subsection 12.3.1), and administrative planning which includes the health physics program (section 12.5).

Specifications and limitations on cobalt content in equipment components will serve to limit radiation doses from crud buildup during both operation and subsequent decommissioning. A summary of the features in Westinghouse pressurized water reactors (PWRs) that reduce occupational exposure are given in WCAP-8872, Design, Inspection, Operation, and Maintenance Aspects of the Westinghouse NSSS to Maintain Occupational Radiation Exposures As Low As Reasonably Achievable, April 1977.

A National Environmental Studies Project(1) study analyzed the decommissioning alternatives for light-water reactors and high-temperature gas-cooled reactors. The majority of the estimated PWR occupational radiation exposure due to removal/dismantling comes from decontaminating the primary and radwaste systems. Estimated occupational radiation exposures from the study are given in table 12.4.4-1. The dominant radioactive isotopes are listed in table 12.4.4-2.

12.4.4.1 Reference

1. Atomic Industrial Forum, Inc., "An Engineering Evaluation of Nuclear Power Reactor Decommissioning Alternatives," AIF/NESP-009, 1976.

TABLE 12.4.1-1

AVERAGE NUMBER OF PERSONNEL PER PWR
UNIT FOR THE PERIOD 1969-1981[3]

Year	Number of Units	Average Number of Personnel/Unit
1969	3	151
1970	3	447
1971	4	226
1972	5	377
1973	12	787
1974	20	485
1975	26	419
1976	30	586
1977	34	614
1978	39	659
1979	42	924
1980	42	1101
1981	44	1076
1969-1981	Overall average	761

NOTE:

During the years 1969-1972, all plants reported collective doses, but a few did not submit the number of personnel that received measurable doses.

TABLE 12.4.1-2

AVERAGE OCCUPATIONAL RADIATION EXPOSURE (MAN-REM DOSE)
PER PWR UNIT FOR THE PERIOD 1969-1981[3]

Year	Number of Units	Average MREM Dose/Unit
1969	4	165
1970	4	684
1971	6	307
1972	8	464
1973	12	783
1974	20	331
1975	26	318
1976	30	460
1977	34	396
1978	39	429
1979	42	516
1980	42	578
1981	44	652
1969-1981	Overall average	488

NOTE:

During the years 1969-1972, all plants reported collective doses, but a few did not submit the number of personnel that received measurable doses.

TABLE 12.4.1-3 (SHEET 1 OF 2)

AVERAGE OCCUPATIONAL RADIATION EXPOSURE (MAN-REM DOSE) BASED ON PWR PLANT AGE(3)

Year of Operation (Range of Months)	No. of Units	Man-rem per Unit
1	35	193
2	42	348
3	41	445
4	37	434
5	34	457
6	28	461
7	25	529
8	17	779
9	10	782
10	6	875
11	5	441
12	2	650
13	2	1870
14	2	2130

Initial Operation Date	Units Considered	
Jan 1968	Haddam Neck	- X X X X X X X X X X X X X
Jan 1968	San Onofre 1	- X X X X X X X X X X X X X
July 1970	Ginna	X X X X X X X X X X X
Dec 1970	Point Beach 1	X X X X X X X X X X X
March 1971	Robinson 2	- X X X X X X X X X X
Dec 1971	Palisades	X X X X X X X X X X
Oct 1972	Point Beach 2	X X X X X X X X X
Dec 1972	Maine Yankee	X X X X X X X X X
Dec 1972	Surry 1	X X X X X X X X X
Dec 1972	Turkey Point 3	X X X X X X X X X
May 1973	Surry 2	X X X X X X X X
July 1973	Oconee 1	X X X X X X X X
Sept 1973	Turkey Point 4	X X X X X X X X
Sept 1973	Fort Calhoun	X X X X X X X X
Aug 1973	Indian Point 2	X X X X X X X X
Dec 1973	Zion 1	X X X X X X X X
Dec 1973	Prairie Island 1	X X X X X X X X
June 1974	Kewaunee	X X X X X X X
Sept 1974	Three Mile Island 1	X X X X X X X
Sept 1974	Zion 2	X X X X X X X
Sept 1974	Oconee 2	X X X X X X X
Dec 1974	Arkansas 1	X X X X X X X
Dec 1974	Oconee 3	X X X X X X X
Dec 1974	Prairie Island 2	X X X X X X X

TABLE 12.4.1-3 (SHEET 2 OF 2)

Year of Operation (Range of Months)	No. of Units	Man-rem per Unit
1	35	193
2	42	348
3	41	445
4	37	434
5	34	457
6	28	461
7	25	529
8	17	779
9	10	782
10	6	875
11	5	441
12	2	650
13	2	1870
14	2	2130

Initial Operation Date	Units Considered	
April 1975	Rancho Seco	- X X X X X X
May 1975	Calvert Cliffs 1	X X X X X X X
Aug 1975	Cook 1	X X X X X X X
Dec 1975	Millstone Point 2	X X X X X X X
May 1976	Trojan	X X X X X X
Aug 1976	Indian Point 3	X X X X X X
Oct 1976	Beaver Valley	X X X X X X
Dec 1976	St. Lucie	X X X X X X
March 1977	Crystal River 3	- X X X X
April 1977	Calvert Cliffs 2	- X X X X
June 1977	Salem	X X X X X
Nov 1977	Davis-Besse	X X X X X
Dec 1977	Farley	X X X X X
June 1978	North Anna 1	X X X X
June 1978	North Anna 2	X X X X
July 1978	Cook 2	X X X X
Dec 1978	Three Mile Island 2	X X X X
March 1980	Arkansas 2	- X

NOTE:

Only PWRs operating at power levels ≥ 450 MWe have been considered, with the exception of San Onofre 1 (430 MWe).

TABLE 12.4.1-4

DISTRIBUTION OF THE NUMBER OF PERSONNEL
(>100 MREM/YEAR)
ACCORDING TO WORK FUNCTION

Work Function	Percentage per NUREG-75/032 (for the Year 1974)[(2)]	Percentage per NUREG-0109 (for the Year 1975)[(1)]	Percentage per NUREG-0713 (for the Year 1981)[(3)]
Reactor operations	19.2	9.1	8.7
Routine maintenance	34.5	59.5	35.0
Inservice inspection	1.4	4.1	5.2
Special maintenance	28.7	16.4	39.3
Waste processing	2.1	7.6	6.9
Refueling	14.1	3.3	4.9
	100.0	100.0	100.0

TABLE 12.4.1-5

DISTRIBUTION OF PERSONNEL (>100 MREM/YEAR)
ACCORDING TO EMPLOYEE CATEGORY

Work Function	Percentage per NUREG-75/032 (for the Year 1974)[2]	Percentage per NUREG-0109 (for the Year 1975)[1]	Percentage per NUREG-0713 (for the Year 1981)[3]
Plant employees	47.4	34.7	22.7
Utility employees	18.1	6.1	9.2
Contract workers	34.5	59.2	68.0
	100.0	100.0	100.0

TABLE 12.4.1-6

PERCENTAGES OF PERSONNEL DOSE
BY WORK FUNCTION[(3)]

Work Function	Percentage of Dose						
	1975	1976	1977	1978	1979	1980	1981
Reactor operations and surveillance	10.8	10.2	10.5	13.3	12.2	9.5	8.7
Routine maintenance	52.6	31.0	28.1	31.5	29.2	35.5	35.0
Inservice inspection	3.0	6.0	6.4	7.7	9.0	5.5	5.2
Special maintenance	19.0	40.0	42.5	35.9	39.4	40.6	39.3
Waste processing	6.9	5.0	5.8	5.0	3.6	3.0	6.9
Refueling	7.7	7.9	6.7	6.6	6.6	6.1	4.9

TABLE 12.4.1-7

ANNUAL OCCUPATIONAL EXPOSURES FOR VARIOUS PWR VENDOR'S UNITS

Approximate Year of Operation	Westinghouse No. of Reactors	Westinghouse Man-Rem/ Year/Unit	CE No. of Reactors	CE Man-Rem/ Year/Unit	B and W No. of Reactors	B and W Man-Rem/ Year/Unit
1	23	193	6	110	6	277
2	27	357	7	460	8	218
3	26	442	7	567	8	346
4	23	464	7	348	7	420
5	21	457	7	508	6	394
6	18	536	5	296	5	329
7	17	580	3	535	5	350
8	13	851	3	591	1	404
9	8	872	2	424	-	-
10	5	869	1	902	-	-
11	5	441	-	-	-	-
12	2	650	-	-	-	-
13	2	1870	-	-	-	-
14	2	2130	-	-	-	-

NOTES:

Only PWRs operating at power levels 450 MWe have been considered, with the exception of San Onofre 1 (430 MWe).

The power plants for this table are the same as those appearing on table 12.4.1-3.

TABLE 12.4.1-8 (SHEET 1 OF 2)

AVERAGE ANNUAL OCCUPATIONAL EXPOSURE AND POWER
FOR INDIVIDUAL PWR[3]

Plant Name	Total Years of Operation (Approximate)	Full Power MWe (net)	Annual Average Percentage of Full Power	Annual Average Man-Rem	Comments
Haddam Neck	14	555	92	601	First year data not available. Highest annual exposures: 1979 - 1161 man-rem; 1980 - 1353 man-rem; 1981 - 1036 man-rem.
San Onofre	14	436	69	700	First year data not available. Highest annual exposures: 1980 - 2387 man-rem; 1981 - 3223 man-rem.
Ginna	11	470	73	626	Percentage of full power (about 52 percent, 51 percent) was low in 1974 and 1976. Highest annual exposures: 1972 - 1032 man-rem; 1974 - 1225 man-rem.
Robinson 2	11	665	75	863	1971 - only 53 percent of full power was produced. Highest annual exposure: 1980 – 1852 man-rem.
Palisades	10	635	51	588	1974 - only produced 1 percent of full power. Highest annual exposure: 1973 - 1133 man-rem.
Maine Yankee	9	810	69	294	Percentages of full power produced were low in 1973 (50 percent) and 1974 (53 percent).

TABLE 12.4.1-8 (SHEET 2 OF 2)

Plant Name	Total Years of Operation (Approximate)	Full Power MWe (net)	Annual Average Percentage of Full Power	Annual Average Man-Rem	Comments
Fort Calhoun	8	478	64	330	1975 - only 53 percent of full power was produced.
Kewaunee	7	512	84	146	________
Rancho Seco	7	873	60	285	1976 - only 31 percent of full power was produced.
Millstone Point 2	6	864	66	612	1978 - highest annual exposure: 1621 man-rem.
Trojan	5	1080	60	356	1978 - only 19 percent of full power was produced.
Beaver Valley	5	810	37	238	1980 - only 5 percent of full power was produced.
St. Lucie	5	777	79	478	________
Crystal River 3	5	782	53	462	1978 - only 40 percent of full power was produced.
Salem	4	1079	51	352	1979 - only 23 percent of full power was produced.
Davis-Besse	4	874	43	725	1980 - only 29 percent of full power was produced.
Farley	4	804	56	424	1979 - only 26 percent of full power was produced.

NOTES:

Only single unit PWRs have been considered.

Only plants that have been in operation for equal to or greater than 4 years were considered.

TABLE 12.4.1-9

AVERAGE INDIVIDUAL EXPOSURE BASED ON PWR PLANT AGE[3]

Year of Operation (approx.)	No. of Units	Average No. of Personnel/ Unit	Average Exposure per (man-rem/ year/unit)	Average Exposure per Individual (rem/year)
1	35	666	193	0.29
2	42	849	348	0.41
3	41	967	445	0.46
4	37	986	434	0.44
5	34	993	457	0.46
6	28	1124	461	0.41
7	25	1323	529	0.40
8	17	1443	779	0.54
9	10	1422	782	0.55
10	6	1268	875	0.69
11	5	832	441	0.53
12	2	1066	650	0.61
13	2	2640	1870	0.76
14	2	2393	2130	0.89

NOTE:

Units considered are the same as those shown on table 12.4.1-3.

TABLE 12.4.1-10 (SHEET 1 OF 2)

CUMULATIVE AVERAGE OF ANNUAL EXPOSURE BY YEARS OF OPERATION

	Years of Operation													
Plant	1	2	3	4	5	6	7	8	9	10	11	12	13	14
Haddam Neck	-	106	689	342	325	697	201	703	449	614	117	1161	1353	1036
San Onofre	-	42	155	50	256	353	71	292	880	847	401	139	2387	3223
Ginna	430	1032	224	1225	538	636	401	450	592	708	655			
Point Beach 1	164	580	294	1475	229.5	185	214.5	160	322	299	298			
Robinson 2	-	215	695	672	1142	715	455	963	1188	1852	783			
Palisades	78	1133	627	306	696	100	764	854	424	902				
Point Beach 2	294	1475	229.5	185	214.5	160	322	299	298					
Maine Yankee	117	420	319	85	245	420	154	462	424					
Surry 1	152	442	824.5	1582.5	1153.5	918.5	1792	1918	2122					
Turkey Point 3	78	227	438	592	518	516	840	825.5	1125.5					
Surry 2	442	824.5	1582	1153.5	918.5	1792	1918	2122						
Oconee 1	517	165.67	342	422.67	464.3	333.67	351.67	403.7						
Indian Point 2	455	352.5	975	267.5	501.5	639.5	485.5	1185.5						
Fort Calhoun	71	294	313	297	410	126	688	458						
Turkey Point 4	227	438	592	518	516	840	825.5	1125.5						
Zion 1	56	63.5	286	501.5	508.5	637	460	860						
Prairie Island 1	18	61.5	223.5	150	110.5	90	176.5	164.5						
Kewaunee	28	270	139	154	127	165	141							
Three Mile Island 1	73	286	359	504	696	197	188							
Oconee 2	165.67	342	442.3	464.33	333.7	351.7	403.7							
Zion 2	63.5	285.5	501.5	508.5	637	460	860							
Arkansas 1	21	289	256	189	369	342	551							
Oconee 3	165.67	342	423	464.33	333.7	351.7	403.7							
Prairie Island 2	61.5	223.5	150	110.5	90	176.5	164.5							

TABLE 12.4.1-10 (SHEET 2 OF 2)

CUMULATIVE AVERAGE OF ANNUAL EXPOSURE BY YEARS OF OPERATION

	Years of Operation													
Plant	1	2	3	4	5	6	7	8	9	10	11	12	13	14
Rancho Seco	-	58	390	323	126	412	402							
Calvert Cliffs 1	74	547	250	402.5	338.5	303.5								
Cook 1	116	299	336	359	246.5	327.5								
Millstone Point 2	168	242	1621	472	636	531								
Trojan	174	319	267	421	609									
Indian Point 2	535	1003	636	308	364									
Beaver Valley	87	190	132	553	229									
St. Lucie	152	337	438	532	929									
Crystal River 3	-	321	495	625	408									
Calvert Cliffs 2	-	250	402.5	338.5	303.5									
Salem	122	584	449	254										
Davis-Besse	48	30	154	58										
Farley 1	108	643	435	511										
North Anna 1	224.5	109	340											
North Anna 2	224.6	109	340											
Cook 2	359	246.5	327.5											
Three Mile Island 2	696	197	188											
Arkansas 2	-	551												
Total	6765	14616	18245	16058	15538	13369	13233	13245	7824	5245	2205	1300	3740	4259
No. of Units	35	42	41	37	34	20	25	17	10	6	5	2	2	2
Average	193	348	445	434	457	461	529	779	782	875	441	650	1870	2130

NOTE:

Plants considered are the same as those shown in table 12.4.1-3.

TABLE 12.4.1-11

OCCUPATIONAL RADIATION EXPOSURE ESTIMATE FOR INSERVICE INSPECTION AND SPECIAL MAINTENANCE (FOR ONE UNIT)[d]

Inservice Inspection	Average Expected Dose Rate (rem/h)	Total Time (man-h)	Frequency	Annual Dose Over 10-Year Period (man-rem)
Inservice inspection of reactor vessel and reactor coolant piping	0.100	520	1 per 10 years[a]	5.2
Insulation removal from reactor vessel nozzle and reactor coolant piping	0.100	200	1 per 10 years[a]	2.0
Snubber inspection of reactor coolant piping	0.005	2000	1 per 10 years[a]	1.0
Steam generator inservices inspection (4 steam generators)	0.125	470	1 per 10 years[a]	5.9
Steam generator eddy current inspection	0.015	600	(b)	1.5
Special Maintenance[c]				
Steam generator tube plugging	0.24	52	1 per 2 years	6.1
Steam generator tube plug welding	0.21	52	1 per 10 years	1.1
Sludge lancing	0.06	160	1 per 2 years	4.9
Control rod drive mechanism repair	0.50	5	1 per 5 years	0.5

a. Inservice inspection is performed over a 10-year period on a schedule defined in the Inservice Inspection Plan.

b. Per Regulatory Guide 1.83, eddy current inspection for steam generator tubes would occur at year 1, year 2, year 5, and year 8. Two steam generators are inspected at year 1 and one steam generator is inspected in subsequent years.

c. These data reflect dose estimate for projected special maintenance and repair tasks and do not include dose estimates for unique tasks that may be performed on a limited basis such as unforeseen major repair tasks or unusual inspection efforts.

d. Elimination of pipe whip restraints in the reactor coolant system main loop piping (GDC-4 exemption, Reference 6) results in a reduction of the occupational radiation exposure estimate as discussed in the safety balance shown in reference 7.

TABLE 12.4.1-12 (SHEET 1 OF 2)

PLANT STAFF CLASSIFICATION - WORKING TIME AND ZONE DISTRIBUTION
(VEGP UNITS 1 AND 2)[a]

Classification of Personnel	No.	Category[b]	Working h/Year	Percentage of Time Spent in Zone I	II	III	IV	V
Plant management	11	0	2000	97	1.5	1.0	0.5	-
Administration	30	0	2000	94	1.5	1.0	0.5	-
Administration	34	M	2000	95	2.5	1.6	0.9	-
Engineering	24	0	2000	94	1.5	1.0	0.5	-
Engineering	40	M	2000	95	2.5	1.6	0.9	-
Engineering	11	R	2000	90	5.0	4.2	0.7	0.1
Operations	60	O	2200	97	1.5	1.0	0.5	-
Operations	100	M	2200	95	2.5	1.6	0.9	-
Operations	32	R	2200	90	5.0	4.2	0.7	0.1
Operations	17	RH	2200	65	25.0	9.3	0.6	0.1
Maintenance	10	0	2200	97	1.5	1.0	0.5	-
Maintenance	151	M	2200	95	2.5	1.6	0.9	-
Maintenance	45	R	2200	90	5.0	4.2	0.7	0.1
Maintenance	4	RH	2200	65	25.0	9.3	0.6	0.1
Radiochemistry	23	0	2200	97	1.5	1.0	0.5	-
Radiochemistry	35	M	2200	95	2.5	1.6	0.9	-

TABLE 12.4.1-12 (SHEET 2 OF 2)

Classification of Personnel	No.	Category[b]	Working h/Year	Percentage of Time Spent in Zone				
				I	II	III	IV	V
Radiochemistry	15	R	2200	90	5.0	4.2	0.7	0.1
Radiochemisty	5	RH	2200	65	25.0	9.3	0.6	0.1
Training	15	0	2000	97	1.5	1.0	0.5	-

a. Does not include the doses due to inservice inspection and special maintenance which are respectively equal to 31.2 man-rem/year and 25.2 man-rem/year for Units 1 and 2 as shown in table 12.4.1-11.

b. Abbreviations:

O - operation
M - maintenance
R - refueling
RH - radwaste handling

TABLE 12.4.1-13

ESTIMATED ANNUAL GAMMA DOSE TO PLANT PERSONNEL[a]
(EXTRACTED NUMBER OF MANHOURS OF OCCUPANCY PER YEAR)[b]

Radiation Zone	Expected Average Dose Rate in Zone (mrem/h)	Operation	Maintenance[c]	Radwaste Handling	Refueling(c)	Total Manhours	Estimated Annual Exposure (man-rem)
I	0.06	375,002	738,340	37,180	201,960	1,352,482	81
II	1.0	5799	19,430	14,300	11,220	50,479	51
III	6.0	3866	12,435	5320	9425	31,046	186
IV	40.0	1933	6995	343	1571	10,842	434
V	100.0	-	-	57	224	281	28
	TOTALS	386,600	777,280	57,200	224,400	1,445,400	780

a. Detailed breakdown by job classification presented in table 12.4.1-12.

b. Does not include the doses due to inservice inspection and special maintenance which are respectively equal to 31.2 man-rem/year and 25.2 man-rem/year for Units 1 and 2 as shown in table 12.4.1-11.

c. Elimination of pipe whip restraints in the reactor coolant system main loop piping (GDC-4 exemption, Reference 6) results in a reduction of the occupational radiation exposure estimate as discussed in the safety balance shown in reference 7.

TABLE 12.4.1-14

DOSES TO PLANT PERSONNEL CAUSED BY AIRBORNE RADIOACTIVITY

Location	Assumed Occupancy (h/year)	Total Body Gamma Dose (man-rem/year)	Lung Dose (man-rem/year)	Inhalation Thyroid Dose (man-rem/year)	Airborne Tritium Dose (man-rem/year)
Auxiliary building corridor	2000 (40 h/week-50 weeks/year)	2.79E-2	3.93E-3	2.80E-1	2.55E-3
Auxiliary building letdown heat exchanger valve gallery	50	2.54E-2	3.64E-3	2.60E-1	2.37E-3
Turbine building	2000 (40 h/week-50 weeks/year)	5.52E-7	6.90E-6	4.96E-4	4.68E-4
Fuel building H-3 only	168 (56 h/week-3 weeks/year)	NA	NA	NA	2.16E-1
Containment (full power)	86	3.59E-1	6.20E-2	6.55E-1	6.67E-2
Containment (refuel) H-3 only	168 (56 h/week-3 weeks/year)	NA	NA	NA	2.16E-1

TABLE 12.4.1-15

TOTAL BODY DOSE FACTORS

Isotope	Total Body Dose Factor[a] (mrem/h per Pci/m^3)
I - 131	3.1×10^{-7}
I - 132	2.0×10^{-6}
I - 133	4.4×10^{-7}
I - 134	2.0×10^{-6}
I - 135	1.5×10^{-6}
Kr - 83m	8.6×10^{-12}
Kr - 85m	1.3×10^{-7}
Kr - 85	1.8×10^{-9}
Kr - 87	6.8×10^{-7}
Kr - 88	1.7×10^{-9}
Kr - 89	1.9×10^{-6}
Kr - 90	1.8×10^{-6}
Xe - 131m	1.0×10^{-8}
Xe - 133m	2.9×10^{-8}
Xe - 133	3.4×10^{-8}
Xe - 135m	3.6×10^{-7}
Xe - 135	2.1×10^{-7}
Xe - 137	1.6×10^{-7}
Xe - 138	1.0×10^{-6}
Ar - 41	1.0×10^{-6}

a. Obtained from WASH-1258, table D-3 for iodines, and from Regulatory Guide 1.109, table B-1 for noble gases.

TABLE 12.4.1-16 (SHEET 1 OF 2)

ASSUMPTIONS USED TO DETERMINE DOSES TO PLANT PERSONNEL
CAUSED BY AIRBORNE RADIOACTIVITY

Isotope	$(MPC)_a$ with the Lung as the Critical Organ[(a)] ($\mu Ci/cm^3$)	Dose to the Thyroid per μCi Inhaled (rem/μCi)[(b)]
I-131	3×10^{-7}	1.48×10^{-0}
I-132	7×10^{-6}	5.35×10^{-2}
I-133	1×10^{-6}	4.00×10^{-1}
I-134	2×10^{-5}	2.50×10^{-2}
I-135	3×10^{-4}	1.24×10^{-1}
Rb-86	7×10^{-8}	-
Co-60	9×10^{-9}	-
Co-58	5×10^{-8}	-
Fe-59	5×10^{-8}	-
Mn-54	4×10^{-8}	-
Sr-89	4×10^{-8}	-
Sr-90	5×10^{-9}	-

$(MPD)_{lung}$ = 15.0 rem/year[(c)]

Breathing rate = $(3.47 \times 10^{-4}\ m^3/s)(10^6\ cm^3/m^3)(3600\ s/h)$

= $1.25 \times 10^6\ cm^3/h$

Tritium Dose

Tritium uptake coefficient = 1

Tritium biological decay constant = $2.41 \times 10^3\ (h^{-1})$

Tritium body mass = 4.3×10^4 g

Tritium effective energy Mev/disintegration = 0.01

TABLE 12.4.1-16 (SHEET 2 OF 2)

Total tritium dose[d](rem/year) =

$$\frac{(2)(BR)(C)(t)(U)(E)(8h/day)}{(D)(M)} \times$$

$$\frac{(3600\,s/h)(1.6\times10^{-6}\ ergs/Mev)(3.7\times10^{4}\ disintegrations/s/\mu Ci)}{(100\ ergs/g/rad)(1\,rad/rem)}$$

$$= (2)\ (t)\ (C)\ (2.57 \times 10^{2})\ (8\ h/day)$$

$$= (4.12 \times 10^{3})\ (t)\ (C)$$

where:

C = air concentration ($\mu Ci/cm^3$).

BR = breathing rate (cm^3/day).

D = tritium biological decay constant (h^{-1}).

E = tritium effective energy (Mev/disintegration).

M = tritium body mass (g).

t = number of 8-h days per year that individual is exposed to tritium environment (day/year).

U = tritium uptake coefficient.

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a. "Report of Committee II on Permissible Dose for Internal Radiation," ICRP Publication 2, p 61.

b. "Calculation of Distance Factors for Power and Test Reactor Sites," TID-14844, p 25.

c. "Basic Radiation Protection Criteria," NCRP Report No. 39, p 92.

d. The factor of 2 used in this equation arises from the fact that the dose due to absorption through the skin is equal to the dose due to the absorption through the lung as stated in ICRP-10, Appendix C.

TABLE 12.4.3-1

CONSTRUCTION WORKER DOSES

Time Period	Craft Requirement	Direct Airborne (man-rem)	Total Body (man-rem)[a]	Skin (man-rem)
3/87[b] - 6/87	565	3.4 E+01	4.8 E-01	4.9 E-01
6/87 - 9/87	328	2.0 E+01	2.8 E-01	2.9 E-01
9/87 - 12/87	162	1.4 E-01	1.4 E-01	1.4 E-01
12/87 - 3/88	110	6.6 E+00	9.4 E-02	9.6 E-02
3/88 - 9/88[c]	82	4.9 E+00	1.4 E-01	1.4 E-01
	Total	7.5 E+01	1.13 E+00	1.16 E+00

a. Total body dose rate is the same for various organs, e.g., gastro-intestinal tract, bone, liver, kidney, thyroid, and lung.

b. Date of Unit 1 commercial operation.

c. Date of Unit 2 commercial operation. The number of construction workers is assumed to be constant between 3/88 and 9/88.

TABLE 12.4.3-2

INDIVIDUAL DOSE

Release Point[a]	Direct Dose (mrem/year)	Total Body Dose[b] (mrem/year)	Skin Dose (mrem/year)
Turbine building vent	-	5.17 E-03	6.28 E-03
Plant vent	-	1.35 E+00	1.43 E+00
Radwaste building vent	-	2.04 E+00	2.04 E+00
Direct	2.6 E-01	-	-
Total	2.6 E-01	3.40 E+00	3.48 E+00

a. The GASPAR code calculates individual doses for various pathways. These doses are totals for the plume, ground, and inhalation pathways.

b. Total body dose is the same for various organs, e.g., gastro-intestinal tract, bone, liver, kidney, thyroid, and lung.

TABLE 12.4.3-3

NEAR FIELD DISPERSION (X/Q) AND DEPOSITION (D/Q) VALUES

Wind Direction (from)	Turbine Building Vent X/Q (s/m^3)	Turbine Building Vent D/Q (m^{-2})[(a)]	Plant Vent X/Q (s/m^3)	Plant Vent D/Q (m^{-2})
N	3.82 E-06	3.82 E-08	-	-
NNE	1.08 E-05	1.08 E-07	4.48 E-06	4.48 E-08
NE	2.06 E-05	2.06 E-07	7.73 E-06	7.73 E-08
ENE	1.65 E-05	1.65 E-07	8.78 E-06	8.78 E-08
E	4.48 E-06	4.48 E-08	1.01 E-05	1.01 E-07
ESE	-	-	-	-
SE	-	-	-	-
Total[(b)]	5.62 E-05	5.62 E-07	3.11 E-05	3.11 E-07

a. D/Q derived by multiplying X/Q by a conservatively assumed deposition velocity of 0.01 m/s.

b. To estimate the worst case X/Q and D/Q, the individual values are totaled. In effect, the worker is assumed to be moving with the wind direction changes or standing at the location of maximum relative concentration for each direction for 24 h/day.

TABLE 12.4.4-1

ESTIMATED OCCUPATIONAL RADIATION EXPOSURE DURING DECOMMISSIONING

Alternative	Man-Rem
A	150
B	130
C	630
A+C[a]	150+310
B+C[a]	130+310

A - Mothballing

B - Entombment

C - Prompt removal/dismantling

C[a] - 102-year delay period before removal/dismantling

a. This information was assembled from reference 1.

TABLE 12.4.4-2

DOMINANT RADIOACTIVE ISOTOPES AT DECOMMISSIONING

	After 2 Years of Decay	After 102 Years of Decay
Vessel and internals	Fe^{55} Co^{60} Ni^{63}	Ni^{63} Ni^{59} Nb^{94}
Other systems	Co^{60}	Sr^{90} Cs^{137}

12.5 HEALTH PHYSICS PROGRAM

12.5.1 ORGANIZATION

12.5.1.1 Program Objectives

The health physics program is organized with the objectives of:

A. Providing administrative control of the activities of plant personnel to ensure that personnel exposure to radiation and radioactive materials is within the guidelines of 10 CFR 20.

B. Providing administrative control of work practices to ensure that the as low as reasonably achievable (ALARA) techniques of paragraph 12.1.3.2 are applied as appropriate.

C. Providing administrative control of effluent releases from the plant to ensure that the releases are maintained ALARA and within the limits of the Offsite Dose Calculation Manual.

D. Providing administrative control of waste shipments from the plant to ensure that all applicable requirements for the shipment and receipt of the material at the burial ground are met.

E. Providing administrative control of records of surveys, radiation monitoring, effluent releases, and waste shipments, to ensure that the requirements of Subpart L of 10 CFR 20 are met.

F. Providing technical support for the radiological training program.

A more detailed discussion of the procedures to accomplish these objectives is presented in subsection 12.5.3.

12.5.1.2 Staff Organization

Responsibility for the operation of the health physics program at the plant is delegated to the health physics manager by the plant manager, who has the overall radiological safety responsibility. Responsibilities of the health physics staff relative to radiation protection are given below.

The plant manager will ensure that all station personnel support the health physics program. The plant manager will participate, via review and approval, in the selection of goals and objectives for the health physics program. Through the plant manager's review and controlling functions, support will be given to the efforts of the health physics manager in achieving the objectives stated in paragraph 12.5.1.1.

The health physics manager has functional control of and is responsible for establishing the health physics program. The health physics manager has the responsibility for ensuring that the ALARA policy is implemented and serves as the radiation protection manager referred to in Nuclear Regulatory Commission Regulatory Guides 8.8 and 8.10. The health physics manager's qualifications are listed in subsection 13.1.3. Additional health physics personnel titles include health physics support supervisor, nuclear specialists, health physics foremen, health physics technicians, and plant health physicist.

12.5.1.3 Program Organization

The applicable portion of the VEGP health physics program was officially initiated when fuel was received for Unit 1 and will be in effect continuously thereafter until the units are decommissioned. This program consists of rules, management and worker philosophies, practices, instructions, and procedures that are used to accomplish the objectives stated above in a practical and safe manner. The program is consistent with the recommendations of Regulatory Guide 8.2.

The health physics program is described in detail in the health physics procedures described in subsection 12.5.3.

The organization and staff will ensure that the radiation surveys required to ensure that the radiological conditions for each job are well defined. The use of the radiation work permit (RWP) will ensure that health physics reviews all jobs for ALARA prior to starting work. The RWP will ensure that the requirements of paragraph 3.a of Regulatory Guide 8.8 are met before work is permitted to start.

For those jobs with significant radiological hazards, a health physics technician may be assigned to provide continuous surveillance for each crew. This continuous coverage will be reserved for significant hazards only, to ensure that exposures to the crew are maintained ALARA. Direct reading dosimeters will be used for all work requiring an RWP. Communications via the plant paging system will be available. These actions will ensure that the requirements of paragraph 3.b of Regulatory Guide 8.8 are met.

Dosage data from completed work will be maintained in plant records for use in planning future work. This will ensure compliance with paragraph 3.c of Regulatory Guide 8.8.

12.5.2 EQUIPMENT, INSTRUMENTATION, AND FACILITIES

12.5.2.1 Facilities

The health physics and radiochemistry facilities are located at el 220 ft in the control building. (See drawing AX1D11A04.) The health physics control point is located at el 220 ft in the northwest corner of the control building.

The health physics control point is located along a principal access corridor. Job planning, radiation work permit (RWP) coordination, and issuance of direct reading dosimeters may be conducted at the control point. Windows at the control point allow observation of the area by health physics personnel to facilitate positive entry control. Portable radiation survey instrumentation, air monitoring survey equipment, self-reading dosimeters, respirators, and other miscellaneous health physics supplies are stored at the health physics control point. The control point is equipped with filing cabinets and desks to provide work space for health physics personnel and storage space for some records.

Anticontamination clothing will be available at local contamination area access points as required. Change facilities will be provided for those areas where and when personnel traffic warrants. Such facilities will consist of change benches, clothing storage, frisker stations, and other health physics support equipment storage, as appropriate for the job in progress.

The equipment and instrumentation described in chapter 12 may not necessarily be the exact instrumentation installed and used at VEGP at the time the plant is operational. State-of-the-art radiation protection equipment will be used if possible.

Health physics sample counting equipment is located in the control building counting room at el 220 ft 0 in. Equipment used for routine counting of smears and air samples such as pancake probe or end window Geiger-Mueller counters, alpha and beta detectors, and/or gasflow proportional counters will be located in this area. Special samples requiring gamma isotopic analysis and/or low level counting will be performed in the counting room.

Personnel decontamination facilities in the west wing of the control building consist of standup shower facilities. The decontamination area may also be used for decontamination of small equipment. Large or highly contaminated equipment may be decontaminated in special areas established by the health physics staff. Decontamination equipment consists of ultrasonic cleaners, sand blasting cabinets, turbulators, stainless steel wash basins, and/or similar equipment. The ventilation systems for the control point are described in subsection 9.4.1. Drainage from the decontamination facilities is held up as liquid radwaste for appropriate treatment.

The radiochemistry and low level laboratories are located in the control building at el 220 ft 0 in., adjacent to the counting room. This area will be used for gamma isotopic analysis and other health physics/radiochemistry special sample counting and analysis.

12.5.2.1.1 Access and Exit of Radiation Controlled Areas

Access to the main radiation controlled area encompassing the auxiliary building, fuel handling building, and containment will normally be through the health physics control point located at the northwest corner of the control building at el 220 ft 0 in. This is normally the only approved ingress to the main radiation controlled area. Exit from the main radiation controlled area will normally be through the same door. Geiger-Mueller tube friskers and portal monitors will be placed at this area for personnel contamination monitoring. Personnel normally pass through a portal monitor prior to leaving the access control area.

12.5.2.2 <u>Health Physics Instrumentation</u>

12.5.2.2.1 Laboratory Instrumentation

Laboratory instrumentation located in the counting room and health physics workroom allows plant personnel to ascertain the radioactive material present in survey samples. Typical samples would be contamination survey smears, airborne survey filters, liquid samples, and charcoal cartridges, but tritium surveys and other samples may be processed also. The laboratory instrumentation is listed in table 12.5.2-1. Each laboratory counting system is checked and calibrated at regular intervals with standard radioactive sources traceable to a National Institute of Standards and Technology (NIST) source, in accordance with plant instructions. Counting efficiency, background count rates, and high voltage settings are checked by plant personnel, in accordance with plant instructions. Instrumentation will also be calibrated after repair, prior to use.

12.5.2.2.2 Whole Body Counting Instrumentation

The whole body counter(s) will be located in an appropriate low background radiation area. The whole body counting facility(s) will be equipped with a high-throughput vertical linear geometry whole body counter or equivalent capable of detecting fractional body burdens of gamma emitting radionuclides. The counter consists of a shadow shield representing a 4-π geometry, multiple detectors, a computer, a multichannel analyzer, data reduction software, and associated electronics. The counter will be capable of detecting radionuclides in the lungs, gastrointestinal tract, and thyroid.

12.5.2.2.3 Portable Survey Instrumentation

Portable survey instrumentation is located at the health physics control point and in-plant control points. This instrumentation will allow plant personnel to perform radiation, contamination, and neutron surveys, as needed, as well as collect samples for airborne analysis.

Each portable survey instrument will be calibrated at least annually, when in use, or after undergoing repair work, prior to use. Calibrations are normally performed using the instrument calibrator (discussed in paragraph 12.5.2.2.5) by plant personnel. Instruments will be source checked to verify proper operation, in accordance with plant instructions. Sufficient quantities of each type of instrument are available to permit calibration, maintenance, and repair, without causing a shortage in operational instrumentation. Portable radiological survey instrumentation is listed in table 12.5.2-2.

12.5.2.2.4 Personnel Monitoring Instruments

Personnel monitoring is provided by use of survey instrumentation (as applicable), air samplers, optically stimulated luminescent dosimeters (OSLDs), and/or direct reading dosimeters.

OSLD badges contain at least two luminescent chips with suitable filters to allow determination of anticipated radiation types.

Direct reading dosimeters will be worn by personnel in the radiologically controlled area as specified by plant instructions. These dosimeters can be used to provide official records of personnel exposure.

If a potential for neutron exposure exists, personnel OSLD badges will be read for exposure to neutron radiation, or a neutron dose rate instrument and personnel stay time will be used to compute neutron exposure in accordance with Regulatory Guide 8.14, section C.1.b.

Since film badges will not be used for personnel monitoring, the performance criterion in Regulatory Guide 8.3 is not applicable.

When audible alarm dosimeters are used as personnel monitoring devices, the guidance for selection and use will generally be in accordance with Regulatory Guide 8.28. Audible alarm dosimeters will be checked electronically before each use. Health physics will determine when dosimeter use is mandatory. If a dosimeter is dropped, proper operation will be verified visually prior to further use.

Extremity dosimetry will be provided as necessary.

The pocket ion chambers used as personnel monitoring equipment will be tested for calibration and leak rate in accordance with requirements of Regulatory Guide 8.4, sections C.1 and C.2.

Personnel survey instrumentation will consist of Geiger-Mueller count rate meters (contamination friskers), portal monitors, and hand and foot counters. These instruments will be calibrated at least annually, as applicable, when in use or after undergoing repair prior to use. Test procedures for Geiger-Mueller counters will be conducted in accordance with those test conditions of Regulatory Guide 8.6 which are specified by the manufacturer. The calibration laboratory is located in the control building at el 240 ft 0 in. Personnel monitoring instrumentation is listed in table 12.5.2-3.

12.5.2.2.5 Health Physics Equipment

Portable air samplers are used to survey airborne radioactive material concentrations. Air samplers are calibrated for flow at least annually. Surveys may be performed for radioactive particulate and radioiodine airborne concentrations.

Portable continuous air monitors are used to monitor airborne concentrations at specific work locations. Local indication and trend information for certain instruments are provided. Alarm setpoints are variable; visual and audible alarms are provided.

Respiratory protective equipment is available at the control point. Self-contained breathing apparatus for emergency use is available at the health physics control point. Equipment will be maintained and used in accordance with 10 CFR 20, Subpart H and Regulatory Guide 8.15.

An instrument calibrator will be used for calibrating gamma dose rate instrumentation. This will be a self-contained, heavily shielded, multiple dose rate calibrator. Neutron, beta, and alpha radiation sources will also be available for source checks and instrument calibration. Calibration sources are traceable to an NIST source.

Protective clothing will be supplied for personnel working in radiologically controlled areas. The clothing required for a specific job will be identified by health physics personnel.

An adequate inventory of protective clothing will be maintained on hand at the access control area or other control points, as necessary to support plant operations and maintenance activities. This clothing will include laboratory coats, coveralls, hoods, caps, plastic oversuits, gloves (plastic, rubber, cloth), shoe covers, boots, and rubbers.

Additional contamination control supplies will be available. These include vacuum cleaners, mops, absorbent paper, plastic sheets and bags, barricades, ropes, signs, and labels. Dedicated and marked disposal containers for discarded and used supplies and clothing are also provided.

A listing of health physics equipment is given in table 12.5.2-4.

12.5.2.2.6 Other Health Physics Instrumentation

The area radiation monitoring system is installed in areas where it is desirable to have constant dose rate information.

Monitors indicate dose rate in the control room and provide appropriate alarms upon reaching a preset dose rate. Fixed continuous airborne radioactivity monitors are also provided at strategic locations, where personnel exposure to airborne radionuclides is likely. More information on these fixed instruments is given in sections 12.3 and 11.5.

A portable monitor system, as required in NUREG-0737, item III.D.3.3, is available. The system uses an iodine silver zeolite or charcoal sample cartridge and a single-channel analyzer. The use of this portable monitor has been incorporated in the emergency procedures. The portable monitor is part of the in-plant radiation monitoring program (paragraph 12.4.1.2.5) to accurately determine the airborne iodine concentration in areas where plant personnel may be present during an accident.

Sampling cartridges could be removed to a low background area (the counting room in the control building) for further analysis. These cartridge samples can be purged of any entrapped noble gases, when necessary, prior to being analyzed.

12.5.3 PROCEDURES

Procedures will be developed to cover all necessary areas of plant operations and maintenance. Section 13.5 provides information on the procedures to be developed. The procedures describe certain methods that will be used to ensure that occupational radiation exposures will be as low as reasonably achievable (ALARA). The Company's commitment to regulatory guides will be incorporated into procedures as appropriate.

Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR Part 20 and shall be approved, maintained, and adhered to for all operations involving personnel radiation exposure. Strict adherence to the VEGP radiation protection procedures will ensure that personnel radiation exposures are both ALARA and within the limits of 10 CFR 20. Policy and operational considerations for radiation protection are set forth in subsections 12.1.1 and 12.1.3. Radiation protection procedures will be subject to the requirements of the quality assurance program for the operational phase of plant life described in the SNC Quality Assurance Topical Report (QATR).

12.5.3.1 Radiation and Contamination Surveys

Health physics personnel normally perform routine radiation and contamination surveys, the techniques of which are delineated in plant procedures. Surveys are performed on a frequency that varies with the potential radiological hazards associated with the given area. Survey frequency will be specified in plant procedures. These surveys consist of radiation measurements and/or contamination surveys, as appropriate for the specific area. Air samples are also routinely taken in portions of the controlled area. Survey information is factored into exposure stay time determinations and radiation work permit (RWP) specifications. (See paragraph 12.5.3.7.) An RWP may specify the need for additional surveys for specific operations and/or maintenance activities. Radiation level surveys will be performed for alpha, gamma, beta, and/or neutron exposure rates, as specified in plant procedures. Contamination surveys are normally performed to establish gross beta-gamma contamination level but may be processed for specific radionuclides (via gamma spectroscopy). Air samples are normally taken to establish airborne concentrations of particulates, noble gases, and/or radioiodine. Availability of current survey information will aid in keeping exposures ALARA.

12.5.3.2 Methods to Maintain Exposures ALARA

Methods to maintain exposures ALARA in accordance with Regulatory Guides 8.8 and 8.10 are not only included in radiation protection procedures but are also contained in applicable operating and maintenance procedures. Some examples of the types of methods that will be used to maintain exposures ALARA are discussed below for the following operational categories.

12.5.3.2.1 Refueling

After the reactor coolant system is depressurized, it is degassed and sampled to verify that the gaseous activity is low, prior to removing the reactor head. After flooding the refueling pool above the reactor, purification of the refueling pool water is continued to maintain exposures from activity in the water ALARA. Movement of irradiated fuel assemblies will be accomplished with the assembly maintained underwater. By following these procedures, the normal radiation level on the refueling bridge is expected to be less than 5 mrem/h. The RWP system is used to maintain positive radiological control over work in progress.

12.5.3.2.2 Inservice Inspection

Prior to entry into radiation areas to perform inspections, personnel may study, as appropriate: blueprints, drawings, photographs, videotapes, previous inspection reports, previous radiation and contamination surveys, or previous RWPs appropriate to the particular job. This will acquaint personnel with the job location, the work to be done, and radiation and contamination levels previously experienced. Surveys are performed to the extent required to determine present contamination and/or radiation levels. From this data, previous data, and past experience of personnel, an RWP (paragraph 12.5.3.7) is issued. Equipment is checked and/or calibrated to verify it is operating properly prior to entry into the radiation area. Temporary shielding will be used, where practicable, to reduce radiation exposure.

12.5.3.2.3 Radwaste Handling

The handling of radwaste has been minimized by plant design. The radwaste system is shielded and incorporates remotely operated liquid radwaste systems. The systems are designed to minimize operator exposure in all waste processing and handling operations. The radwaste system is described in chapter 11.

12.5.3.2.4 Spent Fuel Handling and Spent Fuel Cask Loading

Spent fuel handling and spent fuel cask loading is performed underwater, using the fuel handling cranes and/or manual extension tools. This normally requires a small crew working in the fuel handling area and usually involves little exposure. The RWP system will be used to maintain positive radiological control over this task.

Some of the methods used to maintain exposure ALARA are:

A. Maintain at least 8 ft of water above the fuel assembly to minimize direct radiation.

B. Purify fuel pool water to minimize exposure due to water activity.

C. Cool the spent fuel pool water.

D. Provide continuous air sampling while moving fuel to evaluate airborne activity.

E. Have emergency procedures immediately available.

After the cask is loaded, it is decontaminated using pressurized water to minimize loose contamination and the amount of hand cleaning of the cask.

12.5.3.2.5 Normal Operation

The plant was designed so that significant radiation sources are minimized, shielded, and/or placed in cubicles. Much of the instrumentation required for normal operation reads out remotely in the control room or in other low radiation areas. Instrumentation that cannot be placed remotely or that is read infrequently is situated, where possible, so that it can be read from the entrance to the cubicle or from a low radiation area within the cubicle. Operators are instructed to stay outside cubicles in which radiation levels are high as much as possible, and they are appraised of the areas inside cubicles where the radiation level is usually the lowest. If an operator plans to enter a high radiation area cubicle, he notifies health physics and specifies the cubicle. Upon leaving the health physics control point, he enters exposure data and time spent in the area on the RWP.

Area radiation monitoring equipment, part of the process effluent radiological monitoring system, is available and provides indication of radiation levels and local alarms, except for containment area radiation monitors which do not have local alarms and indicators. The ventilation system is designed to minimize spread of airborne contamination. The RWP system is used to maintain positive radiological control over radiation work. (See paragraph 12.5.3.7.)

12.5.3.2.6 Routine Maintenance

Routine maintenance falls into the categories of preventive maintenance (planned and scheduled maintenance such as lubrication, adjustments, and tests) and corrective maintenance (unscheduled maintenance such as valve packing, pump seal replacement, and stopping leaks). Procedures are written for the usual preventive maintenance jobs and for some recurring corrective maintenance jobs. These procedures specify the precautions to be taken. The procedures list the required lubricants, special tools and equipment, and the acceptance standards. This serves to minimize the time spent in the radiation area.

In addition, the usual preventive maintenance procedure normally states whether an RWP is required. When the RWP is issued, the radiation and/or contamination levels are listed, shielding is specified if appropriate, and additional specific instructions are given to personnel. For corrective maintenance jobs in radiation areas, for which a general procedure is used, a similar approach is used.

Extension tools are used when practical. Detailed surveys are performed and the RWP is issued (if required) with specific instructions. The individuals performing the work may be required to read procedure manuals or may be shown pictures or sketches to aid in understanding what is to be accomplished, how it is to be accomplished as safely and quickly as possible, and what the acceptance criteria are. Additional requirements may be imposed to reduce exposures at the discretion of health physics personnel.

After the job is completed, debriefings may be utilized to obtain input from personnel actually performing the work in addition to supervisory and support personnel. This will assist in revising procedures for ALARA considerations.

12.5.3.2.7 Sampling

Most sampling of radioactive systems is performed inside the hoods in the sampling station, which protect personnel from airborne activity. Protective clothing and gloves are required when sampling radioactive systems to prevent contamination of personnel.

A survey instrument may be used to check radiation levels. The liquid sample container is normally washed with clean water and dried before being brought into the radiochemistry laboratory for analysis. The dose from sample bottles is minimized by grasping the bottle at the top, by using tongs, or by using a sample carrier.

12.5.3.2.8 Calibration

Calibration of most ranges of the portable gamma detection instruments is performed inside a shielded calibrator, thus almost eliminating exposure from calibration of portable instruments. Portable sources used to calibrate fixed instruments (such as the area radiation monitoring system) are transported in shielded containers.

Where possible, fixed instruments requiring routine calibration are situated so that the necessary test signals can be inserted from a low radiation area with the instruments in place.

12.5.3.2.9 Housekeeping

Plant procedures specify that the plant be maintained in a clean condition to avoid, to the extent practicable, conditions which lead to possible contamination. Plant personnel will be made aware of the importance of a clean plant to radiation protection. Housekeeping requirements are performed in accordance with NQA-1-1994, as described in the SNC QATR.

12.5.3.3 Controlling Access and Stay Time

The area within the primary security boundary is broken down into nonradiation (clean) areas and radiation controlled areas (for radiation protection purposes). Radiation controlled areas are further categorized as radiation areas, high radiation areas, very high radiation areas, airborne radioactivity areas, contamination areas, and radioactive materials areas, to comply with 10 CFR 20 and plant procedures and instructions.

Personnel entering the radiation controlled areas of the plant must be trained in radiation protection and emergency procedures as specified in paragraph 12.5.3.8 or must be escorted. Entrance to the radiation controlled area is normally through the access control area (previously discussed in subsection 12.5.2) where an RWP will be issued, if required, for the areas to be entered. (See paragraph 12.5.3.7.)

Radiation, high radiation, and very high radiation areas are segregated and identified in accordance with 10 CFR 20. The entrance to high radiation areas is controlled in accordance with Technical Specifications, section 5.7. Measures taken to control access to very high radiation areas will meet the intent of Regulatory Guide 8.38. Control over entries into radiation controlled areas is provided by using RWPs. (See paragraph 12.5.3.7.)

During major outages, such as refueling, the RWP is prepared to designate which individual job tasks and associated doses are to be included under specific RWPs. Thus, exposure information relating to specific jobs can be accumulated for preplanning sessions in the future. This technique is discussed in subsection 12.1.3.

12.5.3.4 Contamination Control

Contamination limits for personnel, equipment, and areas are listed in the plant procedures. Surveys are performed routinely, as discussed in paragraph 12.5.3.1, to determine contamination levels. Additional surveys may be performed after maintenance work or after an operation that may have increased contamination levels. Any area found contaminated is roped off or otherwise delineated with a physical barrier, posted with appropriate signs, and decontaminated as soon as practicable. In areas where the radiation level is high or where it is considered impractical to decontaminate the area to general radiation controlled area limits, a stepoff pad is used to prevent the spread of contamination.

Tools and equipment used in contaminated areas are monitored and/or bagged (or wrapped in polyethylene sheeting) prior to being removed from the work area to prevent the spread of contamination. All tools and equipment being removed from the radiation controlled area are surveyed for contamination by health physics personnel (or other qualified personnel as specified in plant radiation protection procedures) to ensure that they meet clean area limits. If the tools or equipment do not meet the limits, they are decontaminated to the extent practicable. Decontamination facilities are discussed in subsection 12.5.2.

Control of personnel contamination (external and internal) may be provided by using protective clothing and respiratory equipment. Each individual who exits a contaminated or potentially contaminated area is responsible for surveying himself when he crosses a local control point equipped with either friskers or personnel contamination monitors. Personnel contamination monitors will be used at the control point. If contamination is found, the individual is decontaminated, using the facilities described in subsection 12.5.2, under the direction of health physics or other qualified personnel.

Special coatings are applied to walls and floors of areas containing radioactive fluids, which will aid in decontaminating these areas if it should become necessary.

12.5.3.5 Respiratory Protection

If personnel entry is required into areas where the sources of airborne radioactivity cannot be removed or controlled, occupancy may be restricted and/or respiratory protection equipment may be provided to maintain the total effective dose equivalent ALARA. When airborne radioactivity is detected in excess of the limits in 10 CFR 20, the area is isolated and posted as an airborne radioactivity area and access is controlled. Entry into these areas requires the issuance of an RWP. The RWP system (paragraph 12.5.3.7) provides radiation exposure control by controlling and recording conditions under which work in airborne radioactivity areas is performed. Air sampling techniques are used to evaluate the need for respiratory protective equipment. If needed, selection of the appropriate type of respiratory equipment is then determined. The respiratory protection program is organized to conform to 10 CFR 20, Subpart H and Regulatory Guide 8.15.

The major portion of the respiratory equipment is available in the access control area of the control building. Supplementary emergency respiratory equipment is available in the control room and emergency kits. The respiratory equipment that is available includes full face masks and self-contained breathing equipment. Table 12.5.2-4 lists respiratory equipment.

12.5.3.6 Personnel Dosimetry

All personnel entering a radiation controlled area are required to wear, as a minimum, a direct reading dosimeter. Personnel entering containment are required to wear, as a minimum, a direct reading dosimeter capable of alarming excessive dose and dose rate. Setpoints for dose and dose rate are determined based on the work to be performed. Visitors are restricted from access into high radiation areas and very high radiation areas. Temporary radiation workers and visitors are escorted by a radiation worker and are limited to the requirements of 10 CFR 20.2104 and 10 CFR 20.1502.

Only those individuals who have completed training in radiation protection and emergency procedures will be authorized to enter radiation controlled areas unescorted. When visitors and other persons who have not completed this training enter a radiation controlled area, they will obtain an escort trained in these procedures prior to entering the area.

Direct reading dosimeters will be used to provide the daily measurement of personnel exposure. When OSLDs are required, they will be processed on a routine basis by a dosimetry processor holding a current personnel dosimetry accreditation from the National Voluntary Laboratory Accreditation Program (NVLAP). OSLD badges may require special processing during outages and refueling or when an individual's exposure status is in doubt. Individual dosimeter and OSLD records will be maintained by Health Physics personnel and document control.

Personnel dosimetry record keeping will be in accordance with 10 CFR 20, Subpart L and will meet the intent of Regulatory Guide 8.7.

Whole body counts and/or bioassays will be performed on an individual basis as necessary. These special individual measurements will be initiated when the results of monitoring in the workplace indicate that significant intakes may have occurred or when workers have been associated with known incidents possibly involving significant intakes of radioactivity. Air monitoring and surface contamination tests in the workplace and tests of skin contamination, nose blows, and nasal smears will be used for determining whether special measurements will be required. Investigation levels are established in plant procedures.

The development of plant procedures for bioassay will include the development of detailed criteria for the performance of bioassay and methods of data analysis and interpretation. The criteria and methods used for developing and implementing the bioassay program will be based on the recommendations of Regulatory Guides 8.20, and 8.26, and will meet the intent of Regulatory Guide 8.9.

Control of exposure to declared pregnant female workers will be in accordance with 10 CFR 20.1208. The success of such a limitation of exposure will largely depend upon the female worker's understanding the reasons for such limitations. See paragraph 12.5.3.8 for further discussion. At the time of declaration, their exposure will be limited to 0.5 rem for the duration of their pregnancy.

Exposure data of all personnel will be collected and recorded on form NRC-5, Occupational Exposure Record for a Monitoring Period, or the equivalent. Occupational exposures incurred by individuals prior to working at VEGP will meet the requirements of 10 CFR 20.2104 for determination of prior exposure. Prior dose information may be summarized on form NRC-4, accumulative occupational exposure history, or the equivalent. These records will be maintained at the plant and will be preserved indefinitely or until the Nuclear Regulatory Commission authorizes their disposal pursuant to Subpart L of 10 CFR 20.

Results of unusual or excessive exposures will be reported to the Nuclear Regulatory Commission as required by 10 CFR 20 Subpart M.

12.5.3.7 Radiation Work Permits

The RWP is a document issued by health physics personnel to ensure proper control of work performed in radiologically controlled areas.

The RWPs will be issued by health physics personnel prior to allowing work to be performed in radiologically controlled areas. The health physics foreman or a designated alternate has direct management responsibility for the issuance of RWPs. No RWPs will be issued without approval from the health physics foreman or a designated alternate.

The RWPs identify personnel involved in the job and state the job to be performed, allowable stay times (if appropriate), protective clothing and equipment required, monitoring requirements, and any special instructions or cautions pertinent to radiation hazards. The permit lists the radiological hazards present in the work area, including area dose rate and the presence of hot spots, airborne contamination, loose surface contamination, and other hazards as appropriate. These permits provide the administrative control to ensure that all work is performed in a radiologically safe manner.

Most RWPs will be issued for a specific job. Typically the RWP will be valid only for the duration of the job or some shorter interval specified by health physics personnel. General RWPs will be issued to cover the performance of routine surveillance work in radiation controlled areas. General RWPs will not be issued for work in high radiation areas, very high radiation areas, areas with significant contamination potential, or areas requiring a survey by health physics personnel prior to entry.

Dosimeter readings and entry and exit times will be recorded for each worker on the RWP.

12.5.3.8 Radiation Protection Training

Each member of the permanent operating organization, whose duties entail entering radiologically controlled areas or directing the activities of others who enter radiation controlled areas, will be instructed in the fundamentals of radiation safety. A description of the radiation safety course is given in section 13.2. All permanent plant personnel will be required to attend a retraining program in radiation protection at the required frequency determined per the Systematic Approach to Training process in accordance with VEGP's training programs accredited by the National Academy for Nuclear Training. Personnel whose duties do not require entry into radiation controlled areas will be made aware of the reasons for keeping out of such areas. The training program will be developed based on the recommendations of Regulatory Guides 8.27 and 8.29, except that the training frequency will be determined per the Systematic Approach to Training process in accordance with VEGP's training programs accredited by the National Academy for Nuclear Training. Instructions on risks associated with occupational radiation exposure will be implemented in accordance with the recommendations of Regulatory Guide 8.8.

Considerable time and effort will be devoted to ensure that employees understand radiation and radiation safety as it applies to their work. Supervisors are responsible for ensuring that their employees follow proper radiation protection procedures and instructions. The amount and type of training will depend on the radiological hazards associated with the work they perform. Orientation lectures on radiation and radiation protection will be given to all new employees. Training will continue with detailed discussions of the specific radiological hazards associated with work assignments. In the course of their work, employees will receive additional training in radiation protection practices from supervisors, senior coworkers, and health physics personnel.

TABLE 12.5.2-1

FIXED LABORATORY INSTRUMENTATION

Instrument	Quantity	Radiation Detected	Accuracy	Range	Remarks
Counting laboratory					
Gamma spectrometer	2	Gamma	16% relative to 3 in. x 3 in. NaI	0-2 MeV	Computer based pulse height analyzer system with Germanium detector
Geiger-Mueller counter scaler	1	Beta, gamma	10% Co-60	0-10 counts/min	To be used for counting air samples and smears
Scintillation counter scaler	1	Beta, alpha	Determined during calibration	0-10 counts/min	To be used for counting air samples and smears

TABLE 12.5.2-2

PORTABLE HEALTH PHYSICS SURVEY INSTRUMENTS

Instrument	Quantity	Radiation Detected	Accuracy	Range	Remarks
Ion chamber survey meter	15	Beta, gamma	±10% of full scale	0-50 R/h	"RO-2A" type dose rate instrument
Geiger-Mueller survey meter	12	Beta, gamma	±10% of full scale	0.1 -1000 R/h	Extended probe type; dose rate instrument
Geiger-Mueller survey meter	15	Beta, gamma	±10% of full scale	0-50,000 counts/min	Count rate instrument
Geiger-Mueller survey meter	15	Beta, gamma	±10% of full scale	0-50 mR/h	Survey instrument
Geiger-Mueller survey meter (frisker)	40	Beta, gamma	±10% of full scale	0-50,000 counts/min	Count rate instrument with adjustment alarm for personnel surveys
Neutron survey meter	2	Neutrons	±15% directional response +8% linearity	0-5000 mR/h	BF tube within cadmium loaded polyethylene sphere; neutron dose rate instrument (rem counter)
Alpha survey meter	2	Alpha particles	±10% of full scale	0-500,000 counts/min	Alpha scintillation(ZnS) or gas proportional count rate instrument
Portable high volume air sampler	10	Not applicable	Not applicable	0-5 ft$^{(3)}$/min flowrate	Used for grab samples in work areas
Ion chamber survey meter	2	Beta, gamma	±20% of reading	0-10,000 R/h	Ion chamber dose rate instrument

TABLE 12.5.2-3

PERSONNEL MONITORING EQUIPMENT

Instrument	Quantity	Radiation Detected	Accuracy	Range	Remarks
Personnel dosimeters	130	Gamma	±10%	0-9999 mR	Electronic direct reading
Portal monitor	3	Beta, gamma		0-1000 counts/min	Multiple detector gas flow proportional
Hand and foot counter	1	Beta, gamma		0-10 counts/min	4 channels
Portal monitor	3	Gamma		0-1000 counts/min	Multiple detector gamma scintillation

TABLE 12.5.2-4

HEALTH PHYSICS EQUIPMENT

Instrument	Quantity	Radiation Detected	Accuracy	Range	Remarks
Continuous air monitor	1	Beta, gamma	±10% of reading	0-50,000 counts/min	Adjustable alarm
Instrument calibrator	1	Gamma	-	0.002-500 R/h	Shielded self-contained calibrator
Air purifying respirator	40	-	-	-	Full face – negative pressure
Self-contained breathing apparatus	5	-	-	-	Supplements emergency equipment
Atmosphere supplying respirators	15	-	-	-	1- or 2-piece, ventilated hood, constant airflow

13.0 CONDUCT OF OPERATIONS

13.1 ORGANIZATIONAL STRUCTURE OF APPLICANT

13.1.1 MANAGEMENT AND TECHNICAL SUPPORT ORGANIZATION

This section provides information concerning corporate organization, functions, and responsibilities; participation in the facility design; design review; design approval; construction management; testing; and operation of the plant. The corporate nuclear operations organization and the Southern Nuclear Operating Company (SNC) plant organization are responsible for directing activities at VEGP. The organizations described in chapter 13 support and report to nuclear operations for assigned activities.

13.1.1.1 Design and Operating Responsibilities

The following paragraphs summarize the degree to which design, construction, and preoperational activities were accomplished and describe the specific responsibilities and activities relative to technical support for operations.

13.1.1.1.1 Design and Construction Activities

13.1.1.1.1.1 Principal Site-Related Engineering Work. Principal site-related work (such as meteorology, geology, seismology, hydrology, and demography) is described in chapter 2. The VEGP preoperational monitoring program is described in the environmental report; this program established a preoperational baseline from which to evaluate future monitoring of environmental effects.

13.1.1.1.1.2 Design of Plant and Auxiliary Systems. Units 1 and 2 engineering and construction are complete.

13.1.1.1.1.3 Site Layout with Respect to Environmental Effects and Security Provisions. Site layout with respect to environmental effects is described in chapter 2. Site security with respect to plant geographical layout and equipment is described in the security plan.

13.1.1.1.1.4 Development of Safety Analysis Reports. Overall coordination of the preparation of the original Final Safety Analysis Report (FSAR) and its annual updates was assigned to Southern Company Services (SCS) Nuclear Support and Quality Assurance. Preparation of the individual sections was assigned to the cognizant technical groups within Bechtel, Westinghouse, SCS, and Georgia Power Company (GPC). As of March 22, 1997, SNC, as the exclusive operating licensee, reviews and approves the FSAR and annual updates and maintains overall responsibility for the FSAR. As a result of the consolidation of SCS and

SNC nuclear expertise, and in addition to being the licensee, SNC also serves as its own architect/engineer and performs the functions previously performed by SCS.

13.1.1.1.1.5 Review and Approval of Material and Component Specifications. Project specifications for safety-related equipment are reviewed in accordance with the quality assurance program as described in chapter 17.

13.1.1.1.1.6 Procurement of Materials and Equipment. Procurement of Unit 2 materials and major engineered equipment is complete.

13.1.1.1.2 Preoperational Activities

13.1.1.1.2.1 Development of Human Engineering Design Objectives and Design Phase Review of Proposed Control Room Layouts. The VEGP control room was designed using a reduced-size control board and GPC operator input into the control board control configurations. The design incorporated the human factor design criteria available at that time. An independent evaluation on human factor design has been performed on a mockup of the control room. A detailed discussion of control room design review and human engineering factors is described in chapter 18.

Design recommendations based on the mockup review were incorporated in the control board design or deferred for further evaluation. A detailed design review was conducted on the as-built control room and used the plant simulator.

13.1.1.1.2.2 Development and Implementation of Staff Recruiting and Training Programs. The operating staff is described in subsection 13.1.2. Recruiting of personnel to fill these positions started in 1977. The initial plant staffing is complete. Training programs have been developed for this facility and are described in section 13.2.

13.1.1.1.2.3 Development of Plans for Initial Testing. The nuclear plant general manager (NPGM) was responsible for all aspects of the initial test program of the VEGP. As part of his responsibilities, the NPGM directed the development of the startup manual.

The startup manual defined the startup organization, defined the responsibilities of involved organizations and personnel, delineated the qualifications necessary for startup personnel, and contained the administrative controls necessary for the implementation of the initial test program.

The administrative controls, qualification for testing personnel, and other required procedures for conducting that part of the initial test program after fuel load was included in the plant procedure manual or startup test procedures manual.

13.1.1.1.2.4 Development of Plant Maintenance Programs. The work force assigned to the VEGP provided qualified maintenance personnel prior to initial fuel loading.

Structures, systems, and components that prevent or mitigate the consequences of postulated accidents that could cause undue risk to the health and safety of the public are maintained in accordance with the quality assurance program.

The maintenance staff is sized to perform the routine and preventive maintenance workload. The staff is supplemented by outside contractors as deemed appropriate by plant management. Maintenance is performed under the direction of cognizant team leaders or assistant team leaders and in accordance with accepted work practices.

The scope and frequency of the preventive maintenance considers such factors as past experience with similar equipment, the manufacturer's recommendations, and importance of the equipment to plant operations or nuclear safety. Records are kept to establish the maintenance history of major safety-related equipment. Maintenance and repairs are performed by qualified personnel in accordance with written work orders, maintenance procedures, standing orders, vendor technical manuals, and/or applicable codes and regulations. Qualified maintenance personnel possess the skills to perform work without detailed written procedures. Except for emergencies, maintenance work is preplanned. Training meetings are held to foster safety awareness and quality of workmanship.

13.1.1.1.3 Technical Support for Operations

SNC is responsible to GPC for assuring the availability of, for providing, or for securing adequate technical support for the VEGP. SNC may use the services of SNC, Bechtel, Westinghouse, and others as appropriate. Portions of the technical support from these organizations may be fulfilled through the use of outside contractors.

13.1.1.1.4 SNC Technical Support

The SNC technical support organization includes all major design disciplines and contracts for outside specialty technical support when additional expertise is needed and when major surges in manpower needs occur. SNC and others provide for various assigned support activities, which normally include the following:

A. Architect-engineering services as required for the design-engineering of plant modifications, including maintenance-related design changes, plant improvement-related design changes, and design changes or major plant additions as a result of new regulatory requirements and commitments. These services include both conceptual and detailed design, issue and maintenance of design drawings and specifications, incorporation of as-built notices, procurement, related quality assurance functions, etc.

B. Design-related evaluation and analysis.

C. Site assistance as required on 10 CFR 50.59 evaluations and analyses which are not directly related to design; e.g., with respect to operational requirements, technical specification changes, etc.

D. Engineering and maintenance support to ensure reliable operation of components.

E. Licensing-related support for NRC interface and review of generic issues and plant changes.

F. Nuclear fuel procurement and contract management.

G. Nuclear fuel support, including core analysis.

H. Selected generic issues and projects.

I. Plant chemistry support.

13.1.1.2 Organizational Arrangement

13.1.1.2.1 Nuclear Operations Organization

The nuclear operations organization, under the supervision of the President/CEO, has direct responsibility for the operation and maintenance of Southern Company's nuclear plants. The nuclear operations organization consists of the plant operating staffs and corporate management, planning and performance and quality assurance. Engineering support is provided primarily by the corporate and site engineering organizations as described herein.

The structure of the nuclear operations organization is described in the SNC Quality Assurance Topical Report (QATR). Portions of the SNC Fleet Operations Support, Engineering, General Counsel and External Affairs, and Human Resources organizations are also described in the following paragraphs or in the QATR.

13.1.1.2.1.1 Vice President and General Counsel. The vice president and general counsel reports to the president/CEO. This individual is responsible for the legal, compliance, and external affairs associated with operation of SNC plants. This individual is also responsible for external affairs activities which include governmental affairs and corporate communications. The vice president and general counsel is also the corporate secretary and directs the managing attorney/compliance manager and the public affairs manager.

13.1.1.3 Qualifications

Together, SNC and Georgia Power Company operate electric generating plants with an aggregate capacity in excess of 14,000 MWe. SNC personnel have experience in the designing, constructing, startup testing, operating, and staffing of nuclear generating facilities, including HNP, a nuclear power plant with two boiling water reactors and Joseph M. Farley Nuclear Plant, a nuclear plant with two pressurized water reactors.

The corporate organization is described in the QATR. The ultimate responsibility for design, procurement, construction, testing, quality assurance, and operation of the VEGP rests with the president and CEO, who assigns responsibilities to the various organizations as described in paragraph 13.1.1.2.

Members of corporate headquarters staff available for the technical support of the VEGP possess that combination of education, experience, and skills commensurate with their level of responsibility. They provide reasonable assurance that support services provided in connection with the operation of the VEGP units will not constitute a hazard to the health and safety of the public.

The operating organization for the VEGP is described in the QATR except for those positions described below.

13.1.2 OPERATING ORGANIZATION

13.1.2.1 Plant Organization

The VEGP consists of two nearly identical nuclear generating units. The applicable plant organization is described in the QATR.

13.1.2.2 Plant Personnel Responsibilities and Authorities

The plant personnel responsibilities and authorities are described in the QATR except for those positions described below.

13.1.2.2.1 Operations Supervision and Shift Organization

The Operations Director is responsible to plant management for the operation of Unit 1 and Unit 2. The Operations Director or at least one operations department manager shall possess a Senior Reactor Operator (SRO) license. Reporting to the Operations Director includes: the Shift Operations Manager, Operation Support Manager, and the Operations Services Manager. The Shift Operations Manager provides oversight for licensed activities, training, staffing, procedures, and policies affecting “on-shift” Operations personnel. The Operations Services Manager directs day-to-day planning, provides technical support, prepares for, conducts and oversees outage related activities. The Operations Support Manager provides oversight for staff support functions and coordinates training of operations personnel.

The Shift Manager is responsible for seeing that plant operations are conducted in accordance with appropriate standing orders, unit operating procedures, and technical specifications. The Shift Manager's principal responsibility is ensuring safe operation during his assigned shift as addressed in the requirements of item I.A.1.2 of NUREG-0737. The Shift Manager shall possess an SRO license.

The Shift Technical Advisor position meets the intent of NUREG-0660, as clarified by NUREG-0737, item I.A.1.1. The Shift Technical Advisor position may be eliminated if the qualifications of the Shift Manager, Shift Supervisor, or SRO-licensed Shift Support Supervisor meet the requirements of the Shift Technical Advisor position. Section 13.2 describes Shift Technical Advisor training, and subsection 13.1.3 describes Shift Technical Advisor qualifications.

A Shift Supervisor for each unit is under the supervision of the Shift Manager. Each Shift Supervisor shall possess a SRO license. Each Shift Supervisor is responsible for the safe and efficient operation of the unit. The Shift Supervisor for each unit keeps a record of shift activities and establishes unit load as requested by the System Operator or as emergency conditions dictate.

Also reporting to the Shift Supervisors are the Shift Support Supervisors, Plant Operators, and Systems Operators.

Plant Operators monitor the plant status and operate equipment as needed to maintain control of the various plant processes. Most of their duties are located in the control room; although, they may perform inspections in other areas of the plant. The operating crew may make radiation and contamination surveys within the plant. (In addition to the control room personnel, a chemistry technician is on duty during plant operations.) The Technical Specifications state the shift manning requirements for all modes of operation.

The succession to responsibility for overall plant operation and the authority to issue operating instructions or special orders, in the event of absences, incapacitation of personnel, or other emergencies, are as follows:

A. Plant Manager (PM).

B. Operations Director.

C. Regulatory Affairs Manager.

D. Licensed manager/superintendent designated by PM.

E. Shift Manager (SRO).

F. Shift Supervisor (SRO).

13.1.2.3 Operating Shift Crews

A total manpower in excess of five full shift crews is provided to avoid excessive overtime.

During refueling operations, when the reactor core configuration is being altered, a senior reactor operator will supervise the fuel handling activities in the reactor containment.

Licensed operating personnel are trained in the radiation protection procedures and are capable of performing routine or special radiation surveys using portable radiation detectors. These operators are trained in the use of protective barriers and signs, protective clothing and breathing apparatus, in the performance of contamination surveys, and in performance of checks on radiation monitors. Plant personnel will be trained in the limits of exposure rates and accumulated dose. The shift manager is responsible for implementing the radiation protection program in the absence of the health physics manager or his designated alternate.

13.1.3 QUALIFICATIONS OF PERSONNEL

13.1.3.1 Qualification Requirements

Each member of the unit staff shall meet or exceed the minimum education and experience recommendations of Regulatory Guide 1.8, Revision 2, and for licensed staff, 10 CFR 55.59. Prior to meeting the recommendations of Regulatory Guide 1.8, Revision 2, personnel may be trained to perform specific tasks and will be qualified to perform those tasks independently. Personnel who complete an accredited program which is endorsed by the Nuclear Regulatory Commission (NRC) shall meet the requirements of the accredited program in lieu of the above.

Table 13.1.3-1 shows the education and experience requirements for VEGP organizational titles with equivalent titles taken from ANSI 18.1-1971 and ANSI 3.1-1981 for those positions, as endorsed by Regulatory Guide 1.8, Revision 2, for key plant personnel. Personnel in job positions covered by accredited training programs meet the requirements of those accredited programs as described in section 13.2.

13.1.3.2 ANSI 18.1-1971 Definitions

Nuclear power plant - A nuclear power plant is any plant using a nuclear reactor to produce electric power, process steam, or space heating.

Experience - Experience shall mean actual applicable working experience in design, construction, startup, operation, maintenance, or technical services. Observation of others performing applicable functions shall not be considered acceptable experience.

Academic training - Academic training is successfully completed college-level work leading to a recognized degree.

Related technical training - Related technical training is formal training beyond the high school level in technical subjects associated with the position in question, acquired in training schools or programs conducted by the military, industry, utilities, universities, vocational schools, or others. Such training programs should be of a scheduled and planned length and include text material, lectures, and frequent examinations.

On-the-job training - On-the-job training is participation in nuclear power plant startup, operation, maintenance, or technical services under the direction of appropriately experienced personnel.

13.1.3.3 <u>Qualification of Plant Personnel Using ANSI 18.1-1971</u>

Nuclear power plant experience is nuclear experience acquired in the design, construction, startup, or operation of nuclear power plants. Further, nuclear experience acquired at military, nonstationary, or propulsion nuclear plants may qualify as equivalent experience on a one-for-one time basis.

Nuclear experience acquired in nonpower plants (such as test, research, or production reactors) may qualify as equivalent to nuclear power plant experience on a one-for-one time basis. Only 1 year of such experience will be accredited toward the total nuclear experience qualification.

Training may qualify as experience if acquired in appropriate reactor simulator training programs, on the basis of 1 month's training being equivalent to 3 month's experience. Only one year of such experience will be accredited toward the total nuclear experience qualification.

Training programs, the culmination of which involves actual reactor operation, may qualify as equivalent to nuclear power plant experience on a one-for-one time basis for up to a maximum of 1 year credit.

On-the-job training may qualify as equivalent to nuclear power plant experience on a one-for-one time basis for up to a maximum of 1 year credit toward the nuclear power plant experience requirement.

The qualification of the staff personnel holding key managerial and supervisory positions as described in paragraph 13.1.2.2.1 are provided in table 13.1.3-1.

For qualification of personnel for positions that require a senior reactor operator or reactor operator license, the shift technical advisor and the radiation protection manager will follow the recommendations of ANSI 3.1-1981 as modified by Regulatory Guide 1.8, Revision 2.

TABLE 13.1.3-1 (SHEET 1 OF 3)

ACCEPTABLE EDUCATION AND EXPERIENCE FOR SELECTION AND TRAINING OF VEGP PERSONNEL

		Recommended Experience (years)				License		Suggested Education (years)			
ANSI N18.1 Title	VEGP Title	Total Power Plant	Nuclear Power Plant	Other Applicable	Academic Training	Reactor Operator	Senior Reactor Operator	Academic	Related Technical Training	Amount of Education Creditable for Experience	Remarks
Managers											
Plant manager	Vice President-Vogtle or Plant Manager [(a)]	10	3				X[(a)]	4		4	
Assistant plant manager	Regulatory Affairs managers	10	3				X[(a)]	4		4	
Operation manager	Operations director	8	3				X[(d)]	2 or	2	2	
Technical manager	Engineering director[(e)]/Design/ Engineering support managers/Technical Services manager		1	7				4		4	
Maintenance manager	Maintenance director	7	1					2 or	2	2	
Training manager (b)	Training manager		2	4				4			
Superintendents or supervisors not requiring NRC license											
Radiochemistry	Chemistry manager or plant chemist			5				2 or	2	4	1 year experience in radiochemistry.
	Team leader			7							Knowledge of area of responsibility and high school diploma or equivalent required.

TABLE 13.1.3-1 (SHEET 2 OF 3)

		Recommended Experience (years)				License		Suggested Education (years)			
ANSI N18.1 Title	VEGP Title	Total Power Plant	Nuclear Power Plant	Other Applicable	Academic Training	Reactor Operator	Senior Reactor Operator	Academic	Related Technical Training	Amount of Education Creditable for Experience	Remarks
	Team leader			7							Knowledge of area of responsibility and high school diploma or equivalent required.
Foreman	HP or chemistry foreman			4							High school diploma or equivalent required.
I&C assistant team leaders				5				2 or	2	4	6 months experience in nuclear I&C.
Assistant team leader				4							Knowledge of area of responsibility and high school diploma or equivalent required.
Technical support personnel											
Engineer-in-charge	Engineering director[e]		3		4						
ANSI 3.1 Title											
Radiation protection manager	Health physics manager, plant health physicist, or HP support supervisor		3[c]	4				4			

TABLE 13.1.3-1 (SHEET 3 OF 3)

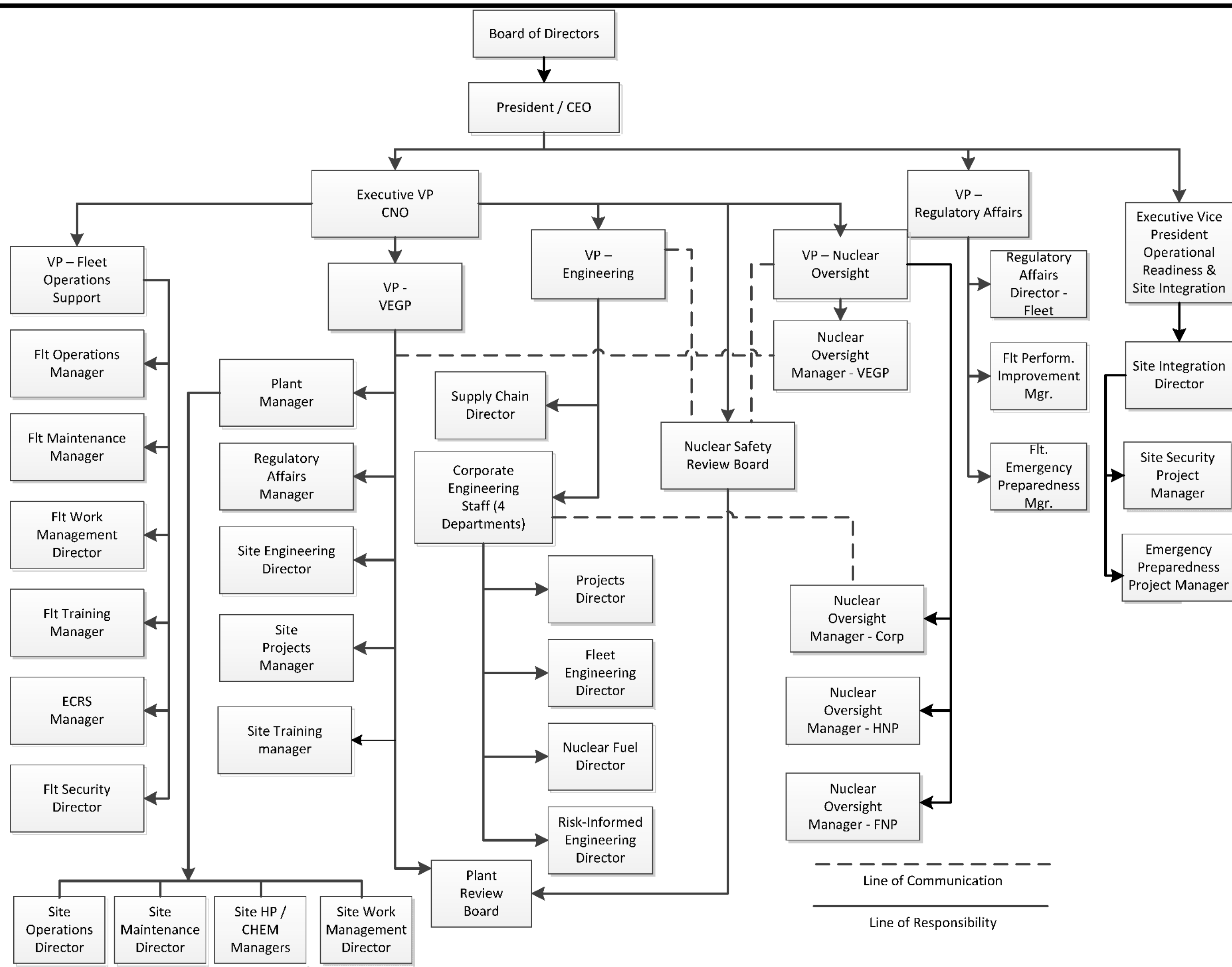
a. Senior reactor operator license is not required for the vice president-Vogtle, the plant manager, or the regulatory affairs manager, but the vice president-Vogtle or the plant manager will have (or have held) an SRO license or have the background required to sit for examination. See ANSI N18.1-1971 Section 4.2 for additional allowances if one or more of the requirements is not met for the Vice President-Vogtle or Plant Manager.

b. Required by NUREG-0737.

c. Either the health physics manager, the health physics support supervisor, or the plant health physicist will be assigned the radiation protection manager responsibilities. In addition to the above, the selected individual will have participated in radiation protection activities during at least 2 months of operation at greater than 20 percent power, a routine (1 to 2 months) refueling outage, and will have at least 6 months of nuclear experience at VEGP.

d. The operations director or at least one Operations Department Manager shall hold an SRO license.

e. The engineering director fulfills both the technical manager and engineer-in-charge functions; therefore, the engineering director must meet both sets of requirements.



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SOUTHERN COMPANY

VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

NUCLEAR OPERATIONS ORGANIZATION
VEGP UNITS 1 AND 2

FIGURE 13.1.1-1

THIS FIGURE HAS BEEN DELETED.

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SOUTHERN COMPANY

VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

SNC TECHNICAL SUPPORT ORGANIZATION
VEGP UNITS 1 AND 2

FIGURE 13.1.1-2

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SOUTHERN COMPANY VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2	NUCLEAR OPERATIONS ORGANIZATION VEGP UNITS 1 AND 2
	FIGURE 13.1.2–1

13.2 TRAINING PROGRAMS

Personnel meet the qualification and training recommendations of Regulatory Guide 1.8, Revision 2, and for the licensed staff, 10 CFR 55.59, before they are considered qualified to perform all duties independently. Prior to meeting the recommendations of Regulatory Guide 1.8, Revision 2, personnel are trained to perform specific tasks and are qualified to perform those tasks independently. Personnel who complete an accredited program endorsed by the Nuclear Regulatory Commission (NRC) meet the requirements of the accredited program in lieu of Regulatory Guide 1.8, Revision 2.

13.2.1 ACCREDITED TRAINING PROGRAMS

The VEGP training programs have been developed in accordance with the systems approach to training as described by the Institute for Nuclear Power Operations (INPO). The National Academy for Nuclear Training, through a formal accreditation process, verifies that VEGP training programs meet the established criteria. VEGP is a branch of the National Academy and has achieved accreditation of the following programs:

- Nonlicensed operator
- Reactor operator
- Senior reactor operator
- Continuing training for licensed personnel
- Shift supervisor
- Shift technical advisor
- Instrument and control (I&C) technician
- Electrical maintenance personnel
- Mechanical maintenance personnel and supervisor
- Radiological protection technician
- Chemistry technician
- Engineering support personnel

The training programs are periodically evaluated and revised as appropriate, and reviewed by management for effectiveness.

13.2.1.1 Training Program Records

Records to document course completion and documentation of qualification are retained as quality records in accordance with the SNC Quality Assurance Topical Report (QATR).

13.2.1.2 Simulator

The VEGP simulators conform to the guidance given in Regulatory Guide 1.149, Revision 4. The certification of the VEGP simulators as plant-referenced simulators is complete and certification shall be maintained as described in 10 CFR 55.45 (b)(5). Exceptions to ANSI 3.5 are listed in the certification report filed with the NRC every 4 years.

13.2.1.3 Instructor Qualification

Accreditation of training programs includes criteria for qualification and requalification of instructors. In addition, the fire protection training instructor also meets the requirements outlined in paragraph 9B.C.3.d(2).

13.2.2 OTHER TRAINING PROGRAMS

13.2.2.1 General Employee Training Program

All persons regularly employed at VEGP are trained in the following areas commensurate with their job duties:

- Fitness for duty
- General plant description
- Radiological protection
- Emergency preparedness
- Industrial safety
- Fire protection
- Security
- Quality assurance

Refresher training will be conducted on a periodic basis.

13.2.2.2 Fire Brigade Training

Personnel assigned to any fire brigade complete initial training and periodic retraining as designated by appendix 9B.

13.2.2.3 Quality Control Training Program

Plant personnel who perform quality control (QC) inspections, both receipt and in-plant, shall meet the minimum education and experience requirements of ASME NQA-1-1994 as described in the SNC QATR prior to being certified to perform inspections. Initial training consists of quality assurance procedures training, codes and standards training, and on-the-job training. On-the-job training for receipt personnel and QC inspection personnel is based on job requirements. Initial training has a minimum duration of 40 h. Material receipt inspection personnel are not required to receive additional training in plant systems.

After completing initial qualification, QC plant inspection and material receipt inspection personnel normally receive training annually to improve and maintain their inspection skills. This annual training will be in subjects applicable to their inspection duties and qualification.

13.2.2.4 Mitigating Core Damage Training Program

The VEGP training program for mitigating core damage is not a separate program for licensed personnel, but is integrated into licensed personnel training, senior reactor operator certified personnel training, and shift technical advisor training. Other personnel, including the health physics manager, chemistry manager, and team leaders and assistant team leaders for I&C technicians in the maintenance department, as well as the plant manager and the operations manager, complete training in mitigating core damage commensurate with their responsibilities.

13.3 EMERGENCY PLANNING

A comprehensive emergency plan for VEGP Units 1 and 2 is provided as a separate volume.

13.4 OPERATIONAL REVIEW

Operating activities that affect nuclear safety are reviewed. The review program was implemented prior to initial fuel loading and ensures review and evaluation of tests and experiments, unplanned events, and proposed changes. The program complies with the requirements of 10 CFR 50.59 relating to proposed changes, tests, and experiments. This program is conducted in accordance with the requirements of the SNC Quality Assurance Topical Report (QATR).

The vice president-Vogtle has responsibility for safe operation of the plant. He is kept abreast of plant operating conditions by the managers, superintendents, and supervisors who are knowledgeable of and experienced in their areas of job responsibility. These individuals monitor operating and maintenance activities as part of their normal duties.

In addition, a formal review program is carried out for changes to systems, procedures, tests, experiments, and after-the-fact review and evaluation of unplanned events that affect nuclear safety. The vice president-Vogtle shall approve, prior to implementation, each proposed test, experiment, or modification to systems or equipment that affect nuclear safety. In addition to this requirement for vice president-Vogtle approval as described above, this program is also implemented through standing committees, as referenced below.

13.4.1 ONSITE REVIEW

Refer to the SNC QATR for discussion related to the Plant Review Board (PRB), which performs the onsite review function.

13.4.2 INDEPENDENT REVIEW

Refer to the SNC QATR for discussion related to the Nuclear Safety Review Board (NSRB), which performs the independent review function.

13.5 PLANT PROCEDURES

This section describes administrative, maintenance, and operating procedures that are used by the operating organization to ensure that routine, off-normal, and emergency activities are conducted in a safe manner. Operations affecting safety are conducted in accordance with detailed written and approved procedures.

13.5.1 ADMINISTRATIVE PROCEDURES

Administrative procedures provide rules, instructions, policies, practices, or guidelines for the plant staff. VEGP administrative procedures establish controls which govern changes to programs and changes or modifications to systems or equipment affecting nuclear safety. These controls ensure that these programs and changes or modifications are independently reviewed by qualified personnel and approved by appropriate personnel.

13.5.1.1 Administrative Procedure Descriptions

A. Procedures for Preparation, Review, and Control of Procedures

These procedures establish the controls for the preparation, review, and control of plant procedures which cover activities described in the SNC Quality Assurance Topical Report (QATR).

Included within these procedures are provisions to ensure that new or revised procedures are reviewed for adequacy by appropriately qualified personnel other than the originator. In accordance with these procedures, a reviewer shall acquire knowledge of the applicable requirements for a procedure and shall verify the proper incorporation of these requirements in the procedure; additionally, a reviewer shall assess whether the procedure or procedure revision potentially involves a change in the VEGP Technical Specifications or requires prior NRC approval pursuant to 10 CFR 50.59. The acceptability of procedure reviews is assured by the review and approval by the responsible VEGP department heads or their designee. Department heads shall further ensure that procedures requiring Plant Review Board (PRB) review in accordance with the SNC QATR are forwarded to the PRB.

For procedures not forwarded to the PRB, reviewers will meet requirements of section 4.4 of ANSI N18.1-1971 for applicable disciplines. For those disciplines not described in section 4.4 of ANSI 18.1-1971, the reviewer will have a minimum of 5 years experience. A maximum of 4 years of this 5 years may be fulfilled by related technical or academic training. Reviewers of quality control inspection procedures shall meet the requirements of ASME NQA-1-1994, as described in the QATR. Also, those procedures not forwarded to the PRB and impacting another department's area of responsibility, shall be forwarded to the impacted departments for their review.

The vice president-Vogtle has ultimate responsibility for all plant procedures. Specific procedures for which vice president-Vogtle approval is required are discussed in plant procedures. Department heads are established as the approving authority for other procedures covering activities within their area of responsibility. Additional guidance regarding this approval authority for specific procedures is provided in plant procedures.

Additional provisions of these procedures exist to ensure that changes or revisions to procedures are reviewed and approved in accordance with the same administrative controls used for review and approval of new procedures. A provision is made to ensure that EOPs and AOPs are reviewed at least every 2 years by a knowledgeable individual to determine whether changes are necessary or desirable.

As a part of the overall quality assurance program, the QA group performs various audits (described in the QATR) to assure that the procedural process is working and that procedures are being properly maintained. Also, provisions exist to ensure that procedures, once approved, are distributed appropriately so that only the most current procedures are used by plant personnel.

B. Procedures for Making Temporary Changes to Procedures

This procedure provides the method for making a temporary change to an approved plant procedure.

Included in this procedure is a provision to ensure that any such change be approved by at least two knowledgeable members of the plant staff, one of whom shall hold a senior operator license. Also included in this procedure is a provision to ensure that temporary changes to procedures are documented and incorporated as appropriate into the next revision of the procedure.

Additional guidance regarding temporary changes to procedures for specific procedures is provided in the QATR.

C. Procedures for Feedback of Operating Experience

These procedures include measures to ensure that pertinent operating experience information originating from both within and outside of the plant organization is fed back to operators and other appropriate personnel in accordance with NUREG-0737, item I.C.5. This information can include industry event reports, vendor reports, inhouse event reports, and Nuclear Regulatory Commission publications. These procedures establish management responsibility for direction, control, and administration of the operations assessment program. Included in these procedures is the identification of organizational responsibilities for reviewing and prioritizing operating experience, and for ensuring distribution of pertinent information to the appropriate plant personnel. Steps exist to ensure that information is reviewed by individuals of appropriate technical knowledge and that appropriate corrective actions (such as procedure or program revisions), if needed, are specified. The procedures include steps which ensure records are maintained to reflect the status and disposition of operating experience evaluations and their associated corrective actions. Completion status and followup verification status of completed action is tracked to ensure implementation. Additional steps ensure that plant personnel do not routinely receive a large volume of operating experience that might obscure the lessons to be learned from more significant events, and also that the program receives periodic evaluation for effectiveness.

D. Procedures for Control Room Access

These procedures give the shift supervisor authority to limit access to the control room to those individuals responsible for the direct operation or support of the plant. Included are steps that direct personnel other than the onshift operations crew and the operations chain of command to request permission of the shift operations crew to enter the limited access area shown in figure 13.5.1-1. Also, these procedures establish good conduct rules for personnel within the control room area to avoid any disruption of operating activity. Responsibility for limiting control room access is a normal shift duty of the shift supervisor. The shift supervisor is additionally given the authority to require that any individual granted access to the control room immediately leave the control room if that individual interferes with the duties of the control room operators in maintaining safe operation of the plant, or exhibits behavior contrary to good conduct rules. These procedures address the requirements of NUREG-0737, item I.C.4, for control room access; while the requirements for establishing the lines of authority, responsibility, and succession in the control room are addressed by the procedures described in section E.

E. Procedures for Operating Duties, Responsibilities, and Authority

These procedures clearly describe the duties, responsibilities, and authority for the control room personnel, which include the shift manager, the shift supervisor, the shift support supervisor, the systems operator, the plant operator, and the shift technical advisor. The command line of authority for these personnel is established by these procedures. Specifically, the shift manager is established as the senior operations representative on each shift and shall have responsibility for the safe and efficient operation of the plant. The shift supervisor is established as having responsibility for the safe and efficient operation of his assigned unit, since a shift supervisor is assigned to each operating unit on each shift. Included in these procedures are provisions for the shift manager to maintain a broad perspective of operational conditions affecting the safety of the plant at all times and provisions for him not to become totally involved in any single operation during plant transients or emergency conditions. Other provisions establish the shift manager as having the authority and responsibility to declare emergencies, and to function as the emergency director, until being relieved of this responsibility by a higher-ranking qualified manager.

Provisions for ensuring that the shift manager is not routinely performing administrative functions that could detract from or that are subordinate to his command function and his responsibility for ensuring the safe operation of the plant are exemplified by the following:

1. An operations individual who is not on shift shall prepare work and vacation schedules.
2. The shift supervisor shall issue clearances for equipment within his assigned unit.
3. The shift supervisor shall have the administrative duty of limiting access to the control room.

The plant manager annually reviews the administrative duties of the shift manager to ensure that he is not routinely performing administrative duties as described above.

The procedures described in this section are in accordance with requirements of NUREG-0737, item I.A.1.2, item I.C.3, and item I.C.4. The process for handling standing orders is addressed in section F.

F. Procedures for Standing Orders

These procedures provide for the issue of temporary instructions to plant operating personnel to address subjects not covered by existing plant operating procedures. Included in these procedures are provisions for maintaining and implementing approved standing orders, and periodically reviewing standing orders for continued applicability.

G. Procedures for Shift Manning and Overtime Restrictions

These procedures establish the normal and minimum shift positions that must be manned for operation of the plant. Included is the number of individuals to fill these normal and minimum positions for both one-unit and two-unit operation. These procedures restrict the use of overtime that may be scheduled to meet the shift crew staffing requirements, such that overtime use does not exceed the requirements of 10 CFR Part 26, Subpart I.

H. Procedures for Shift Relief and Turnover

These procedures ensure that a comprehensive exchange of information takes place between the oncoming and offgoing shift personnel so that the oncoming shift is aware of critical plant status information and system availability prior to assuming duty. Included are provisions to ensure that each oncoming licensed operator reviews and signs the checklist for his position prior to assuming his duties. Both licensed and nonlicensed operators review logs applicable to their positions prior to assuming their duties. In addition, nonlicensed operators complete round sheets early in each shift. All oncoming individuals discuss relevant items affecting plant operation with the offgoing individuals prior to assuming their duties. Provisions also include ensuring that an individual is qualified for the position that he will assume. These procedures establish, as part of the offgoing control room operator responsibility, the need to ensure that his relief is fully aware of existing plant conditions, and that his relief is alert, coherent, and fully capable of performing his assigned duties. Provisions are provided to ensure that other appropriate supervisory personnel, including maintenance performance team shift personnel, develop shift logs to update oncoming shifts of plant activities. These provisions also ensure that oncoming shift personnel review and sign shift logs for their position. These procedures are in accordance with the requirements of NUREG-0737, item I.C.2.

I. Equipment Control Procedures

These procedures provide instructions for releasing plant equipment or systems for maintenance, testing, or inspection; they establish the shift supervisor as the responsible authority for issuing and releasing clearances for equipment to be taken out of service within his assigned unit. The provisions of these procedures include steps to ensure that equipment taken out of service and placed in a controlled status is clearly identified by the use of tagging. Other provisions provide for a second qualified person verifying the isolation or restoration of a safety-related component or system, including proper realignment, unless functional testing can be performed to prove the correct realignment of all equipment, valves, and switches involved. A second verification of safety-related system alignment will not be made in cases of significant radiation exposure, or where alignment changes or alterations of status do not render the component/systems incapable of performing designed safety functions. Additional provisions are made to maintain the status of equipment and to determine the operability of equipment upon return to service; on return to service, a system lineup verification will be made, and additional lineup verifications may be made at periodic intervals while in service. These procedures address the requirements of NUREG-0737, item I.C.6 and item II.K.1.10, and the requirements of ANS N18.7-1976, Section 5.2.6, relative to equipment control.

J. Maintenance and Modification Administrative Control Procedures

Maintenance of equipment and plant modifications important to plant safety are performed in accordance with written procedures as described in paragraph 13.5.2.2. Administrative controls are provided to ensure compliance with applicable codes, regulations, and requirements.

K. Fire Protection Procedures

The VEGP fire protection program is governed and implemented through the use of fire protection procedures. These procedures provide guidelines for the following: administrative controls, system operation, firefighting activities, fire brigade and general personnel training, and agreements with local offsite fire departments, as described in subsection 9.5.1.

L. Crane Operation Procedures

Crane administrative operating procedures require that crane operators who operate cranes over fuel pools be qualified and conduct themselves in accordance with ANSI B30.2-1976 (chapters 2 and 3), Overhead and Gantry Cranes.

M. Temporary Procedures

Temporary procedures are issued, as required, to provide detailed instructions for specific jobs lasting a specific duration.

N. Emergency Core Cooling System Outage Data Collection Procedures

As discussed in Generic Letter 83-37, the NRC has completed their review of ECCS data and determined that no changes in the Technical Specifications were required. Therefore, the surveillance and reporting requirements in the VEGP Technical Specifications adequately address ECCS outages, and this data will no longer be reported to the NRC.

O. Plant Security Procedures

Plant security procedures provide for the implementation of the security plan. (See section 13.6.)

13.5.2 OPERATING AND MAINTENANCE PROCEDURES

13.5.2.1 Operating Procedures

These procedures are for operation of plant equipment. These procedures are developed and provided to ensure an effective system for verifying the correct performance of operating activities and to reduce human error. These procedures meet the intent of NUREG-0737, item I.C.6. Conformance of these procedures with ASME NQA-1-1994 is discussed in the SNC QATR.

Although format can be varied depending on their nature, these procedures usually have the following format:

1.0 PURPOSE

The purpose for which the procedure is intended is to be briefly and clearly stated.

2.0 PRECAUTIONS AND LIMITATIONS

Precautions are to be established to alert the individual performing the task of those important measures which will be used to protect equipment and personnel, or will serve to avoid an abnormal or emergency situation.

Limitations on the parameters being controlled and appropriate corrective measures to return the parameter to normal are established. Control band may be specified where applicable.

3.0 PREREQUISITES OR INITIAL CONDITIONS

Each procedure will identify those independent actions or procedures which must be completed, or those plant conditions which must exist, prior to its use. Prerequisites applicable only to certain sections of a procedure will be so identified.

4.0 MAIN BODY

The main body of a procedure will contain step by step instructions in the degree of detail necessary for a qualified person performing a function or task.

Subsections of this section will vary according to procedure type. Format for each procedure type will be consistent.

5.0 ACCEPTANCE CRITERIA

Procedures will contain, where applicable, acceptance criteria against which the success or failure of test type activity may be judged. This section may be omitted if none are required.

6.0 REFERENCES

References, including reference to Technical Specifications, will be included in procedures as applicable.

These procedures are classified as follows:

A. Plant/Unit Operating Procedures

These are procedures for integrated plant operations, such as mode changes.

B. System Operating Procedures

These procedures are for normal, alternate means of operation and limits of operation of individual plant systems, including removal from service and return to service.

C. Surveillance Procedures (Operations Department Responsibility)

These procedures, which describe how to perform testing functions, are required to verify operational readiness of structures, systems, and components.

D. Annunciator Response Procedures

These procedures specify responses to annunciated alarms. They outline automatic system response and operator actions to mitigate the conditions.

These procedures are indexed by light board number and alarm window coordinates to facilitate operator response. Any annunciator response procedure providing response to an emergency condition reference the appropriate emergency operating procedure.

E. Abnormal Operating Procedures

These procedures specify operator actions for restoring a plant operating variable to its normal controlled value when it departs from its range or operator actions necessary to restore normal operating conditions following a transient. Such actions are invoked following an operator observation or any annunciator alarm indicating a condition which, if not corrected, could degenerate into a condition requiring action under an emergency operating procedure.

F. Emergency Operating Procedures

The emergency operating procedures are developed according to an emergency operating procedure writer's guide.

The emergency operating procedures are developed according to an action plan, which incorporates developmental information provided in the implementation plan guideline published by the Institute of Nuclear Power Operations.

Technical guidelines (emergency response guidelines) and accompanying background/support information are supplied by the Westinghouse Owners Group. The generic background/support information is made plant specific. In addition, each generic technical guideline is reviewed to determine the information needed to generate plant-specific procedures. An emergency operating procedures writer's guide is developed prior to generating these procedures from the Westinghouse Owners Group technical guidelines. In summary, the emergency operating procedures are developed using the Westinghouse Owners Group technical guidelines, plant-specific background/support information, and the emergency operating procedures writer's guide.

Each completed procedure is subjected to a two-part review in accordance with the emergency operating procedures verification program. The first part of this review verifies verbal accuracy, while the second part ensures technical accuracy. Upon completion of the verification program, the procedures are tested in accordance with the emergency operating procedures validation program. The validation program requires that all procedures be validated by at least one of the following (in order of preference for validation): using the plant-specific simulator; actual testing on plant systems; or detailed review.

Use of the Westinghouse Owners Group technical guidelines and supplemental information, in accordance with the emergency operating procedures action plan, meets the intent of NUREG-0899, items I.C.1 and I.C.8, as clarified by NUREG-0737, Supplement 1.

G. Temporary Procedures

Temporary procedures are issued, as required, to provide detailed instructions for specific jobs that are of a specific duration.

13.5.2.2 Maintenance Procedures

The VEGP maintenance procedures are divided into the following categories:

A. Generic Procedures

These procedures, which are general in nature, cover maintenance activities that can be performed on many different pieces of equipment or instrumentation; i.e., vibration analysis, inspections, insulation resistance checks, calibrations, functional tests, etc.

B. Specific Maintenance Activity Procedures

These procedures give instructions for service, repair, calibration, or functional testing of specific pieces of equipment, components, or instrumentation.

C. Surveillance Procedures

These procedures describe how to perform testing functions or parts replacement, as required, to verify the operational readiness of structures, systems, components, and instrumentation.

D. Special Procedures

Procedures for the maintenance, repair, calibration and functional testing of health physics, fire protection, and security equipment are also provided.

The maintenance department has responsibility for work performed in accordance with these procedures. Plant maintenance and modifications are also governed by administrative procedures.

13.5.2.3 Other Procedures and Program

13.5.2.3.1 Procedures

Other procedures are provided in the following areas:

A. Health Physics Procedures

Plant radiation protection procedures are designed to limit and control radiation exposure and the spread of contamination, as well as to meet the requirements of 10 CFR 20 and the as low as reasonably achievable philosophy.

The procedures describe rules, and practices or guidelines, for personnel protection, radiation surveys, decontamination, handling of radioactive or contaminated materials, and implementation of the as low as reasonably achievable program. The health physics department has responsibility for implementing these procedures and for ensuring the compliance of the plant staff with them.

B. Laboratory Procedures

These procedures describe rules, practices, or guidelines for tests, analyses, additions, or dilutions with respect to plant chemistry and radiochemistry. The health physics and chemistry laboratories are responsible for performing these activities.

C. Refueling Procedures

These procedures provide for preparation and performance of refueling operations. They include procedures to disassemble components, refueling equipment preuse maintenance and checkouts, and methods and limits for performing refueling operations.

D. Emergency Plan Implementation Procedures

These procedures provide rules and practices and designate responsibility and authority for classifying emergencies and responses to such emergencies. The plant staff has the responsibility to follow these procedures. Procedures in this section implement the emergency plan.

13.5.2.3.2 Program

The Process Control Program (PCP) is described below.

A. The PCP shall contain the current formulas, sampling, analyses, tests, and determinations to be made to ensure that processing and packaging of solid radioactive wastes based on demonstrated processing of actual or simulated wet solid wastes will be accomplished in such a way as to assure compliance with 10 CFR Parts 20, 61, and 71, State regulations, burial ground requirements, and other requirements governing the disposal of solid radioactive waste.

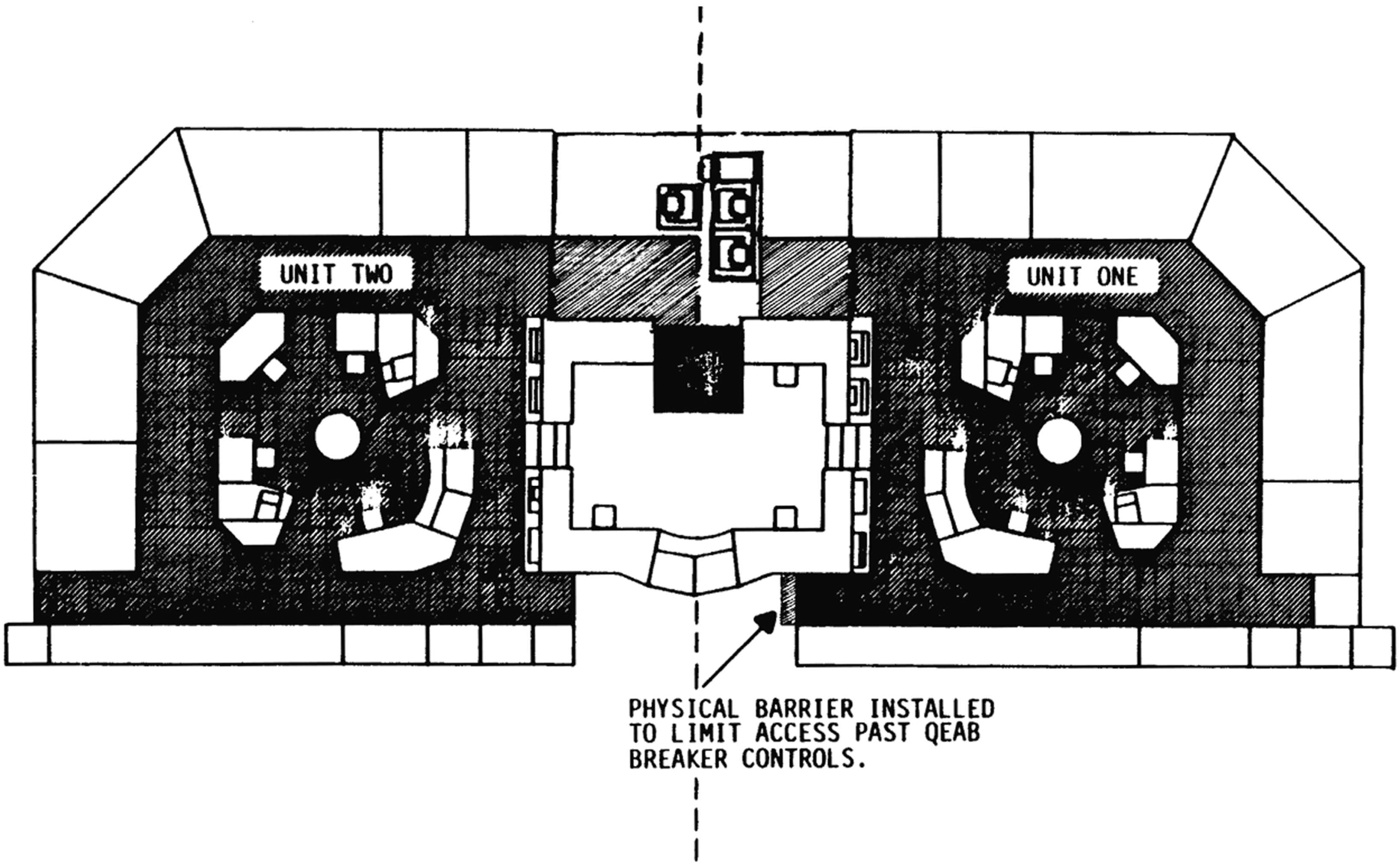
B. Licensee-initiated changes to the PCP:

1. Shall be documented and records of reviews performed shall be retained as required by paragraph 13.7.1. This documentation shall contain:

   a. Sufficient information to support the change together with the appropriate analyses or evaluations justifying the change(s), and

   b. A determination that the change will maintain the overall conformance of the solidified waste product to existing requirements of Federal, State, or other applicable regulations.

2. Shall become effective after review and acceptance by the PRB and the approval of the vice president-Vogtle.



THE SHADED AREAS (DIAGONAL LINES) DESIGNATE THE LIMITED ACCESS AREAS.

REV 14 10/07

SOUTHERN COMPANY *Energy to Serve Your World®*	VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2	LIMITED ACCESS AREA FIGURE 13.5.1–1

13.6 INDUSTRIAL SECURITY

A description of the physical security program for VEGP Units 1 and 2 has been provided to the Nuclear Regulatory Commission as a separate part of the application which is withheld from public disclosure pursuant to paragraph 10 CFR 2.790(d), Rules of Practice.

13.7 PLANT RECORDS

Records documenting the quality of the design, construction, testing, operation, maintenance, and modification of the Vogtle Electric Generating Plant (VEGP) are stored and maintained in accordance with the requirements of Criteria XVII of Appendix B of 10 CFR 50 as described in the SNC Quality Assurance Topical Report (QATR).

13.7.1 RECORD RETENTION

In addition to the applicable record retention requirements of Title 10, Code of Federal Regulations, the following records shall be retained for at least the minimum period indicated.

The following records shall be retained for at least 5 years:

A. Records and logs of plant operation covering time interval at each power level;

B. Records and logs of principal maintenance activities, inspections, repair, and replacement of principal items of equipment related to nuclear safety;

C. All reportable events;

D. Records of surveillance activities, inspections, and calibrations required by technical specifications;

E. Records of changes made to the procedures required by the QATR;

F. Records of radioactive shipments;

G. Records of sealed source and fission detector leak tests and results; and

H. Records of annual physical inventory of all sealed source material of record.

The following records shall be retained for the duration of the plant operating license:

A. Records and drawing changes reflecting plant design modifications made to systems and equipment described in the FSAR;

B. Records of new and irradiated fuel inventory, fuel transfers, and assembly burnup histories;

C. Records of radiation exposure for all individuals entering radiation control areas;

D. Records of gaseous and liquid radioactive material released to the environs;

E. Records of transient or operational cycles for those plant components identified in table 3.9.N.1-2;

F. Records of reactor tests and experiments;

G. Records of training and qualification for current members of the plant staff;

H. Records of inservice inspections performed pursuant to the VEGP inservice inspection program;

I. Records of quality assurance activities required by the FSAR;

J. Records of reviews performed for changes made to procedures or equipment or reviews of tests and experiments pursuant to 10 CFR 50.59;

K. Records of meetings of the PRB and SRB;

L. Records of the service lives of all hydraulic and mechanical snubbers required by the technical requirements manual, including the date at which the service life commences and associated installation and maintenance records;

M. Records of secondary water sampling and water quality;

N. Records of analyses required by the radiological environmental monitoring program that would permit evaluation of the accuracy of the analysis at a later date. This should include procedures effective at specified times and QA records showing that these procedures were followed; and

O. Records of reviews performed for change made to the offsite dose calculation manual (ODCM) and the process control program (PCP).

13.8 RISK INFORMED CATEGORIZATION AND TREATMENT

13.8.1 INTRODUCTION

On November 22, 2004, the NRC issued 10 CFR 50.69 that presented nuclear management with an opportunity to further enhance equipment reliability and plant safety by focusing on those critical structures, systems, and components (SSCs) with the highest safety significance. The rule broadly adjusts the scope of safety-related components that are subject to the existing NRC regulations. The implementation of this new rule is strictly voluntary on the part of each licensee.

Current NRC regulations define the plant equipment necessary to meet the deterministic regulatory basis as “safety-related.” This equipment is subject to NRC special treatment regulations. These special treatment requirements go beyond normal commercial and industrial practices and include quality assurance (QA) requirements, qualification requirements, inspection and testing requirements, and Maintenance Rule requirements. Other plant equipment is classified as “nonsafety-related’” and is typically not subject to special treatment requirements. For those components that are categorized as Low Safety Significant, 10 CFR 50.69 (b)(1) allows compliance with alternative requirements in lieu of the following special treatment requirements.

(i) 10 CFR part 21.

(ii) The portion of 10 CFR 50.46a(b) that imposes requirements to conform to Appendix B to 10 CFR part 50.

(iii) 10 CFR 50.49.

(iv) 10 CFR 50.55(e).

(v) The in-service testing requirements in 10 CFR 50.55a(f):

The in-service inspection and repair and replacement (with the exception of fracture toughness), requirements for ASME Class 2 and Class 3 SSCs in 10 CFR 50.55a(g); and the electrical component quality and qualification requirements in Section 4.3 and 4.4 of IEEE 279, and Sections 5.3 and 5.4 of IEEE 603-1991, as incorporated by reference in 10 CFR 50.55a(h).

(vi) 10 CFR 50.65, except for paragraph (a)(4).

(vii) 10 CFR 50.72.

(viii) 10 CFR 50.73.

(ix) Appendix B to 10 CFR part 50.

(x) The Type B and Type C leakage testing requirements in both Options A and B of Appendix J to 10 CFR part 50, for penetrations and valves meeting the following criteria:

- (A) Containment penetrations that are either 1-in. nominal size or less, or continuously pressurized.
- (B) Containment isolation valves that meet one or more of the following criteria:
  - (1) The valve is required to be open under accident conditions to prevent or mitigate core damage events;

(2) The valve is normally closed and in a physically closed, water-filled system;

(3) The valve is in a physically closed system whose piping pressure rating exceeds the containment design pressure rating and is not connected to the reactor coolant pressure boundary; or

(4) The valve is 1-in. nominal size or less.

(xi) Appendix A to part 100, Sections VI(a)(1) and VI(a)(2), to the extent that these regulations require qualification testing and specific engineering methods to demonstrate that SSCs are designed to withstand the safe shutdown earthquake and operating basis earthquake.

It should be noted that 10 CFR 50.69 does not replace the existing “safety-related” and “nonsafety-related” classification. Instead, 10 CFR 50.69 divides these classifications into two subcategories based on high or low safety significance, such that there are four categories of risk-informed safety class (RISC), as shown below:

RISC-1: safety-related SSCs that perform (high) safety significant functions.

RISC-2: nonsafety-related SSCs that perform (high) safety significant functions.

RISC-3: safety-related SSCs that perform low safety-significant functions.

RISC-4: nonsafety-related SSCs that perform low safety-significant functions.

When applying alternative treatment, 10 CFR 59.69 requires that the licensee “shall ensure, with reasonable confidence, that RISC-3 SSCs remain capable of performing their safety-related functions under design basis conditions, including seismic conditions and environmental conditions and effects throughout their service life.”

Southern Nuclear received approval from the NRC to implement 10 CFR 50.69 at Vogtle 1 & 2 as outlined in reference 1.

13.8.2 SSC CATEGORIZATION

As outlined in the Safety Evaluation Report (SER), Plant Vogtle 1 & 2 will use the methodology outlined in NEI 00-04, 10 CFR 50.69 SSC Categorization Guidance, Revision 0, to categorize SSCs. For pressure retention components, consistent with the NRC SER, Plant Vogtle 1 & 2 will use the passive component categorization method as approved by the NRC for ANO-2 and as outlined in reference 2.

The SSC categorization process is outlined in plant procedures and associated instructions.

10 CFR 50.69 (f)(2) requires updating the UFSAR to reflect which systems have been categorized. The following table is revised as part of the periodic UFSAR update to reflect systems that have been categorized.

System Name	System Designator
Containment Spray	1206
Radiation Monitoring	1609
Component Cooling Water	1203
Essential Chilled Water	1592
Chemical Volume and Control	1208
Nuclear Instrumentation	1602
Plant Safety Monitoring	1623
Nuclear Service Cooling Water	1202

13.8.3 SSC TREATMENT

13.8.3.1 Treatment of Component Categories

The programs or processes that implement the special treatment requirements are revised to recognize that the special treatments no longer apply to RISC-3 components. The programs or processes either allow continued application of the special treatments or acceptable alternative treatments, as applicable, to provide reasonable confidence that these components would perform their safety-related function under design basis accidents.

The following information provides the general approach for applying treatment for the component categories:

A. RISC-1 Components

The purpose of treatment applied to RISC-1 SSCs is to maintain compliance with NRC regulations and to ensure that these SSCs can perform their risk-significant functions consistent with the categorization process. These components continue to receive existing treatment required by NRC regulations and associated Vogtle 1 & 2 implementing procedures.

Some safety-related components may be credited for performing functions that are beyond the design basis or for performing safety-related functions under conditions that are beyond the design basis. The Vogtle 1 & 2 PRA does not take credit for such functions unless there is a basis for confidence that the component will be able to perform the functions (e.g., demonstrated ability of the component to perform the functions under the specified conditions). A technical basis will be confirmed for RISC-1 components credited for performing functions that are beyond the design basis or for performing safety-related functions under conditions that are beyond the design basis and, if not confirmed, a technical basis will be developed for the credit taken, potentially including the creation of a treatment for the SSC that validates the capability credited.

B. RISC-2 Components

The purpose of treatment applied to RISC-2 SSCs is to maintain their ability to perform risk-significant functions consistent with the categorization process. These components will continue to receive any existing special treatment required by NRC regulations and Vogtle 1 & 2 programs. Additionally, the risk-significant functions of these components will receive consideration for enhanced treatment. This consideration is described in paragraph 13.8.3.2.

C. RISC-3 Components

These components may receive alternative treatments, in lieu of previous special treatments, as described in the affected programs.

D. RISC-4 Components

The treatment of these components is not subject to regulatory control.

E. Uncategorized Components

Until a component is categorized, it continues to receive the special treatment required by NRC regulations and associated Vogtle 1 & 2 implementing programs, as applicable.

13.8.3.2 Enhanced Treatment of RISC-2 SSCs

Nonsafety-related HSS components may perform risk-significant functions that are not addressed by the special treatment requirements in NRC regulations or current Vogtle 1 & 2 programs.

When a nonsafety-related component is categorized as HSS, Plant Vogtle determines whether enhanced treatment is warranted to enhance the reliability and availability of the component in support of its HSS function(s). In particular, Plant Vogtle evaluates the treatment applied to the component to ensure that the existing controls are sufficient to maintain the reliability and availability of the component in a manner that is consistent with its categorization. This process evaluates the reliability of the component, the adequacy of the existing controls, and the need for any changes. If changes are needed, additional controls are applied to the component. In addition, the component is placed under the Maintenance Rule monitoring program, if not already scoped in the program. Additionally, as provided in the Nuclear Quality Assurance Topical Report (QATR), nonsafety-related HSS components are subject to the enhanced QA program. These controls will be specifically targeted to ensure reliability of the critical attributes that resulted in the component being categorized as HSS. Components under these controls will remain nonsafety-related, but the enhanced treatments will be appropriately applied to give additional confidence that the component will be able to perform its HSS function(s) when demanded.

These identified processes provide reasonable confidence that HSS components will be able to perform their risk significant functions. The validation of functionality of HSS SSCs (safety-related SSCs for which existing special treatment does not provide the applicable level of confidence and nonsafety-related SSCs) will consist of a documented technical evaluation under the corrective action program to determine what enhanced treatment, if any, is warranted for these SSCs to provide reasonable confidence that the applicable risk significant functions will be satisfied. The performance of these SSCs will be monitored to provide reasonable confidence for their ongoing capability to perform their risk significant functions. The design control process will evaluate facility changes affecting the risk-significant functions of these SSCs.

13.8.4 REFERENCES

1. Safety Evaluation by the Office of Nuclear Reactor Regulation related to Amendment No. 173 to Renewed Facility Operating License NPF-68 and Amendment No. 155 to Renewed Facility Operating License NP-81, Docket Nos. 50-424 and 50-425, Vogtle Electric Generating Plants, Units 1 and 2, ML14237A034.
2. Safety Evaluation by the Office of Nuclear Reactor Regulation Request for Alternative AN02-R&R-004, Revision 1, Request to Use Risk-Informed Safety Classification and Treatment for Repair/Replacement Activities in Class 2 and 3 Moderate and High Energy Systems, Third and Fourth 10-Year In-service Inspection Intervals," Docket No. 50-368, Arkansas Nuclear One Plant Unit 2, ML090930246.

14.0 INITIAL TEST PROGRAM

14.1 SPECIFIC INFORMATION TO BE INCLUDED IN PRELIMINARY SAFETY ANALYSIS REPORTS

This information is applicable only to a Preliminary Safety Analysis Report.

14.2 INITIAL TEST PROGRAM

14.2.1 SUMMARY OF TEST PROGRAM AND OBJECTIVES

This chapter provides information covering the initial test program of the VEGP. It encompasses the period following installation and construction-related inspections and tests of components and systems, through the completion of power ascension testing.

The initial test program established for the VEGP demonstrates that components and systems operate in accordance with design requirements. This test program is based on the requirements of 10 CFR 50, Appendix B, Criterion XI. The initial test program is conducted in two phases: the preoperational test program and the startup test program. At the conclusion of these phases, the plant is ready for commercial operations.

Prior to preoperational testing of a particular system, certain prerequisite and construction tests will be conducted in order to verify the integrity, proper installation, cleanliness, and functional operability of the system components.

Based on the experience gained on Unit 1, the preoperational test phase for Unit 2 has been enhanced to incorporate a new subphase called release/release for test. This allows for a more interactive interface with construction than did Unit 1. The strategy for Unit 2 is to allow the commencement of construction acceptance test, and preventive maintenance on components and systems that installation/construction is complete but has not been turned over to the nuclear operation department. The controls for this interface are described in the VEGP startup manual. Preoperational and acceptance tests will commence only after the component or system has been turned over and certain prerequisite and construction acceptance tests have been conducted to verify the integrity, proper installation, cleanliness, and functional operability of the system components.

Conformance with the recommendations of Regulatory Guide 1.68, Initial Test Programs for Water-Cooled Reactor Power Plants, is discussed in subsection 14.2.3.

14.2.1.1 Preoperational Test Program

The preoperational test program is that part of the initial test program that commences with the completion of construction and construction-related inspections and tests and terminates with the commencement of nuclear fuel loading.

The preoperational test program (table 14.2.1-1) includes tests on both safety-related and nonsafety-related systems. The tests conducted on safety-related systems will demonstrate the capability of the structures, systems, and components to meet performance requirements and design criteria. The nonsafety-related tests will verify the system's or component's operability.

As a minimum, testing will be conducted on those systems that:

A. Are relied upon for safe shutdown and cooldown of the reactor under normal plant conditions and for maintaining the reactor in a safe condition for an extended shutdown period.

B. Are relied upon for safe shutdown and cooldown of the reactor under transient and postulated accident conditions and for maintaining the reactor in a safe condition for an extended shutdown period following such conditions.

C. Are relied upon for establishing conformance with safety limits or limiting conditions for operations.

D. Are classified as engineered safety features actuation systems (ESFAS) or are relied upon to support or ensure operation of ESFAS within design limits.

E. Are assumed to function during an accident or for which credit is taken in the accident analysis.

F. Are used to process, store, control, or limit the release of radioactive material.

The objectives of the preoperational test program are to:

A. Verify that plant components and systems, as constructed, fulfill their design intent.

B. Demonstrate, to the extent practicable, proper system/component response to postulated accidents.

C. Familiarize plant operating, technical, and maintenance personnel with plant operation.

Due to the proven design resulting from the Unit 1 test program, the Unit 2 testing methodology differs from Unit 1 in that, acceptance tests instead of preoperational tests will be performed on the nonsafety-related systems which have tests listed in table 14.2.1-1 and are not required to have preoperational tests performed.

The objective of acceptance tests is to demonstrate the operability of the plant components and systems which are nonsafety-related.

14.2.1.2 Startup Test Program

The startup test program is that part of the initial test program that commences with the start of nuclear fuel loading and terminates with the completion of power ascension testing. The startup tests (table 14.2.1-2):

A. Ensure that fuel loading is accomplished in a safe manner.

B. Confirm the design basis.

C. Demonstrate, where practical, that the plant operates and responds to anticipated transients and postulated accidents.

D. Ensure that the plant can be safely brought to rated capacity and can sustain power operation.

The objectives of the startup test program are to:

A. Accomplish a controlled, orderly, and safe initial core loading.

B. Accomplish a controlled, orderly, and safe initial criticality.

C. Conduct low-power testing sufficient to ensure that design parameters are satisfied and safety analysis assumptions are conservative.

D. Perform a controlled, orderly, and safe power ascension with testing terminating at plant-rated conditions.

E. Provide sufficient testing of transient and accident conditions to verify safe operation during transient or accident conditions.

The completion of startup testing constitutes the completion of the initial test program.

14.2.2 TEST PROCEDURES

The initial test program is conducted in accordance with detailed preoperational and startup test procedures. The general-manager Vogtle nuclear operations maintains the overall responsibility for initial test program procedure preparation, review, and approval. These activities shall be completed so that preoperational test and acceptance test (Unit 2 only) procedures listed in this chapter will be available 60 days prior to implementation and so that startup test procedures will be available 60 days prior to fuel loading for review by the Nuclear Regulatory Commission.

The following paragraphs describe the general methods employed to control procedure development and review. The detailed controls and methods are described in the Startup Manual and Startup Test Administrative Procedures.

14.2.2.1 Procedure Preparation

The general manager-Vogtle nuclear operations is responsible for preparation of initial test program procedures. Technical assistance is provided by Georgia Power Company (GPC), Bechtel Western Power Corporation, Westinghouse Electric Corporation, and Southern Company Services to develop detailed procedures as described below.

The cognizant design organization will review the objectives and acceptance criteria of the initial test program procedures. Comments resulting from the review will be provided for any technical comments not incorporated. This review will occur prior to procedure implementation.

The test procedures are prepared using the latest design information available. This information is utilized in developing the detailed test methods which verify the ability of systems and components to function within their design specifications.

The format and content of the test procedures developed for the initial test program for VEGP reflect the guidance provided in Regulatory Guide 1.68. The format consists of the following sections:

A. Objectives

   This section identifies the objectives of the test. They shall be clear, concise, and not quantitative. Examples of some of the possible contents of this section are listed below:

   1. Normal starting and stopping, both manual and automatic.
   2. Normal operation.
   3. Control from local and remote stations.
   4. Loss of motive or control power alarm verification.
   5. System parameter verification.
   6. System capacity verification.

B. References

Reference documents used in the preparation of the test procedure shall be listed in this section. Drawings, both domestic and vendor, will be listed with the revision of the document used.

C. Test Equipment

This section shall list the test equipment, other than installed plant equipment, used to perform the test and collect data. The name and model number of each piece of equipment shall be listed along with ranges where applicable. The test equipment to be used is determined by the test requirements and vendor information.

D. Notes and Precautions

This section contains any notes and/or precautions applicable during the performance of the test. Items generally pertinent to the entire test are included in this section. Specific precautions are to be included in the text of the test just prior to the step to which they apply.

E. Prerequisites and Initial Conditions

Prerequisites are those items which must be completed prior to performance of the test or subsection of the test.

Conditions that must exist prior to starting the test (i.e., alignment, temporary modifications, support systems, etc.) are initial conditions and shall be listed in this section.

F. Test Procedure

The test procedure section provides the detailed step- by-step instructions required to demonstrate that test acceptance criteria are satisfied and to obtain baseline operating data. The instructions must include activities that demonstrate that system and component performances meet acceptance criteria.

This section is subdivided by headings for the purpose of procedure organization and clarity. Each subdivision consists of a continuous series of operations prepared under the assumption that performance of a subsection, once started, will be carried to completion without interruption.

Test data are normally recorded in the body of the procedure immediately following the steps that set up the required test conditions and adjacent to acceptance criteria where practicable. However, where extensive or repetitive lists of data are required, data sheets may be used.

G. Test Data Sheets

The test results (of more than a half-page accumulation) shall be entered onto data sheets contained in this section. Each data sheet shall be numbered sequentially beginning with data sheet 7.1.

H. System Restoration

This section will provide, if necessary, instructions beyond those in the test procedure section for restoring the systems. Jumper removal and disconnection of temporary instruments are examples of section content.

I. Acceptance Criteria

The acceptance criteria section identifies the criteria which shall be met to verify that the system performance is acceptable.

Values and tolerances shall be specified for quantitative criteria. However, some acceptance criteria may be qualitative in nature.

J. Attachments

This section shall contain any attachments necessary to fully document the test results. Recorder traces, computer printouts, pump curves, etc., may be included as attachments.

14.2.2.2 Procedure Review and Approval

14.2.2.2.1 Preoperational/Startup Test Procedures

The general manager-Vogtle nuclear operations (GMVNO), or his designee is responsible for the review and approval of preoperational test procedures and startup test procedures. The general manager's designee for preoperational test procedures is the assistant plant startup manager; for startup tests his designee is the manager operations. In general preoperational test procedures and startup test procedures go through a minimum of two review cycles prior to submittal to the test review board for preoperational test procedures and plant review board for startup test procedures. Each cycle includes reviews by designated Georgia Power Company personnel, Bechtel Western Power Corporation representative, and/or the Westinghouse Electric Corporation representative as appropriate.

When the Unit 1 preoperational test procedure forms the basis for the Unit 2 preoperational test procedure, only one review cycle will be used prior to submittal to the test review board.

As a minimum, the preoperational and startup test procedures listed in this chapter are submitted for review to the test review board or the plant review board. Test procedures that are not satisfactory are resolved under the direction of the board prior to recommendation for approval. The test review board or the plant review board recommends approval to the GMVNO or his designee of procedures found acceptable. The GMVNO or his designee approves the procedure.

Subsequent revisions to the preoperational test procedure and startup test procedure are handled in a manner similar to a new procedure.

Qualifications of the above plant personnel who review and approve preoperational and startup test procedures are addressed in section 13.1.2.

The above processes are further described in the VEGP startup manual and in the startup test administrative procedures.

14.2.2.2.2 Acceptance Test Procedures (Unit 2 only)

The general manager-Vogtle nuclear operations, or his designee is responsible for the review and approval of acceptance test procedures for nonsafety related systems/components identified in table 14.2.1-1. The general manager's designee for acceptance test procedures is the assistant plant startup manager (APSM). In general, acceptance tests go through a minimum of two review cycles prior to submittal to the assistant plant startup manager for approval. Each cycle includes reviews by the designated Georgia Power Company test supervisor, and either the Bechtel Western Power Corporation, or the Westinghouse Electric Corporation representative, whichever is designated responsible for the procedure.

When the Unit 1 preoperational test procedure forms the basis for the Unit 2 acceptance test procedure, one review cycle will be used prior to submittal to the assistant plant startup manager for approval.

Subsequent revisions to the acceptance test procedure is handled in a manner similar to a new procedure.

Qualification of the above referenced plant personnel who review and approve test procedures is addressed in section 13.1.2.

The above process is further described in the VEGP startup manual.

14.2.2.3 Conduct of the Test Program

Georgia Power Company administers and conducts the initial test program using approved procedures. System turnover is made by the construction department to the nuclear operations department.

Upon acceptance of the turnover, administrative controls (detailed in the startup manual) become effective.

All phases of testing (with the exception of certain construction-performed tests) are under the control of the test supervisor.

Construction acceptance testing generally begins at the time of equipment/system turnover, release, or release for test. This testing must be completed or excepted (as provided for in the startup manual) prior to beginning the preoperational test. The procedure for development of construction acceptance testing procedures is provided in the startup manual.

Whenever practical, cleanliness verification testing will be performed, using approved procedures, prior to the initiation of fluid flow in the system during preoperational testing. Upon completion of construction acceptance testing, subject to the priorities of the startup schedule(s), the preoperational tests are performed.

The preoperational test phase will be governed by procedures in the startup manual. The startup test phase will be governed by plant procedures as supplemented by the Startup Test Administrative Procedures.

14.2.2.4 Review, Evaluation, and Approval of Test Results

14.2.2.4.1 Preoperational/Startup Test Procedure

Upon completion of a preoperational test and/or startup test, the test supervisor prepares a test results package and submits it for review.

The general manager-Vogtle nuclear operations (GMVNO) or his designee is responsible for the review and approval of preoperational test and startup test results. The general manager's designee for preoperational test results review and approval is the plant support manager; for startup tests results review and approval, his designee is the operations manager. In general, preoperational and startup test results are reviewed and approved by the same organizations and the same or higher level of authority that reviewed and approved the preoperational test procedure and startup test procedure. This includes a review of comments by the Georgia Power Company test supervisor, Bechtel Western Power Corporation representative, and/or Westinghouse Electric Corporation representative, as appropriate, prior to submittal of a preoperational and startup test results package to the appropriate review board. As a minimum, results of preoperational and startup tests listed in this chapter are submitted to the appropriate review board.

The test review board reviews the preoperational test results package and the plant review board reviews the startup test results package. Test results packages that are not satisfactory are resolved under direction of the board prior to recommendation of the package for approval. The test review board and plant review board recommend approval to the GMVNO or his designee of test results packages found acceptable. The GMVNO or his designee approves the test results package as complete. Completed test result packages become part of the permanent plant records.

Qualifications of the above referenced plant personnel who review and approve preoperational and startup test results is addressed in section 13.1.2.

The above process is further described in the VEGP Startup Manual and the Startup Test Administrative Procedures.

14.2.2.4.2 Acceptance Test (Unit 2 only)

Upon completion of an acceptance test procedure, the test supervisor prepares a test results package and submits it for review.

The general manager-Vogtle nuclear operations, or his designee is responsible for the review and approval of acceptance test results. The general manager designates the plant support manager or his designee for acceptance test results review and approval. In general, acceptance test results are reviewed and approved by the same organizations and same or higher level of authority that reviewed the procedure. Test results that are not satisfactory are resolved by the test engineering superintendent prior to submittal to the plant support manager for approval.

Qualification of the above referenced plant personnel who review and approve test results is addressed in section 13.1.2.

The above process is further described in the VEGP startup manual.

14.2.3 CONFORMANCE OF TEST PROGRAM WITH REGULATORY GUIDES (HISTORICAL)

[The initial test program for VEGP Units 1 and 2 is, in general, consistent with the requirements of Regulatory Guide 1.68. The following is a list of regulatory guides that are specifically associated with VEGP's initial test program and will be conformed with except as noted in FSAR section 1.9:

A. Regulatory Guide 1.18, Structural Acceptance Test for Concrete Primary Reactor Containments.

B. Regulatory Guide 1.20, Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing.

C. Regulatory Guide 1.30, Quality Assurance Requirements for the Installation, Inspection, and Testing of Instrumentation and Electric Equipment (Safety Guide 30).

D. Regulatory Guide 1.37, Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants.

E. Regulatory Guide 1.41, Preoperational Testing of Redundant Onsite Electric Power Systems to Verify Proper Load Group Assignments.

F. Regulatory Guide 1.52, Design, Testing, and Maintenance Criteria for Engineered Safety Features Atmosphere Cleanup System Air Filtration and Absorption Units of Light-Water-Cooled Nuclear Power Plants.

G. Regulatory Guide 1.68, Initial Test Programs for Water-Cooled Nuclear Power Plants.

H. Regulatory Guide 1.68.2, Initial Startup Test Program to Demonstrate Remote Shutdown Capability for Water-Cooled Nuclear Power Plant.

I. Regulatory Guide 1.68.3, Preoperational Testing of Instrument and Control Air Systems.

J. Regulatory Guide 1.79, Preoperational Testing of Emergency Core Cooling Systems for Pressurized Water Reactors.

K. Regulatory Guide 1.95, Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release.

L. Regulatory Guide 1.108, Periodic Testing of Diesel Generators Used as Onsite Electric Power Systems at Nuclear Power Plants.

M. Regulatory Guide 1.116, Quality Assurance Requirements for Installation, Inspection, and Testing of Mechanical Equipment and Systems.

N. Regulatory Guide 1.129, Maintenance Testing and Replacement of Large Lead Storage Batteries for Nuclear Power Plants.

O. Regulatory Guide 1.139, Guidance for Residual Heat Removal (For Comment).

P. Regulatory Guide 1.140, Design, Testing, and Maintenance Criteria for Normal Ventilation Exhaust System Air Filtration and Absorption Units of Light-Water-Cooled Nuclear Power Plant.]

14.2.4 UTILIZATION OF REACTOR OPERATING AND TESTING EXPERIENCE IN THE DEVELOPMENT OF THE TEST PROGRAM

The general manager-Vogtle nuclear operations (GMVNO) is responsible for ensuring that the VEGP initial test program will be developed utilizing information gained from operating and testing experiences in other similar nuclear plants.

Various documents (listed below) will be reviewed on a continuing basis to determine whether there is any information that should be included in VEGP's initial test program. Any pertinent information will be routed to the GMVNO or his designee for review by the appropriate startup personnel and, if necessary, for inclusion in the initial test program. Specific procedures are included in the plant procedures manual:

- Nuclear Regulatory Commission inspection and enforcement bulletins, notices, and circulars.

- Significant event report.
- Significant operating event report.
- Westinghouse data letters and technical bulletins.

The GMVNO also receives operating and testing information through the Westinghouse Owners Group, from the Institute of Nuclear Power Operations, and from Georgia Power Company employees who have been temporarily stationed at other nuclear plants that are conducting the initial test program.

14.2.5 TRIAL USE OF PLANT OPERATING AND EMERGENCY PROCEDURES

The plant operating procedures are utilized, where applicable, during the initial test program to support testing, maintain plant conditions, and facilitate training. The trial use of operating procedures serves to familiarize operating personnel with systems and plant operation during the testing phase and also serves to ensure the adequacy of the procedures under actual or simulated operating conditions. Emergency procedures are verified to the extent practical during the plant startup as plant conditions, testing, and training warrant; surveillance tests are performed as the test program permits to demonstrate the adequacy of the procedure.

VEGP will conform with the requirements of TMI action item I.G.1 of NUREG-0737. During low-power natural circulation testing, certain operators will be designated to conduct the test to gain hands-on experience in this mode of operation. Other operators will observe the conduct of the test as the startup and shift schedules permit.

14.2.6 INITIAL FUEL LOADING AND INITIAL CRITICALITY

Prior to the commencement of fuel loading, required preoperational test procedures are evaluated, and appropriate remedial action is taken if the acceptance criteria are not satisfied.

14.2.6.1 <u>Initial Fuel Loading</u>

The reactor containment structure will be completed, and the containment integrity established and maintained during fuel loading.

Fuel handling tools and equipment will have been checked out and dry runs conducted in the use and operation of the equipment.

The reactor vessel and associated components will be in a state of readiness to receive fuel. Water level will be maintained above the bottom of the nozzles and recirculation maintained to ensure a uniform boron concentration.

Plant management exercises the overall responsibility and direction for initial core loading. The overall process of initial core loading is, in general, directed from the operating floor of the containment structure. Standard procedures for the control of personnel and the maintenance of containment security are established prior to fuel loading.

The core is assembled in the reactor vessel and is submerged in reactor grade water containing enough dissolved boric acid to maintain a calculated core effective multiplication factor of 0.95 or lower. The refueling cavity may be dry during initial core loading. Core moderator chemistry conditions (particularly, boron concentration) are prescribed in the core loading procedure document and are verified periodically by chemical analysis of moderator samples taken prior to and during core loading operations.

Core loading instrumentation consists of two permanently installed source range (pulse type) nuclear channels and temporary incore pulse-type channels. The permanent channels, when responding, are monitored in the main control room by licensed plant operators; the temporary channels are installed in the containment structure and are monitored by test personnel. At least one permanent channel is equipped with an audible count rate indicator. Both plant channels have the capability of displaying the neutron flux level on a strip chart recorder. The temporary channels indicate on scalers, with a minimum of one channel recorded on a strip chart recorder.

Minimum count rates of at least 1/2 counts/sec, attributable to core neutrons, are required on at least two of the available pulse-type nuclear channels at all times following installation of the initial nucleus of fuel assemblies. A response check of nuclear instruments to a neutron source shall be performed within 8 h prior to loading of the core or upon resumption of loading if delay is for more than 8 h.

At least two artificial neutron sources are introduced into the core at appropriate specified points in the core loading program to ensure a detector response of at least 1/2 counts/sec attributable to neutrons.

Fuel assemblies together with inserted components (control rod assemblies, burnable poison inserts, source spider, or thimble plugging devices) are placed in the reactor vessel one at a time according to a previously established and approved sequence which was developed to provide reliable core monitoring with minimum possibility of core mechanical damage. The core loading procedure documents include detailed tabular check sheets which prescribe and verify the successive movements of each fuel assembly and its specified inserts from its initial position in the storage racks to its final position in the core. Multiple checks are made of component serial numbers and types at successive transfer points to guard against possible inadvertent exchanges or substitutions of components, and fuel assembly status boards are maintained throughout the core loading operation.

An initial nucleus of approximately eight fuel assemblies, one of which contains a neutron source, is the minimum source-fuel nucleus which permits subsequent meaningful inverse count-rate monitoring. This initial nucleus is determined by calculation to be markedly subcritical ($k_{eff} \leq 0.95$) under the required conditions of loading.

Each subsequent fuel addition is accompanied by detailed neutron count-rate monitoring to determine that the just-loaded fuel assembly does not excessively increase the count rate and that the extrapolated inverse count-rate ratio is not decreasing for unexplained reasons. The results of each step are evaluated before the next prescribed step is started. The final as-loaded core configuration is subcritical ($k_{eff} \leq 0.95$) under the required loading conditions.

Criteria for safe loading require that loading operations stop immediately if:

A. An unanticipated increase in the neutron count rate by a factor of two occurs on all responding nuclear channels during any single loading step after the initial nucleus of fuel assemblies is loaded.

B. An unanticipated increase in the count rate by a factor of five occurs on any individual responding nuclear channel during any single loading step after the initial nucleus of fuel assemblies is loaded.

C. A decrease in boron concentration greater than 20 ppm is determined from two successive samples of reactor coolant system water until the decrease is explained.

An alarm in the containment and main control room is coupled to the source range channels with a setpoint equal to or less than five times the current count rate; this alarm automatically alerts the loading operation personnel of the high count rate and requires an immediate stop of operations until the situation is evaluated. Normally, the alarm used for this purpose is the containment evacuation alarm. In the event that the evacuation alarm is actuated during core loading and the alarm is validated by a second responding detector, the containment is evacuated. After it has been determined that no hazards to personnel exist, preselected personnel are permitted to reenter the containment vessel to evaluate the cause and determine future action.

Core loading procedures specify the condition of fluid systems to prevent inadvertent dilution of the reactor coolant, specify the movement of fuel to preclude the possibility of mechanical damage, prescribe the conditions under which loading can proceed, identify responsibility and authority, and provide continuous and complete fuel and core component accountability.

14.2.6.2 Postloading Tests

Upon completion of core loading, the reactor upper internals and the pressure vessel head are installed, and additional mechanical and electrical tests are performed prior to initial criticality. The final pressure test is conducted after filling and venting is completed to check the integrity of the vessel head installation.

Mechanical and electrical tests are performed on the control rod drive mechanisms (CRDMs). These tests include a complete operational checkout of the mechanisms and calibration of the rod position indication system.

Tests are performed on the reactor trip circuits to test manual trip operation. The actual control rod assembly drop times are measured for each control rod assembly. The reactor control and protection system is checked with simulated signals to produce a trip signal for the various conditions that require plant trip.

At times that the CRDMs are being tested, the boron concentration in the coolant moderator is maintained such that the shutdown margin requirements specified in the Technical Specifications are met. During individual rod cluster control assembly or rod cluster control bank motion, source range instrumentation is monitored for unexpected changes in core reactivity.

A complete functional electrical and mechanical check is made of the incore nuclear flux mapping system at the operating temperature and pressure.

After final precritical tests, nuclear operation of the reactor begins. This final phase of startup and testing includes initial criticality, low-power testing, and power level escalation. The purpose of these tests is to establish the operational characteristics of the unit and core, to acquire data for the proper calibration of setpoints, and to ensure that operation is within license requirements. A brief description of the testing is presented in the following sections. Table 14.2.1-2 summarizes the tests which are performed from initial core loading to rated power.

Precritical tests shall be completed prior to initial criticality and the results evaluated. Prerequisites for performing a test are specified in the individual test procedure. The sequence of testing is outlined in a startup test sequence, such that required prerequisite testing is completed prior to performing subsequent testing. Any special test instruments required are specified to be installed, calibrated, and checked in the test procedure that specifies the test equipment. Where these test instruments are not left installed for future use, they are removed from the systems and removal is verified.

Initial criticality is achieved by a combination of shutdown and control bank withdrawal and reactor coolant system (RCS) boron concentration reduction. A predicted boron concentration for criticality will be determined for the precritical shutdown and control bank configuration specified in the initial criticality procedure. This configuration will require all banks to be fully withdrawn, with the exception of the last control bank, which will remain partially inserted to provide effective control when criticality is achieved.

Prior to beginning the approach to initial criticality, a neutron count rate of at least 1/2 count/s shall register on the startup channels, and the signal-to-noise ratio should be known to be greater than two. Core response during rod bank withdrawal and subsequent RCS boric acid concentration reduction will be monitored in the main control room by observing the change in neutron count rate as indicated by the permanent startup nuclear instrumentation.

Inverse count-rate ratio monitoring, using data from the normal plant source range instrumentation, is used as an indication of the proximity and rate of approach to criticality. Inverse count-rate ratio data are plotted as a function of rod bank position during rod motion and as a function of primary water addition during RCS boron concentration reduction.

Initially, the shutdown and control banks for control rods are withdrawn incrementally in the normal withdrawal sequence, leaving the last withdrawn control bank inserted as specified in the procedure.

The boron concentration in the RCS is then reduced. Criticality is achieved during boron dilution or by subsequent rod withdrawal following boron dilution. The rate of boron dilution, and hence the rate of approach to criticality, may be reduced as the reactor approaches criticality to ensure that effective control is maintained. At no time may a sustained startup rate of 1 decade/min be exceeded. Throughout this period, samples of the primary coolant are obtained and analyzed for boron concentration.

14.2.6.3 Low-Power Testing

Following initial criticality, a program of reactor physics measurements is undertaken to verify that the basic static and kinetic characteristics of the core are as expected, and that the values of the kinetic coefficients assumed in the safeguards analysis are conservative.

Procedures specify the sequence of tests and measurements to be conducted, and the conditions under which each is to be performed to ensure both safety of operation and the validity and consistency of the results obtained. If test results deviate significantly from design predictions, if unacceptable behavior is revealed, or if unexplained anomalies develop, the plant is brought to a safe, stable condition and the situation reviewed to determine the course of subsequent plant operation.

These measurements are made at low power and primarily at or near normal operating temperature and pressure. Measurements are made to verify the calculated values of control rod bank reactivity worths, the isothermal temperature coefficient under various core conditions, differential boron concentration reactivity worth, and critical boron concentration as functions of control rod configuration. In addition, measurements of the relative power distributions are made, and concurrent tests are conducted on the instrumentation, including source and intermediate range nuclear channels.

Gamma and neutron radiation surveys are performed at selected points throughout the plant. Periodic sampling is performed to verify chemical and radiochemical analysis of the reactor coolant.

14.2.6.4 Power Level Ascension

After the operating characteristics of the reactor have been verified by low-power testing, a program of power level ascension brings the unit to its full-rated power level in successive stages. At each successive stage, hold points are provided to evaluate and approve test results prior to proceeding to the next stage. The minimum test requirements for each successive stage of power ascension are specified in the applicable startup test procedures.

Measurements are made to determine the relative power distribution in the core as functions of power level and control assembly bank position.

Secondary system heat balance measurements ensure that the indications of power level are consistent and provide for calibration of the power range nuclear channels. The ability of the RCS to respond effectively to signals from primary and secondary instrumentation under a variety of conditions encountered in normal operations is verified.

At prescribed power levels, the dynamic response characteristics of the primary and secondary systems are evaluated. System response characteristics are measured for design step-load changes, rapid load reduction, and plant trips.

Adequacy of radiation shielding is verified by gamma and neutron radiation surveys at selected points throughout the plant at various power levels. Periodic sampling is performed to verify the chemical and radiochemical analysis of the reactor coolant.

14.2.7 TEST PROGRAM SCHEDULE

The sequential schedule for the preoperational and startup test program is provided in figure 14.2.7-1.

The general manager-Vogtle nuclear operations or his designee is responsible for the preparation, review, and maintenance of detailed schedules for testing.

Preoperational testing is scheduled to commence approximately 18 months prior to fuel loading. Included in the Unit 1 preoperational test schedule is the time needed to energize and test the plant's switchyard and associated equipment. The preoperational and construction acceptance tests are performed and sequenced during this period as a function of system turnover, system interrelationships, and acceptance for testing.

Startup testing is scheduled to be conducted over a period of approximately 4 to 6 months, commencing with fuel loading. The sequential schedule for startup testing ensures, insofar as practicable, that test requirements are completed prior to exceeding 25-percent power for all plant structures, systems, and components that are relied upon to prevent, limit, or mitigate the consequences of postulated accidents.

As shown in figure 14.2.7-1, there are 18 months between the fuel loading dates of Unit 1 and Unit 2. This results in Unit 2 test phases starting at least 18 months after the equivalent Unit 1 test phase. Also, Unit 1 is scheduled to complete fuel loading prior to construction acceptance testing and preoperational tests starting on Unit 2. This prevents any significant division of responsibility or dilution in the required personnel to conduct either unit's test program.

14.2.8 INDIVIDUAL TEST DESCRIPTIONS

The test abstracts provided are intended to show what systems/components will be tested and what minimum information will be verified. Information to be verified has been listed under the test abstract most likely to cover the required testing.

14.2.8.1 Preoperational Test Procedures

The following paragraphs give the test abstracts for each preoperational test listed in table 14.2.1-1.

14.2.8.1.1 Main Steam System Preoperational Test

A. Objective

To demonstrate the functional performance of the main steam system, including the following:

1. Acceptable closing times of the main steam isolation valves (MSIVs) and the MSIV bypass valves with steam lines at normal operating temperature and pressure.
2. Operability of the power-operated atmospheric relief valves with steam lines at normal operating temperature and pressure.
3. Operability of the main steam safety valves in a bench test at temperature and pressure with steam as the pressurizing fluid.

B. Prerequisites

The required portions of the following prerequisites are completed as necessary to support the preoperational test:

1. Construction acceptance testing completed.
2. Component testing and instrument calibration completed.
3. Test instrumentation available and calibrated.
4. Support systems available.

C. Test Method

1. Verify manual and automatic system controls.
2. The MSIV system's manual and automatic operation and MSIV and MSIV bypass valve closure times will be demonstrated during hot functional testing.
3. Demonstrate operability of main steam safety valves and seal leaktightness in a bench test at temperature and pressure with steam as the pressurizing fluid.
4. Demonstrate operability of the power-operated atmospheric relief valves during hot functional testing.
5. Verify alarms and status lights are functional.

D. Acceptance Criterion

The main steam system operates as described in section 10.3.

14.2.8.1.2 Steam Dump Preoperational Test

A. Objective

To demonstrate operation of the steam dump system.

B. Prerequisites

The required portions of the following prerequisites are completed, as necessary, to support the preoperational test:

1. Construction acceptance testing completed.
2. Component testing and instrument calibration completed.
3. Test instrumentation available and calibrated.
4. Support systems available.

C. Test Method

1. Verify manual and automatic system controls.
2. Verify alarms, indicating instruments, and status lights are functional.
3. Demonstrate dynamic operation of steam dump to condenser during hot functional testing.

D. Acceptance Criterion

The steam dump system operates as described in paragraph 7.7.1.8.

14.2.8.1.3 Condensate System Preoperational Test

A. Objectives

1. To demonstrate that the condensate pumps provide rated flow and pressure with operation of controls, alarms, and interlocks.
2. To demonstrate the automatic operation of system valves along with status lights and alarms.
3. To demonstrate the operation of the hotwell level control system.

B. Prerequisites

1. Required construction acceptance testing has been completed.
2. Required electrical power supplies and control circuits are operational.
3. The turbine plant closed cooling water system, the demineralized water system, and the condensate storage tank are available for use.
4. The feedwater system is available to receive flow from the condensate pump discharge header.

C. Test Method

1. Simulate pump overloads and discharge pressure signals to verify alarms and pump operation.
2. Operate pumps and verify pump performance characteristics.
3. Verify operation, status lights, and alarms of system valves and switches.

D. Acceptance Criteria

1. Condensate pumps, control valves, and isolation valves perform as described in subsection 10.4.7.
2. The hotwell level control system operates as described in subsection 10.4.1.

14.2.8.1.4 Main Feedwater System Preoperational Test

A. Objective

To demonstrate the proper operation of the feedwater control system valves, the steam generator feedwater pumps (SGFPs), manual and automatic function indications, and alarms.

B. Prerequisites

1. Required construction activities including component testing, instrument calibration, and system flushing/cleaning are complete.
2. Required test equipment has been calibrated and is operable.
3. Required ac and dc power is available.
4. Instrument air is available to air-operated valves.
5. The main turbine steam seal system is available to provide seal steam and packing exhaust for the SGFP turbines.
6. The main condenser is available to receive steam flow from the SGFP turbine.
7. The condensate system is available to supply suction and seal injection for the SGFPs.
8. Steam is available to the SGFP turbines.
9. The closed cooling water system is available to provide cooling water to the SGFP lube oil coolers.

C. Test Method

1. Manual and automatic operation of the feedwater system control valves (MFCVs and CBVs) to demonstrate the modes of operation, interlocks, response, failure mode indications, and alarms.
2. The SGFPs are operated; pump performance and flow and status indications are recorded. Simulated signals are used to verify low feedwater header discharge pressure alarm annunciation and low suction pressure alarm and trip functions.

3. Simulated signals are used to verify the operation of the condensate miscellaneous drain tank level alarm annunciation.

D. Acceptance Criterion

The main feedwater system operates as discussed in subsection 10.4.7.

14.2.8.1.5 Motor-Driven Auxiliary Feedwater System Preoperational Test

A. Objectives

1. To demonstrate the ability of the motor-driven auxiliary feedwater system to supply feedwater to the steam generators.
2. To verify operation of system valves and components.
3. To verify the operation of the condensate storage degasifiers.

B. Prerequisites

The required portions of the following prerequisites are completed, as necessary, to support the preoperational test:

1. Construction acceptance testing completed.
2. Component testing and instrument calibration completed.
3. Test instrumentation available and calibrated.
4. Support systems available.

C. Test Method

1. Verify manual and automatic system controls.
2. Verify system flowpaths, flowrates, and operating parameters within limits.
3. Verify pump and valve operating parameters.
4. Verify instruments, status lights, indications, and alarms.
5. Verify 48-h continuous operation and operation of the HVAC system for the auxiliary feedwater pumphouse (paragraph 14.2.8.1.35).
6. Verify 1-h operation on restart after cooldown.

D. Acceptance Criterion

The motor-driven auxiliary feedwater system operates within design limits, as described in subsections 10.4.7 and 10.4.9.

14.2.8.1.6 Turbine-Driven Auxiliary Feedwater System Preoperational Test

A. Objective

To demonstrate that the turbine-driven auxiliary feedwater system supplies feedwater to the steam generators.

B. Prerequisites

The required portions of the following prerequisites are completed as necessary to support the preoperational test:

1. Construction acceptance testing completed.
2. Component testing and instrument calibration completed.
3. Test instrumentation available and calibrated.
4. Support systems available.

C. Test Method

1. Verify manual and automatic system controls.
2. Verify system flowpaths, flowrates, and operating parameters within limits.
3. Verify turbine, pump, and valve operating parameters.
4. Verify that instruments, status lights, indications, and alarms are functional.
5. Verify 48-h continuous operation and operation of the HVAC system (paragraph 14.2.8.1.35).
6. Verify 1-h operation on restart after cooldown.
7. Verify cold quick start.
8. Verify start and 2-h operation with loss of ac power. This will include defeat of ventilation fan.

D. Acceptance Criterion

The turbine-driven auxiliary feedwater system operates within design limits, as described in subsection 10.4.9.

14.2.8.1.7 Reactor Coolant System (RCS) Preoperational Test

A. Objectives

1. To verify the control circuitry and operation of the system valves.
2. To demonstrate the operation of the head vent system.
3. To verify the operation of the boron analyzer.
4. To verify the operability of the pressurizer safety valves in a bench test at temperature and pressure with steam as the pressurizing fluid.

B. Prerequisite

Required component testing and instrument calibration are complete.

C. Test Method

1. The control circuitry and operation of system valves will be verified.
2. During a fill and vent of the RCS, the head vent will be operated.

3. The boron analyzer will be functionally tested using known concentrations of boric acid solutions.

4. The operability of the pressurizer safety valves will be demonstrated in a bench test at temperature and pressure with steam as the pressurizing fluid.

D. Acceptance Criteria

1. The valves respond to control signals. (See drawings 1X6AA02-225, 1X6AA02-226, 1X6AA02-227, 1X6AA02-228, 1X6AA02-229, 1X6AA02-230, 1X6AA02-231, 1X6AA02-232, 1X6AA02-233, 1X6AA02-234, 1X6AA02-235, 1X6AA02-236, 1X6AA02-237, 1X6AA02-238, 1X6AA02-239, 1X6AA02-240, 1X6AA02-494, 1X6AA02-495, 1X6AA02-496, and 1X6AA02-519.)

2. The readings of the boron analyzer are in accordance with the known boron concentration. (See table 7.7.1-2.)

3. The pressurizer safety valves actuate at a pressure of 2485 $\pm$ 25 psig.

14.2.8.1.8 Pressurizer Relief Tank (PRT) Preoperational Test

A. Objectives

1. To demonstrate that design PRT spray flow can be obtained against design backpressure in the tank.

2. To verify the operation of the PRT nitrogen pressurization system.

3. To demonstrate the filling and draining operation of the PRT.

4. To demonstrate a cooldown of the PRT using the reactor coolant drain tank (RCDT) heat exchanger.

5. To demonstrate the operation of system valve control circuitry.

B. Prerequisites

1. Required component testing and instrument calibration is complete.

2. The RCDT part of the liquid waste processing system is available to recirculate the PRT.

3. Reactor makeup is available to supply water to the PRT.

4. The nitrogen gas system is available to supply the PRT.

5. The plant is in hot functional testing for the PRT cooldown demonstration.

C. Test Method

1. With design backpressure in the PRT, the required spray flow is pumped to the PRT.

2. While draining and filling the PRT, the nitrogen pressurization system operation is observed.

3. During hot functional testing, the PRT is cooled down through the RCDT heat exchanger.

4. The control circuitry for system valves will be verified.

D. Acceptance Criteria

1. The required spray flow is obtained during design conditions. (See NSSS Startup Manual, SU-2.1.4.)

2. The pressurization system properly pressurizes and vents the PRT. (See NSSS Startup Manual, SU-2.1.4.)

3. The PRT can be cooled down by the RCDT heat exchanger. (See table 5.4.11-1.)

4. The system valves operate as designed. (See NSSS Startup Manual, SU-2.1.4.)

14.2.8.1.9 RCS Hydrostatic Preoperational Test

A. Objective

To verify the integrity and leaktightness of the RCS and the high-pressure portions of associated systems by performing a hydrostatic pressure test in conformance with Section III of the American Society of Mechanical Engineers (ASME) Code.

B. Prerequisites

1. Required component testing and instrument calibration are complete.

2. The reactor coolant pumps are available to support this test.

3. The positive displacement charging pump or a test pump is available.

4. The reactor vessel's lower and upper internals and the closure head are installed. The studs are tensioned to the design value for the associated hydrostatic test pressure.

5. Temporary temperature instrumentation is installed for measuring the temperature of the steam generator tube sheets, the bottom of the pressurizer, and the closure flange of the reactor vessel.

6. System relief valves and instrumentation that are within the test boundaries are either recalibrated or are verified to be able to withstand the hydrostatic test pressure.

7. Welds that are within the test boundaries have been verified as ready for hydrostatic testing.

C. Test Method

1. The system will be filled using the chemical and volume control system (CVCS) and residual heat removal system (RHRS).

2. The reactor coolant pumps will be operated, as required, to vent the system and to establish the required hydro temperature.

3. The system will then be pressurized to test pressure in stages; system welds, piping, and components will be monitored for leaks at each stage.

D. Acceptance Criterion

There shall be no leaks at welds or piping within the test boundaries during the final inspection. However, evidence of leaks at valves, flanges, or mechanical fittings does not invalidate the hydro. If there is evidence of leakage at a weld or piping, the leak must be repaired prior to the final inspection, or the leak may be isolated, repaired, and retested at a later date.

14.2.8.1.10 RCS Leak Rate Preoperational Test

A. Objectives

1. To determine during hot functional testing the amount of identified and unidentified leakage from the RCS and verify that the leakage is within allowable limits.
2. The operation of the containment particulate and radioactive gas monitoring portions of the leak detection system are verified in the process radiation monitoring system preoperational test.

B. Prerequisites

Hot functional testing is in progress for the determination of the amount of leakage.

C. Test Method

The identified and unidentified RCS leakage rates are determined by monitoring the system water inventory over a specified period of time.

D. Acceptance Criteria

1. The identified leakage rate is less than 10 gal/min.
2. The unidentified leakage rate is less than 1 gal/min.

14.2.8.1.11 Pressurizer Pressure and Level Control Preoperational Test

A. Objectives

1. To demonstrate the stability and response of the pressurizer pressure control system, including the verification of alarm and control functions.
2. To demonstrate the stability and response of the pressurizer level control system, including the verification of alarm and control functions.
3. To perform preliminary adjustment of the pressurizer continuous spray flow valves. Final adjustment of the continuous spray flow valves is performed during startup testing.
4. To demonstrate operation and valve response times for the pressurizer power-operated relief valves.
5. To demonstrate operation of the automatic block valves.

B. Prerequisites

1. Required component testing and instrument calibration are complete.
2. Required electrical power supplies and control circuits are energized and operational.

3. The plant is at normal operating temperature and pressure with reactor coolant pumps running, and hot functional testing is in progress.
4. The pressurizer continuous spray flow valves are full open.
5. The letdown and charging portions of the CVCS are available to vary pressurizer level.
6. The PRT is at a normal operating level and is aligned for normal operation.
7. The liquid waste processing system is available for cooldown of the PRT via the RCDT heat exchanger.

C. Test Method

1. Pressurizer pressure and level are varied separately, and the ability of the control system to automatically control and stabilize pressurizer pressure and level is verified.
2. Pressurizer pressure and level are varied separately, and the pressure and level alarm and control setpoints are verified.
3. Pressurizer continuous spray flow valve initial setting determination is made, and proper spray line temperatures are verified.
4. Pressurizer power-operated relief valves are operated, and response times are determined.
5. Pressurizer block valves are operated and their interlocks demonstrated.

D. Acceptance Criteria

1. The pressurizer pressure and level control systems respond to an increase or decrease in pressurizer pressure and level. (See drawings 1X6AA02-225, 1X6AA02-226, 1X6AA02-227, 1X6AA02-228, 1X6AA02-229, 1X6AA02-230, 1X6AA02-231, 1X6AA02-232, 1X6AA02-233, 1X6AA02-234, 1X6AA02-235, 1X6AA02-236, 1X6AA02-237, 1X6AA02-238, 1X6AA02-239, 1X6AA02-240, 1X6AA02-494, 1X6AA02-495, 1X6AA02-496, and 1X6AA02-519.)
2. Pressurizer pressure and level control system alarm and control functions have been operationally verified. (See drawings 1X6AA02-225, 1X6AA02-226, 1X6AA02-227, 1X6AA02-228, 1X6AA02-229, 1X6AA02-230, 1X6AA02-231, 1X6AA02-232, 1X6AA02-233, 1X6AA02-234, 1X6AA02-235, 1X6AA02-236, 1X6AA02-237, 1X6AA02-238, 1X6AA02-239, 1X6AA02-240, 1X6AA02-494, 1X6AA02-495, 1X6AA02-496, and 1X6AA02-519.)
3. Pressurizer continuous spray flow valve initial settings are determined, and spray line temperatures are maintained within design specifications (precautions, limitations, and setpoints).
4. Pressurizer power-operated relief valve operating times are within design specifications. (See NSSS Startup Manual, SU-2.1.2.)
5. Pressurizer power-operated relief block valve low pressure isolation and manual override are operable.

14.2.8.1.12 Reactor Coolant Pump Initial Operation Preoperational Test

A. Objectives

1. To measure and record system and operating parameters of the reactor coolant pumps during cold hydrostatic testing.
2. To measure and record pump operating parameters during hot functional testing.
3. To verify the operation of the associated oil lift pumps.

B. Prerequisites

1. Required component testing and instrument calibration are complete.
2. The CVCS is available to provide seal water to the reactor coolant pumps.
3. The auxiliary component cooling water (ACCW) system is available for cooling.
4. The RCS is filled and pressurized.

C. Test Method

1. The reactor coolant pumps and associated oil lift pumps will be operated during cold hydrostatic testing, and operating data will be recorded.
2. The reactor coolant pumps and associated oil lift pumps will be operated during hot functional testing, and operating data will be recorded at various temperature plateaus.

D. Acceptance Criterion

The reactor coolant pump and oil lift pump operating characteristics are within design specifications as described in subsection 5.4.1.

14.2.8.1.13 RCS Hot Functional Preoperational Test[(1)]

A. Objectives

1. To operate the RCS at full-flow conditions for a minimum of 240 h to provide the necessary vibration cycles on the reactor vessel's internal components prior to their final inspection before core loading. At least one-half of this operating time must occur with the reactor coolant temperature at or above 515°F.
2. To provide coordination and establishment of initial conditions necessary for the conduct of those preoperational tests to be performed during heatup, normal operating temperature and pressure, and cooldown of the RCS.
3. To provide the plant operators an opportunity to actually control the plant from an operations viewpoint and put into practice the plant operation

[1] See footnote 10 of table 14.2.1-1 for Unit 2 specific scope.

procedures for the RCS, auxiliary systems, and principal secondary systems prior to nuclear operation.

B. Prerequisites

1. Required component testing and instrument calibration are complete.
2. The reactor vessel's internals and head are installed.
3. All systems and components required to support heatup, operation at normal temperature and pressure, and cooldown of the RCS are available.
4. The diesel generators are operable and are ready for emergency power requirements.

C. Test Method

1. The plant is taken from cold shutdown conditions to hot standby for the first time prior to placing nuclear fuel in the reactor core.
2. The required energy input for this operation will be from operating reactor coolant pumps and the pressurizer heaters. The preoperational tests that require these hot and/or dynamic conditions will be conducted during hot functional testing as coordinated by this procedure.

D. Acceptance Criterion

1. The RCS has operated at full-flow conditions for a minimum of 240 h with at least half of it above 515°F.
2. The acceptance criteria for individual systems are a part of the individual test procedures sequenced by this procedure.

14.2.8.1.14 Reactor Internals and RCS Vibration Test

A. Objectives

1. To demonstrate that dynamic effects experienced during both steady flow and flow-induced transients will not cause structural damage or malfunction of the RCS, pressurizer surge line, and pressurizer relief line piping and components.
2. To perform a post-hot functional inspection of the reactor vessel and internals for vibration-induced damage and to document inspection results.

B. Prerequisites

1. Special test instrumentation for monitoring system and component vibration is installed at the required locations.
2. RCS hot functional testing is in progress.
3. The inspection of the reactor vessel and internals will be performed following cooldown from hot functional testing at full flow for greater than 240 h.

C. Test Method

1. The dynamic response of the RCS, pressurizer surge line, and pressurizer

pressurizer relief line piping and components is verified during both steady flow and flow-induced transients.

2. Following cooldown from hot functional testing, the reactor vessel and internals are inspected and the results documented.

D. Acceptance Criteria

1. The dynamic response of the RCS, pressurizer surge line, and pressurizer relief line piping and components is acceptable when compared with analytical results.

2. An inspection of the reactor vessel and internals, following 240 h of full-flow operation, has been performed, and inspection results are in accordance with drawing AX6DD401. (See paragraph 3.9.B.2.1.)

14.2.8.1.15 Resistance Temperature Detectors (RTDs)/Thermocouple Cross Calibration Preoperational Test

A. Objectives

1. To collect data to verify expected resistance versus temperature characteristics of installed RTDs and millivolt versus temperature characteristics of thermocouples.

2. To determine RTD deviation from RCS average temperature for individual RTDs and isothermal corrections for individual thermocouples.

3. To demonstrate that temperature instrumentation is reading correctly after disconnecting special test instrumentation and reconnecting RTDs to normal plant terminations.

B. Prerequisites

1. The reactor vessel upper internals are installed and thermocouple wiring is complete.

2. The temperature instruments have been aligned in accordance with manufacturer's calibration data.

3. Hot functional testing is in progress.

4. The special test instrumentation required for test performance is available and has been calibrated.

5. The plant computer is operational to obtain thermocouple data in the trend mode while taking RTD readings.

C. Test Method

1. RTD resistance data is taken at specified temperatures during hot functional testing, and the RTD resistance versus temperature characteristics is verified.

2. Thermocouple data is taken at the same temperature points used for recording RTD data, and the thermocouple millivolt versus temperature characteristics is verified.

3. RTD deviation from RCS average temperature and isothermal corrections for individual thermocouples are determined from the recorded data.

4. After disconnecting special test instrumentation and reconnecting RTDs to normal plant terminations, temperature data are taken to verify that all temperature instruments are reading correctly.

D. Acceptance Criteria

1. Test data are recorded for future alignment purposes only, and specific acceptance criteria are not provided. However, any individual RTD reading that differs from the calculated average temperature by a specified amount will not be used for average temperature calculations.

2. All temperature instruments disconnected for measurement indicate approximately reactor coolant system average temperature when their RTDs are reconnected to normal plant terminations.

14.2.8.1.16 RTD Bypass Flow Measurements

A. Objectives

1. To determine the flowrate necessary to achieve the design reactor coolant transport time in each RTD bypass loop.

2. To measure the flowrate in each RTD bypass loop and ensure that the transport times are acceptable.

B. Prerequisites

1. Required component testing and instrument calibration are complete.

2. The installed pipe length measurements are made with the plant cold before insulation is installed.

3. The bypass loop flow measurement will be performed during hot functional testing.

C. Test Method

1. The flowrate necessary to achieve the design reactor coolant transport time for each hot and cold leg bypass loop is calculated utilizing the piping length of each leg.

2. The hot and cold leg RTD bypass loop flow data are SU-2.1.9.)

D. Acceptance Criterion

The flowrate in each hot and cold leg RTD bypass loop is within design specifications. (See NSSS Startup Manual, SU-2.1.9.)

14.2.8.1.17 RHRS Preoperational Test

A. Objectives

1. To demonstrate operation of the residual heat removal (RHR) pumps and system valves and their associated control and interlock circuitry, including response to simulated safety signals.

2. To demonstrate RHR pump and system performance during discharge to the reactor coolant hot and cold loops and during RCS recirculation and to verify the head/flow characteristics of each installed pump.
3. To demonstrate RHR pump cold leg injection branch line orifice sizing and flowrates and pump runout flowrates.
4. To demonstrate RHR pump runout flowrates during reactor coolant hot and cold leg recirculation.
5. To demonstrate the time required for the RHR pumps to reach their rated flow.
6. To demonstrate RHRS operation during RCS heatup with letdown through the RHRS (to be performed during hot functional test).
7. To demonstrate RHRS operation during RCS cooldown (to be performed during hot functional test).
8. To demonstrate that each RHR pumproom cooler fan starts and stops when the associated RHR pump is started and stopped.
9. To demonstrate operation of the RHRS during filling and draining of the refueling cavity (to be performed during integrated control logic (safety injection) testing).

B. Prerequisites

1. Required component testing, instrument calibration, and system flushing/cleaning are complete.
2. Required electrical power supplies and control circuits are energized and operational.
3. The refueling water storage tank (RWST) contains an adequate supply of demineralized water for performance of the test.
4. The reactor vessel head is removed and provisions have been made to remove water from the vessel.
5. For the RCS recirculation portion of the test, RCS cleaning is complete, and vessel water level is above the nozzles.
6. Temporary suction strainers are installed for both RHR pumps.
7. The component cooling water (CCW) system is available to supply water to the RHR heat exchangers and the pump seal coolers.
8. The nuclear service cooling water (NSCW) system is available to supply cooling water to the RHR pump motors.

C. Test Method

1. System component control and interlock circuits are verified, including the operation of the RHR pumps and systems valves on receipt of simulated safety signals.
2. RHR pump and system performance characteristics are verified during discharge to the reactor coolant hot and cold loops and during RCS recirculation.

3. RHR pump cold leg injection branch line orifice sizing, branch line flowrates, and pump runout flowrates are verified during simulated cold leg injection.

4. RHR pump runout flowrates are verified during reactor coolant hot and cold leg recirculation.

5. RHRS operation is verified during RCS heatup and cooldown in conjunction with the RCS hot functional test.

6. Operation of the RHRS during filling and draining of refueling cavity is verified in conjunction with integrated control logic (safety injection) testing.

7. RHR pumproom cooler fan and RHR pump interlocking is verified during pump operation.

D. Acceptance Criteria

1. RHRS components respond properly to normal control and interlock signals and to simulated safety signals. (See NSSS Startup Manual, SU-2.4.1.)

2. RHR pump and system performance characteristics are within design specifications. (See NSSS Startup Manual, SU-2.3.3.)

3. RHR pump cold leg injection branch line orifice sizing, branch line flowrates, and RHR pump runout flowrates are within design specifications. (See NSSS Startup Manual, SU-2.3.3.)

4. RHR pump runout flowrates during reactor coolant hot and cold leg recirculation are within design specifications. (See NSSS Startup Manual, SU-2.3.3.)

5. The time required for each RHR pump to reach rated flow conditions is within design specifications. (See NSSS Startup Manual, SU-2.3.3.)

6. RHR pump and room cooler fan interlock operates properly. (See drawings 1X5DN030-1, 1X5DN030-3, 1X5DN030-4, 1X5DN030-5, and 1X5DN065-1.)

7. The RHR system functions as designed during RCS heatup and cooldown. (See NSSS Startup Manual, SU-2.4.1.)

8. The RHR functions as designed during refueling cavity filling and draining. (See paragraph 5.4.7.2.3.6.)

14.2.8.1.18 Chemical and Volume Control System

A. Objectives

1. To demonstrate the operation and verify the operating characteristics of the positive displacement charging pump.

2. To demonstrate the ability of the CVCS to supply adequate seal water injection flow to the reactor coolant pumps.

3. To verify operation of the control circuitry of the charging pumps and system valves including their response to safety signals.

4. To demonstrate that each room cooler fan starts and stops with its respective charging pump.
5. To demonstrate that the auxiliary spray will cool the pressurizer during cooldown of the RCS.
6. To demonstrate that the volume control tank level control system operates as designed.

B. Prerequisites

1. The required component testing and instrumentation calibration are complete.
2. The NSCW system is available to provide cooling water to the centrifugal charging pumps.
3. The ACCW system is available to provide cooling water to the positive displacement charging pump and the seal water heat exchanger.
4. Initial pump testing will be performed with the reactor head removed. Auxiliary spray cooldown of the pressurizer will be made during hot functional testing.

C. Test Methods

1. The centrifugal and positive displacement charging pumps will be operated, the appropriate room cooler interlocks will be verified, and performance characteristics of the positive displacement charging pump will be verified.

   Performance characteristics of the centrifugal charging pumps will be verified in the safety injection preoperational test.
2. Seal injection throttle valves will be adjusted to establish flow within the required tolerances for each reactor coolant pump.
3. During hot functional testing, a cooldown of the pressurizer using auxiliary spray will be demonstrated.
4. The control circuitry of the charging pumps and system valves, including their response to safety signals, will be verified.
5. Levels and pressure in the volume control tank will be varied and system response will be recorded.

D. Acceptance Criteria

1. The positive displacement charging pump operating characteristics are within design specifications. (See table 9.3.4-2.)
2. Seal water injection flow to each reactor coolant pump has been established in accordance with the limits imposed in NSSS Startup Manual 2.3.3.
3. System valve and pump control circuits and interlocks operate as designed. (See drawings 1X6AA02-225, 1X6AA02-226, 1X6AA02-227, 1X6AA02-228, 1X6AA02-229, 1X6AA02-230, 1X6AA02-231, 1X6AA02-232, 1X6AA02-233, 1X6AA02-234, 1X6AA02-235, 1X6AA02-236, 1X6AA02-237, 1X6AA02-238, 1X6AA02-239, 1X6AA02-240, 1X6AA02-494, 1X6AA02-495, 1X6AA02-496, and 1X6AA02-519.)

14.2.8.1.19 Letdown System Preoperational Test

A. Objectives

1. To verify operation of the normal and excess letdown systems.
2. To verify operation of the system valves and control circuitry including the response to safety signals.
3. To demonstrate the ability of the charging and letdown system to maintain solid RCS pressure control.

B. Prerequisites

1. Required component testing and instrument calibration are complete.
2. The system tests will be performed during hot functional testing.

C. Test Method

1. Control circuitry for system valves including the response to safety signals will be verified.
2. The system will be operated during hot functional testing, and system performance characteristics will be recorded.
3. During hot functional test heatup, with the system solid, the charging and letdown system pressure control will be demonstrated by varying the charging and letdown flows and observing the system response.

D. Acceptance Criteria

1. System valves and interlocks operate as designed. (See NSSS Startup Manual SU-2.2.1; DC-1208; and FSAR table 6.2.4-1.)
2. System operating characteristics are within the design limitations (tables 9.3.4-1 and 9.3.4-2).

14.2.8.1.20 Boron Thermal Regeneration System Preoperational Test (See table 14.2.1-1 for Unit 2.)

A. Objectives

1. To verify the operation of the boron thermal regeneration system, including the chiller units, pumps, and other system components.
2. To demonstrate that the system will heat and cool the letdown as designed.

B. Prerequisites

1. Required component testing and instrument calibration are complete.
2. Portions of this test involving the heating and cooling cycles will be performed during hot functional testing.

C. Test Method

1. Control circuitry and operation of system valves and motors will be verified.

2. The system will be operated during hot functional testing, and system performance characteristics will be recorded.

D. Acceptance Criteria

1. System valves, motors, and interlocks operate as designed. (See NSSS Startup Manual, SU-2.2.3.)
2. The system functions to heat and cool the letdown to within design limitations. (See NSSS Startup Manual, SU-2.2.3.)

14.2.8.1.21 Boric Acid Blending Operation

A. Objectives

1. To verify the operation of the boric acid transfer pumps and the ability of the boric acid blending system to make up at design flowrates and boric acid concentrations to the CVCS in each mode of operation.
2. To verify the operation of system valves and control circuitry, including the response to safety signals.
3. To demonstrate the heating system for the boric acid batch tank.
4. To demonstrate the ability of the boric acid system to supply an emergency boration flow to the charging pump suction.

B. Prerequisites

1. The required component testing and instrument calibration are complete.
2. The reactor makeup water system is available to supply water to the boric acid blending tee and batch tank.
3. Auxiliary steam is available to heat the batch tank as required.
4. A charging pump must be available for operation.
5. Parts of the system that contain boric acid at 4-percent concentration will have to be kept at least at 65°F.

C. Test Method

1. The boric acid pumps will be operated, and performance characteristics will be verified.
2. The control circuitry for system valves and motors will be verified.
3. The makeup and blending system controls will be demonstrated in all modes of operation: auto makeup, dilute, alternate dilute, borate, and manual.
4. Emergency boration to charging pump suction operation will be demonstrated.
5. Steam heating of the boric acid batch tank will be demonstrated.

D. Acceptance Criteria

1. The boric acid transfer pump operating characteristics are within the design specifications. (See table 9.3.4-2.)

2. System valves and interlocks operate as designed. (See NSSS Startup Manual, SU-2.2.2.)
3. The flowrates and boric acid concentrations associated with the blender are within the design specifications for each operating mode. (See NSSS Startup Manual, SU-2.2.1.)
4. The emergency boration flowrate to the charging pumps is within the design specifications. (See NSSS Startup Manual, SU-2.2.2.)

14.2.8.1.22 Safety Injection System (SIS) Preoperational Test

A. Objectives

1. To demonstrate centrifugal charging pump performance and to verify that the head/flow characteristics of each installed pump are approximately the same.
2. To demonstrate operation of the safety injection pumps and system valves and their associated control circuitry including response to safety signals.
3. To demonstrate safety injection pump performance and to verify that the head/flow characteristics of each installed pump are approximately the same.
4. To demonstrate miniflowrate for the centrifugal charging pumps and safety injection pumps.
5. To position the centrifugal charging pump cold leg injection line throttling valves and to demonstrate injection branch line flow balancing and pump runout flowrate.
6. To position the safety injection pump cold and hot leg injection line throttling valves and to demonstrate injection branch line flow balancing and pump runout flowrates.
7. To demonstrate the time required for the centrifugal charging and safety injection pumps to reach their rated flow.
8. To demonstrate operation of the boron injection tank outlet motor-operated valves under maximum differential pressure conditions.
9. To demonstrate that each safety injection pump room cooler fan starts and stops when the associated safety injection pump is started and stopped.

B. Prerequisites

1. Required component testing, instrument calibration, and system flushing/cleaning are complete.
2. Required electrical power supplies and control circuits are energized and operational.
3. The RWST contains an adequate supply of demineralized water for performance of the test.

4. The reactor vessel head is removed and provisions have been made to remove water from the vessel.

5. Temporary suction strainers are installed for both safety injection pumps.

6. The NSCW system is available to supply cooling water to the centrifugal charging pump and safety injection pump motors and oil coolers.

7. The ACCW system is available to supply cooling water to the CVCS seal water heat exchanger.

C. Test Method

1. System component control and interlock circuits, including the operation of the safety injection pumps and system valves on receipt of simulated safety signals, are verified.

2. Centrifugal charging pump and safety injection pump performance characteristics are verified during simulated reactor coolant cold leg injection.

3. The centrifugal charging pump cold leg injection line throttling valves are properly positioned, and injection branch line flows and pump runout flowrates are verified.

4. The safety injection pump cold and hot leg injection line throttling valves are properly positioned, and injection branch line flows and pump runout flowrates are verified.

5. The boron injection tank outlet valves operated under conditions of maximum differential pressure, and operation is verified.

6. Safety injection pump room cooler fan and safety injection pump interlocking is verified during pump operation.

D. Acceptance Criteria

1. SIS components respond to normal control and interlock signals and to simulated safety signals. (See figure 6.3.2-1.)

2. The centrifugal charging pumps, safety injection pumps, and their associated system performance characteristics are within design specifications. (See NSSS Startup Manual, SU-2.3.3.)

3. Centrifugal charging pump and safety injection miniflowrates are within design specifications. (See NSSS Startup Manual, SU-2.3.3.)

4. The centrifugal charging pump cold leg injection line throttling valves are positioned to provide branch line flows and pump runout flowrates within design specifications. (See NSSS Startup Manual, SU-2.3.3.)

5. The safety injection pump cold and hot leg injection line throttling valves are positioned to provide branch line flows and pump runout flowrates within design specifications. (See NSSS Startup Manual, SU-2.3.3.)

6. The time required for the centrifugal charging pumps and safety injection pumps to reach rated flow conditions is within design specifications. (See NSSS Startup Manual, SU-2.3.3.)

7. Boron injection tank outlet valve response times are within design specifications. (See table 6.3.2-1.)

8. Safety injection pump and room cooler fan interlock operates. (See drawings 1X5DN030-1, 1X5DN030-3, 1X5DN030-4, 1X5DN030-5 and 1X5DN065-1.)

14.2.8.1.23 Safety Injection Check Valve Preoperational Test

A. Objectives

1. To demonstrate the operability of the various safeguard systems injection line check valves which are expected to be at higher than ambient temperature when the RCS is at normal operating temperature.
2. To demonstrate the integrity of the accumulator and safeguard systems injection line check valves by performing a backleakage check.
3. To demonstrate that each accumulator motor-operated isolation valve opens on reset of the pressurizer pressure (P-11) manual block.

B. Prerequisites

1. Required component testing, preoperational testing, and instrument calibrations are complete.
2. Required electrical power supplies and control circuits are energized and operational.
3. RCS pressure and temperature are established near the normal operating values, and hot functional testing is in progress.

C. Test Method

1. Flow from a safety injection pump is directed through the SIS test line to the RCS, and the accumulator and injection line check valve operability is verified.
2. SIS test line valves are aligned to provide a leakoff path from each of the accumulator and injection line primary and backup check valves, and valve backleakage is verified.
3. Accumulator motor-operated isolation valve opening is verified on reset of the pressurizer pressure (P-11) manual block.

D. Acceptance Criteria

1. The accumulator and injection line check valves operate as demonstrated by verification of flow through the check valves.
2. Check valve integrity test results (measured backleakage flowrate) conform to technical specification requirements (table 16.3.4-1).
3. Accumulator motor-operated isolation valves open automatically on pressurizer pressure (P-11) manual block.

14.2.8.1.24 Safety Injection Accumulator Testing

A. Objectives

1. To demonstrate operation of the safety injection accumulators and system valves and their associated control and interlock circuitry.

2. To demonstrate accumulator injection and obtain blowdown rate data during a low-pressure blowdown of each accumulator.

B. Prerequisites

1. Required component testing, instrument calibration, and system flushing/cleaning are complete.

2. Required electrical power supplies and control circuits are energized and operational.

3. The reactor vessel head and reactor internals have been removed, and provisions have been made to remove water from the vessel during the test.

4. The nitrogen system is available to supply nitrogen to the accumulators.

5. The RWST contains an adequate supply of demineralized water for the performance of this test.

6. The SIS is available for filling the accumulators.

C. Test Method

1. Accumulator system component control and interlock circuits, including their response to simulated safety injection and containment isolation signals, are verified.

2. Each accumulator is filled and partially pressurized with the discharge valves closed. The discharge valves are opened, discharging the accumulators to the reactor vessel, and accumulator blowdown rate data are determined.

3. Each accumulator discharge line isolation valve is operated under maximum differential pressure conditions of normal accumulator precharge pressure and zero reactor coolant pressure.

D. Acceptance Criteria

1. Safety injection accumulator system components respond to normal control and interlock signals and to simulated safety injection and containment isolation signals. (See drawings 1X4DB119, 1X4DB120, and 1X4DB121.)

2. Safety injection accumulator system valve response times are within design specifications. (See table 6.3.2-1.)

14.2.8.1.25 Boron Injection Tank and Recirculation Pump Test

This test has been deleted.

14.2.8.1.26 Containment Spray System Preoperational Test

A. Objectives

1. To demonstrate that the spray nozzles in the containment spray headers are clear of obstructions.
2. To demonstrate operation of the containment spray pumps and system valves and their associated control circuitry.
3. To demonstrate containment spray pump and system performance during operation in the test and simulated accident modes.
4. To demonstrate proper pump performance with simultaneous operation of two containment spray pumps, two centrifugal charging pumps, two safety injection pumps, and two RHR pumps.
5. To demonstrate that each containment spray pump room cooler fan starts and stops when the associated containment spray pump is started and stopped.
6. To demonstrate eductor performances by measuring spray additive flowrates.

B. Prerequisites

1. Required component testing, instrument calibration, and system flushing/cleaning are complete.
2. Required electrical power supplies and control circuits are energized and operational.
3. The RWST contains an adequate supply of demineralized water for the performance of this test.
4. Temporary suction strainers are installed for both containment spray pumps.
5. Temporary piping is installed from the spray header test connections.
6. A source of compressed air is available to pressurize the spray headers.
7. The NSCW system is available to supply cooling water to the containment spray pump motors.

C. Test Method

1. System component control circuits, including the operation of the containment spray pumps and system valves on receipt of simulated safety signals, are verified.
2. Containment spray pumps and system performance characteristics are verified during operation in the test and simulated accident modes.
3. Spray additive eductor operating characteristics are verified during system operation.
4. Airflow is initiated through the containment spray headers, and unobstructed flow is verified through each nozzle.
5. Containment spray pumproom cooler fan and containment spray pump interlocking is verified during system operation.

6. The containment spray pumps, centrifugal charging pumps, safety injection pumps, and RHR pumps are operated simultaneously, and adequate suction flow is verified.

D. Acceptance Criteria

1. Containment spray system components respond to normal control signals and to simulated containment spray actuation load, load shed, and load sequencing signals. (See drawings 1X3D-BD-J01A, 1X3D-BD-J01B, 1X3D-BD-J02A, 1X3D-BD-J02B, 1X3D-BD-J02C, and 1X3D-BD-J02D.)
2. Containment spray pump and system performance characteristics are within design specifications. (See NSSS Startup Manual, SU-4.1.8.)
3. Spray additive eductor operating characteristics are within design specifications. (See FSAR 6.2.2-4.)
4. All containment spray nozzles are unobstructed, as evidenced by air passing through each nozzle. (See NSSS Startup Manual, SU-4.1.8.)
5. Containment spray pump and room cooler fan interlocking operates properly. (See drawings 1X5DN030-1, 1X5DN030-3, 1X5DN030-4, 1X5DN030-5, and 1X5DN065-1.)
6. Pump pressures, flow, and available net positive suction head requirements are met for the containment spray pumps, centrifugal charging pumps, safety injection pumps, and RHR pumps when all eight pumps are run simultaneously. (See NSSS Startup Manual 2.3.3.)

14.2.8.1.27 Reactor Makeup Water Storage Tank and Degasifier System Preoperational Test

A. Objective

To demonstrate operation of the reactor makeup water storage tank and degasifier system.

B. Prerequisites

The required portions of the following prerequisites are completed as necessary to support the preoperational test:

1. Construction acceptance testing completed.
2. Component testing and instrument calibration completed.
3. Test instrumentation available and calibrated.
4. Support systems available.

C. Test Method

1. Verify manual and automatic system controls.
2. Verify system flowrates.

D. Acceptance Criterion

The reactor makeup water storage tank and degasifier system operates as described in subsection 9.2.7.

14.2.8.1.28 Refueling Water Storage Tank

A. Objectives

1. To demonstrate operation of refueling water storage and sludge mixing/heating system components and their associated control and interlock circuitry.
2. To demonstrate sludge mixing pump performance during circulation operation.
3. To demonstrate operation of the electric circulation heater.

B. Prerequisites

1. Required component testing and instrument calibration are complete.
2. Required electrical power supplies and control circuits are energized and operational.
3. The demineralized water system is available for filling the RWST.

C. Test Method

1. System component control circuits, including response to normal control, interlock, and alarm signals, are verified.
2. The sludge mixing/heating system is operated to circulate the tank contents, and system performance characteristics are verified.
3. The electric circulation heater capacity is verified by measuring heater voltage and amperage during operation.

D. Acceptance Criteria

1. The RWST and sludge mixing/heating system components respond properly to normal control, interlock, and alarm signals.
2. The sludge mixing system performance characteristics are within system design specifications.
3. The electric circulation heater capacity is within design specifications.

14.2.8.1.29 Condenser Air Ejection System Preoperational Test

A. Objectives

To demonstrate operation of each steam jet air ejector (SJAE), vacuum pump, motor-actuated valve, and the condenser air ejection portion of the turbine building heating, ventilation, and air-conditioning (HVAC) system.

B. Prerequisites

1. Required construction acceptance testing is complete.
2. Permanently installed instrumentation necessary to support the test has been properly calibrated and is operable.
3. The turbine building HVAC system is available.
4. Appropriate ac and dc power sources are available.

5. The condensate and feedwater system, auxiliary steam system, and turbine plant cooling water system are available.

6. Turbine is on turning gears.

C. Test Method

1. Achieve and maintain a condenser vacuum to verify that:

   a. Mechanical vacuum pumps operate in accordance with design.

   b. The SJAEs operate in accordance with design.

2. Operate mechanical equipment in the system to verify operation.

3. Simulate a high radiation level and verify operation of the condenser air ejection filtration dampers.

D. Acceptance Criterion

Equipment listed in the objectives performs as described in subsections 9.4.4 and 10.4.2.

14.2.8.1.30 Circulating Water System Preoperational Test

A. Objectives

To demonstrate operation of the circulating water pumps, the system's associated motor-actuated valves, and the turbine plant cooling tower blowdown valve.

B. Prerequisites

1. Required construction acceptance testing has been completed.

2. Required permanently installed instrumentation has been properly calibrated and is operable.

3. Appropriate ac and dc power sources are available.

4. The following support systems are available:

   a. Instrument and service air systems.

   b. River makeup water pump system.

   c. Traveling water screen and wash system.

   d. Utility water system.

   e. Turbine plant cooling water system.

C. Test Method

1. Check circulating water pump controls.

2. Check motor-, control-, and air-operated valves.

3. Simulate high circulating water conductivity, and verify operation of the turbine plant cooling tower blowdown valves.

4. Test circulating water pumps and verify conformance to their design specifications.

D. Acceptance Criterion

Equipment associated with the circulating water system performs as described in subsection 10.4.5.

14.2.8.1.31 Spent Fuel Pool Cooling and Cleanup System (SFPCCS) Preoperational Test

A. Objectives

1. To demonstrate operation of the SFPCCS components and their associated control circuitry.
2. To demonstrate spent fuel pool cooling pump, purification pump, skimmer pump, and system performance during operation in their various modes.
3. To demonstrate that the spent fuel pool can be filled from the demineralized water system, RWST, or the reactor makeup water system.
4. To demonstrate that the antisiphon design prevents gravity drainage of the spent fuel pool.
5. To demonstrate the operability of the low level water alarm.
6. To verify the operability of the gates and drains and to perform leak tests of the gates, drains, and gaskets in the refueling canal and fuel storage pool.

B. Prerequisites

1. Required component testing, instrument calibration, and system flushing/cleaning are complete.
2. Required electrical power supplies and control circuits are energized and operational.
3. The CCW system is available to provide cooling water to the spent fuel pool cooling heat exchangers.
4. The RWST contains an adequate supply of demineralized water for the performance of this test.
5. The demineralized water system and reactor makeup water system are both available to provide fill water to the spent fuel pool.
6. The spent fuel pool gate has been installed to keep the fuel transfer canal dry, and the fuel transfer tube valve is closed.
7. The spent fuel pool is filled to the normal level with demineralized water.

C. Test Method

1. System component control circuits, including response of the spent fuel pool cooling pumps to simulated safety signals, are verified.
2. The spent fuel pool cooling, spent fuel pool purification, and spent fuel pool skimmer pumps are operated in their various modes, and the pump and system performance characteristics are verified.

3. The ability to partially fill the spent fuel pool from the RWST, demineralized water system, and reactor makeup water system is verified.

4. The spent fuel pool cooling system piping is altered to allow gravity draining of the spent fuel pool by siphon effect. The minimum level to which the spent fuel pool can be drained is verified.

5. The gates, drains, and gaskets in the refueling canal and fuel storage pool are checked for unacceptable leakage.

D. Acceptance Criteria

1. The SFPCCS components respond properly to normal control signals and to simulated load shed signals. (See drawing 1X4DB130.)

2. The spent fuel pool cooling pumps, purification pumps, skimmer pumps, and system performance characteristics are within design specifications. (See table 9.1.3-1.)

3. The spent fuel pool can be filled from the RWST, demineralized water system, or the reactor makeup water system. (See Design Basis DC-1213.)

4. The spent fuel cooling system antisiphon design prevents draining the spent fuel pool below an acceptable minimum level. (See Design Basis DC-1213.)

14.2.8.1.32 ACCW Preoperational Test

A. Objectives

1. To demonstrate the capability of the ACCW system to provide cooling water flow during shutdown with loss of offsite power and containment isolation modes of operation.

2. To verify operation of system valves and control circuitry and pump control circuitry, including the response to safety signals.

3. To demonstrate the operation and verify the operating characteristics of the ACCW pumps.

B. Prerequisites

1. Required component testing and instrument calibration are complete.

2. Demineralized water or reactor makeup water is available for system makeup.

3. NSCW is available.

C. Test Method

1. The control circuits of the ACCW pumps and valves will be verified.

2. The ACCW pumps will be operated, and performance characteristics will be verified.

3. The system flows will be verified in the shutdown with loss of power and containment isolation modes of operation.

D. Acceptance Criteria

1. The performance characteristics of the ACCW pumps are as designed. (See table 9.2.8.2.)
2. The pump and valve controls and interlocks operate as designed. (See drawings 1X4DB138-1, 1X4DB138-2 and 1X4DB139.)
3. Components that are supplied with ACCW receive flows that are within the design specifications in each operating mode. (See table 9.2.8-1.)

14.2.8.1.33 NSCW System Preoperational Test

A. Objective

To demonstrate operation of the NSCW system.

B. Prerequisites

The required portions of the following prerequisites are completed, as necessary, to support the preoperational test:

1. Construction acceptance testing completed.
2. Component testing and instrument calibration completed.
3. Test instrumentation available and calibrated.
4. Support systems available.

C. Test Method

1. Verify manual and automatic system controls.
2. Verify system flowrates.
3. Verify alarms and status lights are functional.
4. Verify adequate NPSH for pump operation at tower basin minimum level and maximum design temperature.

D. Acceptance Criterion

The NSCW system operates within design limits, as described in subsection 9.2.1.

14.2.8.1.34 Component Cooling Water Preoperational Test

A. Objectives

1. To demonstrate the capability of the CCW system to provide cooling water during normal operation, normal cooldown, and postulated loss-of-coolant accident (LOCA) modes of operation.
2. To verify operation of system valves and control circuitry.
3. To demonstrate the operation and verify the operating characteristics of the CCW pumps, including their response to safety signals.

B. Prerequisites

1. Required component testing and instrument calibration are complete.
2. Demineralized water or reactor makeup water is available for system makeup.
3. NSCW is available to cool the CCW motors.

C. Test Method

1. The control circuitry of the CCW pumps and valves will be verified.
2. The CCW system pumps will be operated, and performance characteristics will be verified.
3. System flows will be verified in each mode of operation.

D. Acceptance Criteria

1. The performance characteristics of the CCW pumps are within design specifications. (See table 9.2.2-1.)
2. Components that are supplied with CCW receive flows that are within the design specifications in each of the operating modes. (See table 9.2.2-2.)
3. The pump control and interlocks operate as designed. (See drawings 1X5DN091-1, 1X5DN091-2, 1X5DN091-3, 1X5DN092-1, and 1X5DN092-2.)

14.2.8.1.35 Auxiliary Feedwater Pumphouse HVAC System Preoperational Test

A. Objective

To demonstrate operation of the auxiliary feedwater pumphouse HVAC system.

B. Prerequisites

The required portions of the following prerequisites are completed, as necessary, to support the preoperational test:

1. Construction acceptance testing is complete.
2. Component testing and instrument calibration are complete.
3. Test instrumentation is available and calibrated.
4. Support systems are available.
5. The turbine- and motor-driven auxiliary feedwater pumps are available for operation.

C. Test Method

1. Verify manual and automatic system controls.
2. Verify alarms, and status lights are functional.
3. Verify design airflow.
4. Each engineered safety feature (ESF) fan will be shown to meet design requirements with the motor-driven pump running and without assistance from any non ESF equipment.

5. The non-ESF fan will be tested with the turbine-driven pump operating.

D. Acceptance Criterion

1. The auxiliary feedwater pumphouse HVAC system operates as described in subsection 9.4.8.
2. The ESF fan maintains the motor-driven pump room at or below 120°F (9.4.8.2.3).
3. The non-ESF fan maintains the turbine-driven pump room at or below 120°F (9.4.8.2.3).

14.2.8.1.36 Fuel Handling Building HVAC System Preoperational Test

A. Objective

To demonstrate operation of the fuel handling building HVAC system.

B. Prerequisites

The required portions of the following prerequisites are completed, as necessary, to support the preoperational test:

1. Construction acceptance testing is complete.
2. Component testing and instrument calibration are complete.
3. Test instrumentation is available and calibrated.
4. Support systems are available.

C. Test Method

1. Verify manual and automatic system controls.
2. Verify alarms, indicating instruments, and status lights are functional.
3. Verify design airflow.
4. Demonstrate ability of fuel handling building exhaust system to maintain a negative pressure in the fuel handling building.
5. Verify normal exhaust system isolation and post-accident exhaust system initiation on simulated high radiation signal from the exhaust ductwork radiation instrumentation.

D. Acceptance Criterion

The fuel handling building HVAC system operates as described in subsection 9.4.2.

14.2.8.1.37 (Deleted)

14.2.8.1.38 Essential Chilled Water System Preoperational Test

A. Objective

To demonstrate operation of the essential chilled water system.

B. Prerequisites

The required portions of the following prerequisites are completed as necessary to support the preoperational test:

1. Construction acceptance testing is complete.
2. Component testing and instrument calibration are complete.
3. Test instrumentation is available and calibrated.
4. Support systems are available.

C. Test Method

1. Verify manual and automatic system controls.
2. Verify alarms and status lights are functional.
3. Verify system flowrates.

D. Acceptance Criterion

The essential chilled water system operates as described in subsection 9.2.9.

14.2.8.1.39 Control Building HVAC Preoperational Test

A. Objective

To demonstrate operation of the control building HVAC system, including electrical tunnel ventilation and steam tunnel ventilation systems.

B. Prerequisites

The required portions of the following prerequisites are completed, as necessary, to support the preoperational test:

1. Construction acceptance testing is complete.
2. Component testing and instrument calibration are complete.
3. Test instrumentation is available and calibrated.
4. Support systems are available.

C. Test Method

1. Verify manual and automatic controls in the normal and emergency operating modes.
2. Verify alarms and status lights are functional.
3. Verify design airflow.
4. Verify control building isolation on simulated signal from the intake ductwork radiation instrumentation.

D. Acceptance Criterion

The control building HVAC system operates as described in section 6.4, subsections 9.4.1 and 9.4.5, and paragraph 9.4.9.2.

14.2.8.1.40 Auxiliary Building HVAC System Preoperational Test

A. Objective

To demonstrate operation of the auxiliary building HVAC system.

B. Prerequisites

The required portions of the following prerequisites are completed, as necessary, to support the preoperational test:

1. Construction acceptance testing is complete.
2. Component testing and instrument calibration are complete.
3. Test instrumentation is available and calibrated.
4. Support systems are available.

C. Test Method

1. Verify manual and automatic controls in the normal and emergency operating modes.
2. Verify alarms and status lights are functional.
3. Verify design airflow.
4. Verify auxiliary building isolation on simulated containment ventilation isolation signal.

D. Acceptance Criterion

The auxiliary building HVAC system operates as described in subsection 9.4.3.

14.2.8.1.41 Diesel Generator Building HVAC Preoperational Test

A. Objective

To demonstrate operation of the engineered safety features (ESF) fans, non-ESF fans, non-ESF unit heaters, automatic dampers, fuel oil day tank exhaust fans, and all alarms and status indication.

B. Prerequisites

1. Required construction acceptance testing is complete.
2. Required system instrumentation and test equipment is available and calibrated.
3. Required diesel generator building HVAC system air handling ductwork has been tested and balanced as necessary.
4. Required ac and dc power is available.

C. Test Method

1. Cycle the thermostat for each non-ESF unit heater and fan and verify operation.
2. Simulate start and interlock signals for each ESF fan (with its breaker in the test position) and verify operation and annunciation.

3. Simulate high and low room air temperature signals and verify alarm annunciation.

4. Operate diesel fuel oil day tank exhaust fans and verify operation and indication.

5. Verify design airflow.

6. Operate dampers under simulated normal and emergency conditions and verify operation and indication.

D. Acceptance Criteria

1. Each of the diesel generator building train A and train B unit heaters automatically start with a signal from their respective temperature controller.

2. Diesel generator building train A and train B ESF supply fans maintain respective room temperature at 120°F or below with the diesel generator running.

3. All high and low air temperature alarms annunciate properly.

4. Diesel generator building train A and train B ESF supply fans operate on the proper signals.

5. The diesel generator building train A and train B non-ESF exhaust fans operate on the proper signals.

6. The train A and train B automatic dampers operate properly in normal and emergency operation.

14.2.8.1.42 Containment Heat Removal System Preoperational Test

A. Objective

To demonstrate operation of the containment heat removal system.

B. Prerequisites

The required portions of the following prerequisites are completed, as necessary, to support the preoperational test:

1. Construction acceptance testing is complete.

2. Component testing and instrument calibration are complete.

3. Test instrumentation is available and calibrated.

4. Support systems are available.

C. Test Method

1. Verify manual and automatic system controls.

2. Verify alarms, indicating instruments, and status lights are functional.

3. Verify design airflow.

4. Verify heat removal capability during hot functional testing with RCS at operating temperature.

D. Acceptance Criterion

The containment heat removal system operates as described in section 6.2 and subsection 9.4.6.

14.2.8.1.43 Containment Integrated Leak Rate Test (ILRT) Preoperational Test

A. Objective

To demonstrate that the total leakage from the containment does not exceed the maximum allowable leakage rate at the calculated peak containment internal pressure.

B. Prerequisites

1. The containment local leak rate tests are complete.
2. Containment isolation valves are closed by normal actuation methods.
3. Portions of fluid systems, which are part of the containment boundary that may be opened directly to the containment or outside atmosphere under post-accident conditions, are opened or vented to the appropriate atmosphere to place the containment in conditions as close to post-accident conditions as possible.
4. Containment penetration, including equipment and personnel airlocks, is closed.

C. Test Method

1. The ILRT is conducted, using the absolute method described in the American National Standard (ANSI) N45.4-1972 and ANSI/ANS-56.8-1981 which describe containment system leakage testing requirements. Measurements will be taken to calculate the leakage rate for the containment.
2. A verification test will be conducted to confirm the capability of the data acquisition and reduction system to satisfactorily determine the calculated integrated leakage rate. The verification test is accomplished by imposing a known leakage rate on the containment, or by pumping back a known quantity of air into the containment through a calibrated flow measurement device.

D. Acceptance Criterion

The containment integrated leakage does not exceed the maximum allowable leakage rate as defined in 10 CFR 50, Appendix J.

14.2.8.1.44 CRDM Cavity and Vessel Support Cooling System Preoperational Test

A. Objective

To demonstrate operation of CRDM cavity and vessel support cooling system.

B. Prerequisites

The required portions of the following prerequisites are completed, as necessary, to support the preoperational test:

1. Construction acceptance testing is complete.
2. Component testing and instrument calibration are complete.
3. Test instrumentation is available and calibrated.
4. Support systems are available.

C. Test Method

1. Verify manual and automatic system controls.
2. Verify design airflow.
3. Verify alarms, indicating instruments, and status lights are functional.

D. Acceptance Criterion

The CRDM cavity and vessel support cooling system operates as described in subsection 9.4.6.

14.2.8.1.45 Post-LOCA Purge Exhaust System Preoperational Test

A. Objective

To demonstrate operation of the post-LOCA purge exhaust system.

B. Prerequisites

The required portions of the following prerequisites are completed, as necessary, to support the preoperational test:

1. Construction acceptance testing is complete.
2. Component testing and instrument calibration are complete.
3. Test instrumentation is available and calibrated.
4. Support systems are available.

C. Test Method

1. Verify manual and automatic controls.
2. Verify design airflow.
3. Verify alarms and status lights are functional.

D. Acceptance Criterion

The post-LOCA purge exhaust system operates as described in subsection 6.2.5.

14.2.8.1.46 Hydrogen Monitor and Removal System Preoperational Test

A. Objective

To demonstrate operation of the hydrogen monitor (electric recombiner) and removal system.

B. Prerequisites

The required portions of the following prerequisites are completed, as necessary, to support the preoperational test:

1. Construction acceptance testing is complete.
2. Component testing and instrument calibration are complete.
3. Test instrumentation is available and calibrated.
4. Support systems are available.

C. Test Method

1. Verify manual and automatic controls.
2. Verify alarms and status lights are functional.
3. Verify design airflow.

D. Acceptance Criterion

The hydrogen monitor and removal system operates as described in subsection 6.2.5.

14.2.8.1.47 Containment Air Purification Cleanup System Preoperational Test

A. Objective

To demonstrate operation of the containment air purification cleanup system.

B. Prerequisites

The required portions of the following prerequisites are completed, as necessary, to support the preoperational test:

1. Construction acceptance testing is complete.
2. Component testing and instrument calibration are complete.
3. Test instrumentation is available and calibrated.
4. Support systems are available.

C. Test Method

1. Verify manual and automatic controls.
2. Verify alarms and status lights are functional.
3. Verify design airflow.
4. Verify particulate removal of HEPA filter.

D. Acceptance Criterion

The containment air purification cleanup system operates as described in subsection 9.4.6.

14.2.8.1.48 Gaseous Waste Processing System Preoperational Test

A. Objectives

1. To demonstrate operation of the waste gas compressors, catalytic hydrogen recombiners, and waste gas system components and their associated control and interlock circuitry.
2. To demonstrate waste gas compressor, catalytic hydrogen recombiner, and waste gas system performance during operation in their various modes.
3. To demonstrate operation of the recycle holdup tank vent eductor.
4. To demonstrate operation and performance of the gas decay tank drain pump and the drain system components.

B. Prerequisites

1. Required component testing, instrument calibration, and system flushing/cleaning are complete.
2. Required electrical power supplies and control circuits are energized and operational.
3. The auxiliary gas system is available to supply nitrogen to the waste gas system and oxygen to the catalytic hydrogen recombiners.
4. The ACCW system is available to supply cooling water to the waste gas compressors and catalytic hydrogen recombiners.
5. The reactor makeup water system is available to provide water to the waste gas compressors, catalytic hydrogen recombiners, and waste gas decay tank drain header.
6. The instrument air system is available to provide instrument air to waste gas system instruments and control valves.
7. A temporary source of hydrogen is available for checking the operation of the catalytic hydrogen recombiner.

C. Test Method

1. System component control circuits are verified, including response to normal control, interlock, and alarm signals.
2. The waste gas compressors and catalytic hydrogen recombiners are operated in their various modes, and their performance characteristics are verified.
3. Recycle holdup tank venting is performed, and the satisfactory operation of the recycle holdup tank vent eductor is verified.
4. The gas decay tank drain pump and system performance characteristics are verified during operation to remove water from a gas decay tank.

D. Acceptance Criteria

1. The waste gas compressors, catalytic hydrogen recombiners, and waste gas system components respond to normal control, interlock, and alarm signals. (See NSSS Startup Manual, SU-4.1.6.)

2. The waste gas compressors, catalytic hydrogen recombiners, and waste gas system are operable and their controls operate as described in section 11.3.2.

3. The recycle holdup tank vent eductor operates properly to perform recycle holdup tank venting. (See NSSS Startup Manual, SU-4.1.6.)

4. The gas decay tank drain pump and drain system performance characteristics are within design specifications. (See table 11.3.2-7.)

14.2.8.1.49 Liquid Waste Processing System Preoperational Test (See table 14.2.1-1 for Unit 2)

A. Objectives

1. To verify operation of the control circuitry of the liquid waste processing pumps, including the waste monitor, waste evaporator concentrates holdup tank, laundry and hot shower tank, chemical drain, spent resin sluice, and RCDT.

2. To demonstrate the operation and verify the operating characteristics of the liquid waste processing system, including pumps, valves, and tanks.

B. Prerequisites

Required component testing and instrument calibration are complete.

C. Test Method

1. The control circuitry and operation of system pumps and valves, including response to safeguards signals and high radiation isolation signals, will be verified.

2. The system will be operated and performance characteristics will be verified.

D. Acceptance Criteria

1. The performance characteristics of the system pumps are as designed. (See table 11.2.1-2.)

2. The pump and valve controls and interlocks are as designed. (See NSSS Startup Manual, SU-4.1.4.)

14.2.8.1.50 Waste Evaporator Preoperational Test (To be performed prior to utilization of the system)

A. Objectives

1. To demonstrate the operability of the waste evaporator and its associated pumps, valves, tanks, and control circuits.

2. To verify the capability of the waste evaporator to produce the required distillate output.

B. Prerequisites

1. Required component testing and instrument calibration are complete.

2. The ACCW system is available to supply cooling water.

3. The auxiliary steam system is available to supply steam to the waste evaporator when required.

4. Water is available with sufficient boron concentration to produce a batch of 12-percent boric acid in the evaporator when required.[(2)]

C. Test Method

1. The control circuitry and operation of system pumps and valves will be verified.

2. The system will be operated using demineralized water, and performance characteristics will be measured.

3. The system will be operated using borated water, and the evaporator output will be verified.

D. Acceptance Criteria

1. System pumps, valves, and interlocks operate as designed. (See NSSS Startup Manual, SU-4.1.5.)

2. The system operating characteristics are within the design limitations. (See table 11.2.1-2.)

14.2.8.1.51 Backflushable Filter System Preoperational Test

A. Objective

To demonstrate operation of the backflushable filter system.

B. Prerequisites

The required portions of the following prerequisites are completed, as necessary, to support the preoperational test:

1. Construction acceptance testing is complete.

2. Component testing and instrument calibration are complete.

3. Test instrumentation is available and calibrated.

4. Support systems are available.

C. Test Method

1. Verify manual and automatic (Unit 1 only) system controls.

2. Verify system flowrates.

3. Verify alarms, indicating instruments, and status lights are functional.

D. Acceptance Criterion

The backflushable filter system operates as described in section 11.4.

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[2] Initial operation is planned for 4-percent boric acid and will be preoperationally tested accordingly. Later, operation up to 12-percent boric acid will be permitted following preoperational testing to 12-percent boric acid.

14.2.8.1.52 (Deleted)

14.2.8.1.53 Boron Recycle System Preoperational Test

A. Objectives

1. To demonstrate the operability of the boron recycle system, including the recycle evaporator and its associated pumps, valves, tanks, and control circuits.
2. To verify the capability of the recycle evaporator to produce the required distillate output.

B. Prerequisites

1. Required component testing and instrument calibration are complete.
2. The ACCW system is available to supply cooling water.
3. The auxiliary steam system is available to supply steam to the system when required.
4. Water is available with sufficient boron concentration to produce a batch of 4-percent boric acid in the evaporator when required.

C. Test Method

1. The control circuitry and operation of system pumps and valves will be verified.
2. The system will be operated using demineralized water, and performance characteristics will be measured.
3. The recycle evaporator will be operated using borated water, and the evaporator output will be verified.

D. Acceptance Criteria

1. System valves, pumps, and interlocks operate as designed. (See NSSS Startup Manual, SU-4.1.2.)
2. The system operating characteristics are within design limitations. (See table 9.3.4-3.)

14.2.8.1.54 Containment, Auxiliary, Control, and Fuel Handling Buildings Drains System Preoperational Test (See table 14.2.1-1 for Unit 2.)

A. Objectives

1. To demonstrate operation of the containment, auxiliary, control, and fuel handling buildings drains system.
2. To demonstrate the ability of the containment cooler condensate measuring system to detect an increase in the amount of water collected by the system.

B. Prerequisites

The required portions of the following prerequisites are completed, as necessary, to support the preoperational test:

1. Construction acceptance testing is complete.
2. Component testing and instrument calibration are complete.
3. Test instrumentation is available and calibrated.
4. Support systems are available.
5. Verify system supply/inlet flow paths.

C. Test Method

1. Verify manual and automatic system controls.
2. Verify system discharge flow paths.
3. Verify alarms, indicating instruments, and status lights are functional.
4. A known amount of water is added to the containment coolers condensate measuring system, and its detection ability is verified.

D. Acceptance Criterion

1. The containment, auxiliary, control, and fuel handling buildings drains system operates as described in subsection 9.3.3.
2. The detection ability of the containment coolers condensate measuring system is verified.

14.2.8.1.55 Diesel Generator Fuel Oil System Preoperational Test

A. Objectives

1. To demonstrate the capability of the system to provide an adequate fuel supply to the diesel generator fuel oil day tanks.
2. To demonstrate operability of associated instrumentation and controls.

B. Prerequisites

1. Required component testing, instrument calibration, and system flushing/cleaning are complete.
2. Required electrical power supplies and control circuits are operational.

C. Test Method

1. Fuel oil is transferred from the fuel oil storage tank to the fuel oil day tanks by means of the transfer pumps. Appropriate flow parameters are recorded.
2. Simulation of day tank levels is utilized to verify the transfer pump automatic operation.

D. Acceptance Criterion

The performance of the diesel generator fuel oil system is in accordance with the design criteria and subsection 9.5.4.

14.2.8.1.56 Service Air System Preoperational Test

A. Objectives

1. To demonstrate operation of air compressors, air dryers, and isolation valves.
2. To verify that the system meets design air quality requirements.

B. Prerequisites

The required portions of the following prerequisites are completed, as necessary, to support the preoperational test:

1. Construction acceptance testing has been completed.
2. Permanently installed instrumentation is properly calibrated and operable.
3. Turbine plant closed cooling water system is available.
4. Appropriate ac and dc power sources are available.

C. Test Method

1. Simulate temperature and pressure signals to verify alarms and operation of isolation valves.
2. Operate compressors to verify operation.
3. Simulate safety signal to verify operation of the containment isolation valves.
4. Take batch samples of the service air for analysis.

D. Acceptance Criterion

Compressors, air dryers, and isolation valves perform as described in subsections 6.2.4 and 9.3.1.

14.2.8.1.57 Instrument Air System Preoperational Test

A. Objectives

1. To demonstrate operation of air dryers and isolation valves.
2. To verify that the system meets design air quality requirements.
3. To demonstrate that equipment designed to be supplied with instrument air is not supplied with service air.
4. To demonstrate that operation of components requiring large quantities of air does not cause excessive instrument air system pressure transients.

B. Prerequisites

1. Required construction acceptance testing is complete.
2. Required permanently installed instrumentation is properly calibrated and operable.
3. Service air system is operable.
4. Appropriate ac and dc power sources are available.

C. Test Method

1. Simulate pressure signals to verify alarms and operation of isolation valves.
2. Operate air dryers to verify operation.
3. Simulate safety signals to verify operation of the containment isolation valves.
4. Take batch samples of the instrument air for analysis.
5. Verify that, with service air shut off, there are no crossties from the service air system that introduce air into the instrument air system.
6. Simultaneously operate large users of instrument air and monitor the instrument air system pressures.

D. Acceptance Criterion

Air dryers and isolation valves perform as described in subsections 6.2.4 and 9.3.1.

14.2.8.1.58 Fire Protection System Preoperational Test

A. Objective

To demonstrate operation of the fire protection system.

B. Prerequisites

The required portions of the following prerequisites are completed, as necessary, to support the preoperational test:

1. Construction acceptance testing is complete.
2. Component testing and instrument calibration are complete.
3. Test instrumentation is available and calibrated.
4. Support systems are available.

C. Test Method

1. Demonstrate operation of the fire detection system.
2. Demonstrate operation of the fire protection water system:
   a. Demonstrate the head and flow characteristics of the diesel engine-driven fire water pumps, the electric motor-driven fire water pumps, and the operation of all auxiliaries.
   b. Verify control logic.
   c. Demonstrate flow paths of the fire protection system.
3. Demonstrate proper operation of the fire protection halon system.
4. Verify installation of portable fire protection apparatus.

D. Acceptance Criterion

The fire protection system operates as described in subsection 9.5.1.

14.2.8.1.59 Fuel Handling and Vessel Servicing Preoperational Test

A. Objectives

1. To demonstrate operation of the spent fuel handling tool, new fuel assembly handling tool, control drive shaft unlatching tool, and irradiation sample handling tool.
2. To demonstrate operation of the reactor vessel stud tensioner, and the polar crane, including the control circuits and interlocks.
3. To verify the crane has completed static testing at 125 percent and operational testing at 100 percent of rated load.

B. Prerequisites

1. The required component testing and instrument calibration are complete.
2. The fuel transfer system is available to support the testing of the fuel handling equipment.
3. The dummy subassembly and an irradiation sample holder are available for testing the fuel handling tool.

C. Test Method

1. Verify operation of the fuel handling equipment using the fuel transfer system and a dummy fuel assembly.
2. Verify fuel handling tool operations.
3. Verify control circuitry for the polar crane.
4. Verify the polar crane static and dynamic load testing has been completed.

D. Acceptance Criteria

1. The fuel handling tools are able to perform according to design. (See paragraph 9.1.4.2.4 and 9.1.4.3.14.)
2. The polar crane and its associated interlocks and control circuits perform as specified in its equipment technical manual.
3. The polar crane has completed static testing at 125 percent and operational testing at 100 percent of rated load (see ANSI B30.2, OSHA P 1910.)

14.2.8.1.60 Fuel Building Hoists and Elevator Preoperational Test

A. Objectives

1. To demonstrate operation of the spent fuel cask bridge crane, fuel handling machine, and new fuel elevator, including their control circuits and interlocks.
2. To verify the spent fuel cask bridge crane has completed static testing at 125 percent and operational testing at 100 percent of rated load.
3. To verify the fuel handling machine has completed static testing at 125 percent and operational testing using the dummy fuel assembly.

B. Prerequisite

Required component testing and instrument calibrations are complete.

C. Test Method

1. Verify the operability of the control circuits and associated interlocks of the cranes and elevator.

2. Verify the spent fuel cask bridge static and dynamic loading testing has been completed.

3. Verify that the fuel handling machine static and dynamic load testing has been completed.

D. Acceptance Criteria

1. The fuel building hoists and new fuel elevator perform all functions and operations as specified in their equipment technical manuals.

2. The spent fuel cask bridge crane has completed static testing at 125 percent and operational testing at 100 percent of rated load. (ANSI B30.2, OSHA P 1910.)

3. The fuel handling machine can lift 125 percent of its rated load and can transfer the dummy fuel assembly between the transfer cart and the fuel racks. (FSAR paragraphs 9.1.4.3.1.D and 9.1.4.3.1.F.)

14.2.8.1.61 Fuel Transfer System Preoperational Test

A. Objectives

1. To demonstrate operation of the fuel transfer system control circuits and associated interlocks.

2. To verify the ability of the refueling machine, new fuel elevator, fuel transfer car, spent fuel bridge crane, spent fuel cask handling crane, and associated fuel handling tools to transfer a dummy fuel assembly.

3. To perform refueling machine static load testing at 125 percent of rated load and operational load testing using the dummy fuel assembly.

B. Prerequisites

1. The required component testing and instrument calibration are complete.

2. There is a means available to lubricate the bearings of the fuel transfer equipment.

3. A dummy fuel assembly is available.

C. Test Method

1. The refueling machine will be load tested at 125 percent of rated load.

2. A dummy fuel assembly will be transferred from the new fuel pit to the refueling machine in the containment and back to the spent fuel pool to verify the operation of the fuel transfer equipment.

D. Acceptance Criteria

1. The fuel transfer system interlocks and interlock bypasses perform as specified in the equipment technical manual.

2. The fuel transfer equipment is able to transfer a dummy fuel assembly in and out of containment in accordance with the equipment technical manual.

3. The refueling machine can lift 125 percent of its rated load and can transfer the dummy fuel assembly between the transfer cart and the reactor vessel (paragraph 9.1.4.3.1.F (safety)).

14.2.8.1.62 Refueling Machine Preoperational Test

A. Objectives

1. To demonstrate the operation of refueling machine, its control circuits, and interlocks.

2. To demonstrate the handling of thimble plugs and rod cluster control assembly (RCCA) by the refueling machine.

B. Prerequisites

1. Required component testing and instrument calibration are complete.

2. A dummy fuel assembly is available.

3. The lower internals are installed in the reactor vessel.

C. Test Method

1. Operability of the refueling machine and control circuits of the bridge, trolley, hoist, and gripper interlocks will be verified.

2. Handling of thimble plugs and RCCAs will be demonstrated.

D. Acceptance Criteria

1. Control circuits and interlocks associated with the refueling machine allow the equipment to perform all necessary functions in accordance with the equipment technical manual.

2. The refueling machine is capable of proper handling of thimble plugs and RCCAs.

14.2.8.1.63 Refueling Machine Indexing Preoperational Test

A. Objectives

1. To verify the indexing of the refueling machine and establish bridge rail reference points for future operations.

2. To demonstrate the ability to transfer the dummy fuel assembly to the reactor vessel and determine that the lower core plate pins align with the bottom nozzle of the fuel assembly.

B. Prerequisites

1. Required component testing and instrument calibration are complete.
2. A dummy fuel assembly is available.
3. The lower internals are installed in the reactor vessel.

C. Test Method

1. Correct indexing will be verified using the dummy fuel assembly at the following core locations:
   a. At each location across the core in an east-west direction.
   b. At each location across the core in a north-south direction.
   c. At each location around the outer core periphery.
2. The dummy fuel assembly will be lowered into position in the core, and its alignment will be verified.

D. Acceptance Criteria

1. The refueling machine core indexes to all core locations.
2. The lower core plate pins align with the bottom nozzle of the dummy fuel assembly when it is transferred to a core location.

14.2.8.1.64 Diesel Generator Preoperational Test

A. Objectives

1. Demonstrate the operability of each diesel breaker and associated interlocks.
2. Demonstrate the operation of air start, lube oil, and jacket cooling water systems.
3. Demonstrate diesel reliability by performing 35 consecutive starts with no failures of each diesel.
4. Demonstrate the ability of the diesel generators to synchronize with the offsite power system.
5. Determine the fuel oil consumption of each diesel while operating under continuous rating load conditions.
6. Verify that all automatic diesel generator trips except engine overspeed, low lube oil pressure, high jacket water temperature, and generator differential are automatically bypassed upon loss of voltage on the emergency bus concurrent with a safety injection signal.
7. Verify that, with the diesel generator operating in the test mode connected to its bus, a simulated safety injection signal overrides the test mode by returning the diesel generator to standby operation.
8. Verify that the following diesel generator lockout features prevent the diesel generator starting:
   a. Turning gear engaged.
   b. Emergency stop.

9. Demonstrate full load carrying capability for 24 h.

B. Prerequisites

1. Required construction acceptance tests are complete.
2. Required electrical power supplies and control circuits are operational.
3. The NSCW system is available.
4. The diesel generator fuel oil system is available.
5. Adequate ventilation for the diesel generator building is available.

C. Test Method

1. The control logic of the diesel breakers, diesel start circuit, and support pumps and valves will be verified.
2. The operability of the diesel air start, lube oil, and jacket cooling water system will be verified.
3. There will be a demonstration of 35 consecutive starts of each diesel.
4. The diesel generators will be started, voltage and frequency control demonstrated, phase rotation verified, and the diesels synchronized to offsite power and loads.
5. During the testing, fuel oil consumption will be monitored with the diesels operating at the continuous load rating.
6. With a simulated loss-of-offsite-power signal, the proper diesel trips and bypasses will be verified.
7. With the diesel connected to its bus, a simulated safety injection signal will cause it to return to standby operation.
8. The diesel generator lockout features will be verified.
9. Demonstrate full load carrying capability for 24 h, of which 22 h are at a load equivalent to the continuous rating of the diesel generator and 2 h at a load equivalent to the 2-h rating of the diesel generator.

D. Acceptance Criteria

1. The controls, interlocks, and operation of the diesel generator breakers and support systems are as designed.
2. Each diesel must have 35 consecutive starts within the required time without a failure.
3. Each diesel generator is capable of being synchronized with offsite power.
4. Upon receipt of safeguards signals, the diesels operate as designed.
5. The diesel generator lockout features operate in accordance with design.
6. The diesel fuel oil consumption does not exceed the design requirements.
7. Each diesel satisfactorily completes the full-load test.

14.2.8.1.65 Auxiliary Building Flood Retaining Rooms Alarm System Preoperational Test

A. Objective

To demonstrate operation of auxiliary building flood retaining rooms alarm.

B. Prerequisites

The required portions of the following prerequisites are completed, as necessary, to support the preoperational test:

1. Construction acceptance testing is complete.
2. Component testing and instrument calibration are complete.
3. Test instrumentation is available and calibrated.
4. Support systems are available.

C. Test Method

Verify alarms, indicating instruments, and status lights.

D. Acceptance Criterion

The auxiliary building flood retaining room alarm system operates as described in subsection 9.3.3.

14.2.8.1.66 Main and Unit Auxiliary Transformers Preoperational Test

A. Objectives

1. To demonstrate the operation of protective relaying, alarms, and control devices associated with the main and unit auxiliary transformers.
2. To demonstrate the energization of the unit auxiliary transformers.

B. Prerequisites

1. Required construction acceptance testing has been completed.
2. Related ac and dc power sources are available.

C. Test Method

1. Simulate fault conditions to verify alarms and operation of protective relaying circuits.
2. The unit auxiliary transformers are energized. Voltage and phase rotation are verified and recorded.

D. Acceptance Criterion

Performance of the main and unit auxiliary transformers is in accordance with the design criteria and subsection 8.3.1.

14.2.8.1.67 Reserve Auxiliary Transformers Preoperational Test

A. Objectives

1. To demonstrate the operation of protective relaying, alarms, and control devices associated with the startup transformers.
2. To demonstrate the energization of the startup transformers.

B. Prerequisites

1. Required construction acceptance testing has been completed.
2. Related ac and dc power sources are available.

C. Test Method

1. Simulate fault conditions to verify alarms and operation of protective relaying circuits.
2. The transformers are energized. Voltage and phase rotation are verified and recorded.

D. Acceptance Criterion

The performance of the startup transformers is in accordance with the design criteria and subsection 8.3.1.

14.2.8.1.68 Non-Class 1E ac Distribution Preoperational Test

A. Objectives

1. To demonstrate the operability of bus supply breaker control logic and interlocks.
2. To demonstrate the automatic fast transfer from the unit auxiliary source or startup source is within design specifications.
3. To demonstrate the operation of system instrumentation and controls.
4. To demonstrate the energization of all buses and distribution panels.

B. Prerequisites

1. Required component testing and instrument calibration are complete.
2. Required electrical power supplies and control circuits are operational.

C. Test Method

1. Operability of bus supply breakers is demonstrated with the bus deenergized.
2. Simulation of automatic fast transfer initiation signals is utilized to demonstrate automatic fast transfer scheme operability as applicable.
3. Buses are energized from their alternate supplies. Voltage, phase rotation, and phase angles are measured and recorded.

D. Acceptance Criterion

The 13.8-kV, 4.16-kV, 480-V, and instrument 120-V-ac systems (non-class 1E) perform as described in subsection 8.3.1.

14.2.8.1.69 125-V dc (non-class 1E) Preoperational Test

A. Objectives

1. To demonstrate the capacity of each battery.
2. To demonstrate the capability of the batteries to supply the required loads for the required period of time.

B. Prerequisites

1. Required construction acceptance testing has been completed.
2. Required electrical power supplies and control circuits are operational.
3. Battery room ventilation is available.
4. Required resistive load devices are available.

C. Test Method

1. To determine capacity, each battery will be discharged, using the design-specified rate, to its minimum design voltage limit.
2. Each battery will be discharged using the required discharge rates and time periods.
3. Each individual battery cell will be measured following the discharge test.

D. Acceptance Criteria

1. Battery capacity will be verified to be 90 percent of the manufacturer's rating.
2. Each battery can carry the required load for the required time period.
3. Each battery cell is within the minimum design voltage limit at the end of the service test.

14.2.8.1.70 4.16-kV (class 1E) System Preoperational Test

A. Objectives

1. To demonstrate the operation of normal and alternate incoming supply breakers of 4.16-kV (class 1E) buses.
2. To demonstrate that the buses can be energized by the normal and alternate sources via these breakers.
3. To verify operation of interlocks, protective relaying, and ESF sequencer trip signals.

B. Prerequisites

1. Required construction acceptance testing has been completed.
2. Instruments required to perform the preoperational test shall be calibrated and operable prior to their use in that test.
3. Appropriate ac and dc power sources available.

C. Test Method

1. Operate breaker from control points and verify operation of switches, indicating lights, interlocks, and alarms.
2. Simulate abnormal conditions and verify responses and alarms.
3. The 4.16-kV (class 1E) buses are energized via their normal and alternate sources; bus voltage and phase rotation are recorded.

D. Acceptance Criterion

The 4.16-kV (class 1E) system performs as described in subsection 8.3.1.

14.2.8.1.71 Class 1E Standby Power Supply (4.16-kV Diesel Generator Sequencer) Preoperational Test

A. Objectives

1. Verify the control logic and operation of bus undervoltage and degraded grid undervoltage relays.
2. Verify the control logic of the load shed and load sequencer.

B. Prerequisites

1. Required construction acceptance tests are complete.
2. Required electrical power supplies are operational.
3. The required instrumentation calibration is complete.

C. Test Method

1. The sequencer control logic will be verified by simulating a loss of offsite power and by simulating a safeguard initiation and recording the response of the sequencer cabinets.
2. The logic for the bus undervoltage and degraded grid relays will be verified.

D. Acceptance Criteria

1. The sequencer, the bus undervoltage relays, and the degraded grid relays operate in accordance with design.
2. The loading intervals for the sequencer are within the design limits.

14.2.8.1.72 480-V (class 1E) Switchgear Preoperational Test

A. Objective

To perform the initial operation of the 480-V class 1E load centers and to demonstrate that they can be energized and operated.

B. Prerequisites

1. Required construction acceptance testing has been completed.
2. Required permanently installed instrumentation is properly calibrated and operable.

3. Required electrical ac power supplies and dc control circuits are operational and available.

C. Test Method

1. The manual controls, annunciation, and instrumentation for the 480-V class 1E load centers and their 4-kV feeder breakers are checked for correct operation.
2. The 480-V class 1E load centers are energized. Voltage and phase rotation are verified and recorded.

D. Acceptance Criteria

1. The manual controls, automatic sequencer controls, annunciation, and instrumentation for the 480-V class 1E load centers and their 4-kV feeder breakers operate correctly.
2. The voltage and phase rotation of each 480-V class 1E load center is within design specifications.
3. All 480-V class 1E load centers must perform as described in subsection 8.3.1.

14.2.8.1.73 480-V (class 1E) Motor Control Center Preoperational Test

A. Objectives

1. To demonstrate the operation of associated controls, interlocks, alarms, and solid-state trip devices.
2. To demonstrate energization of the applicable motor control centers (MCCs).

B. Prerequisites

1. Required construction acceptance testing has been completed.
2. Required ac and dc power sources are available.

C. Test Method

1. Operability of all MCC supply breakers is demonstrated.
2. Fault conditions that verify alarms and operation of solid state trip devices and protective relaying are simulated.
3. The MCCs are energized, with voltage and phase rotation recorded and verified.

D. Acceptance Criterion

The performance of the 480-V MCC (class 1E) system is in accordance with the design criteria and subsection 8.3.1.

14.2.8.1.74 125-V dc class 1E Preoperational Test

A. Objectives

1. To demonstrate the battery capacity.

2. To demonstrate the capability of the batteries to supply the required loads for the required period of time.

3. To demonstrate the ability of the battery chargers to carry the load.

4. To demonstrate the ability of the battery chargers to recharge the battery in the required time period, while carrying the largest combined continuous steady-state load expected on the dc system.

B. Prerequisites

1. Required construction acceptance testing has been completed.

2. Required electrical power supplies and control circuits are operational.

3. Required battery room ventilation is available.

4. Required resistive load devices are available.

C. Test Method

1. To determine capacity, each battery will be discharged, using the 2.75-h rate, to its minimum design voltage limit.

2. The battery will be charged to its fully charged condition, while the battery charger is simultaneously carrying the largest combined continuous steady-state load expected on that dc system.

3. Each battery will be discharged using the required discharge rates and time periods.

4. The battery charger will be subjected to its rated load for the required 8-h period.

5. Each individual battery cell will be measured following the discharge test.

D. Acceptance Criteria

1. Battery capacity will be verified to be 90 percent of the manufacturer's rating.

2. The battery charger can charge the battery to a fully charged condition following a design basis discharge within 12 h while carrying the largest combined continuous steady-state load expected on that dc system.

3. Each battery can carry the required load for 2 3/4 h without reaching its minimum discharged state.

4. The battery charger can carry nameplate-rated load as follows:

   a. Systems A and B - 400 A for 8 h.
   b. System C - 300 A for 8 h.
   c. System D - 200 A for 8 h.

14.2.8.1.75 Vital 120-V-ac class 1E Preoperational Test

A. Objectives

1. To demonstrate that the 120-V (class 1E)-ac distribution panels can be fed from their normal source inverters and from their backup source transformers by manual transfer.

2. To verify operation of system instrumentation and controls, including breaker protective interlocks.

B. Prerequisites

1. Required construction acceptance testing is complete.
2. Required component testing and instrumentation calibration are complete.
3. Appropriate ac and dc power sources are available.

C. Test Method

1. Normal source inverters are energized from 125-V-dc switchgear, and inverter input voltages are recorded.
2. The 120-V (class 1E)-ac distribution panels are energized from normal source inverters, and distribution panel voltages are recorded.
3. Normal source inverters are energized from a 480-V MCC (where available), and inverter input voltages are recorded.
4. The 120-V (class 1E)-ac distribution panels are energized from normal source inverters, and ac distribution panel voltages are recorded.
5. The 120-V (class 1E)-ac distribution panels are energized from their backup source transformer by manual transfer, and distribution panel voltages are recorded.
6. The system breakers are operated, and breaker interlocks are verified.

D. Acceptance Criterion

All 120-V (class 1E)-ac distribution panels perform as described in subsection 8.3.1, and their operation is in conformance with the Technical Specification limits.

14.2.8.1.76 Essential Lighting System Preoperational Test

A. Objectives

1. To demonstrate the capability of the essential lighting system to provide adequate illumination levels.
2. To verify operation of system instrumentation and controls.

B. Prerequisites

1. Required construction acceptance testing is complete.
2. Required electrical power supplies and control circuits are operable.
3. The essential lighting system has been energized.

C. Test Method

System function, correct electrical feeds, and correct fixture location will be verified per plan drawings.

D. Acceptance Criterion

The essential lighting system functions as described in subsection 9.5.3 provides illumination to areas of the plant designated by the VEGP design manual.

14.2.8.1.77 Emergency Lighting System Preoperational Test

A. Objectives

1. To demonstrate the capability of the emergency lighting system to provide adequate illumination levels.
2. To verify operation of system instrumentation and controls.

B. Prerequisites

1. Required electrical power supplies and control circuits are operable.
2. The emergency lighting system has been energized.

C. Test Method

1. The ability of the emergency lighting system is verified by simulating a loss of the normal and essential lighting and observing that the emergency system automatically activates.
2. Illumination levels and operation times will be verified.

D. Acceptance Criterion

The performance of the emergency lighting system tested is in accordance with the design criteria and subsections 8.3.1 and 9.5.3.

14.2.8.1.78 Offsite Communication System Preoperational Test

A. Objective

To demonstrate operation of the offsite communication system.

B. Prerequisites

The required portions of the following prerequisites are completed, as necessary, to support the preoperational test:

1. Construction acceptance testing is complete.
2. Support systems are available.

C. Test Method

1. Verify operation of the plant telephone branch exchange (PABX) system to provide adequate communication between onsite stations.
2. Verify operation of the PABX system in relation to the following offsite communication systems:
   a. Hotline to Georgia Power Company (GPC) general office production department.
   b. Southern Bell telephone system lines.
   c. Two telephone lines from GPC's Atlanta private branch exchange.

D. Acceptance Criterion

The offsite communication system operates as described in subsection 9.5.2.

14.2.8.1.79 Inplant Communication System Preoperational Test

A. Objective

To demonstrate the adequacy of the inplant communication system to provide reliable communications between plant areas and to verify the operability of the emergency alarm system.

B. Prerequisites

The required portions of the following prerequisites are completed, as necessary, to support the preoperational test:

1. Construction acceptance testing is complete.
2. Component testing and instrument calibration are complete.
3. Test instrumentation is available and calibrated.
4. Support system is available.
5. Plant equipment that contributes to the ambient noise level is in operation if possible.

C. Test Methods

1. Verify the telephone/page system operability.
2. Verify the sound-powered system operability.
3. Verify the tone generator operability.

D. Acceptance Criterion

The inplant communication system operates as described in subsection 9.5.2.

14.2.8.1.80 Heat Tracing System Preoperational Test

A. Objective

To demonstrate operation of the heat tracing system.

B. Prerequisites

The required portions of the following prerequisites are completed, as necessary, to support the preoperational test:

1. Construction acceptance testing is complete.
2. Component testing and instrument calibration are complete.
3. Test instrumentation is available and calibrated.
4. Support systems are available.

C. Test Method

1. Verify the operability of the heat tracing circuits.

2. Verify alarms, indicating instruments, and status lights are functional.

D. Acceptance Criterion

The heat tracing system automatically controls the associated heat tracing circuits, in accordance with system operation requirements as described in subsection 9.3.4.

14.2.8.1.81 (Material deleted)

14.2.8.1.82 Plant Annunciator System Preoperational Test

A. Objectives

1. To demonstrate operation of the plant annunciator system.
2. To verify that the annunciator will transfer to the alternate power source upon loss of primary power source.

B. Prerequisites

The required portions of the following prerequisites are completed, as necessary, to support the preoperational test:

1. Construction acceptance testing is complete to the extent required to perform the preoperational test.
2. Component testing and instrumentation calibration are complete.
3. Test instrumentation is available and calibrated.
4. Support systems are available.

C. Test Methods

1. Self-test features of the system shall be tested and verified to be operational.
2. Verify appropriate annunciator response to each retransmitting contact when it is actuated and verify the isolation device panels are operational.
3. Transfer of power sources shall be tested and verified operational.

D. Acceptance Criterion

The plant annunciator system operates as described in chapter 18.

14.2.8.1.83 Emergency Response Facility Post-Accident Monitoring and Sampling Systems Preoperational Test

A. Objective

To demonstrate operation of the emergency response facility post-accident monitoring and sampling systems.

B. Prerequisites

The required portions of the following prerequisites are completed, as necessary, to support the preoperational test:

1. Construction acceptance testing is complete.
2. Component testing and instrument calibration are complete.
3. Test instrumentation is available and calibrated.
4. Support systems are available.

C. Test Method

1. Verify instrument response to simulated external inputs.
2. Verify sampling system operation.

D. Acceptance Criteria

1. Post-accident sampling is described in subsection 9.3.2. The post-accident sampling system (PASS) was deleted in Technical Specification Amendments 123/101.
2. The post-accident monitoring system operates as described in subsection 7.5.3.
3. Verify the turbine plant sampling system operates as described in subsection 9.3.2.
4. Verify the nuclear sampling system operates as described in subsection 9.3.2.

14.2.8.1.84 Reactor Protection System and Engineered Safety Features Actuation System (ESFAS) Logic Preoperational Test

A. Objectives

1. To demonstrate operation of the ESFAS and reactor protection system and their ability to initiate appropriate reactor trip and safety actuation signals on receipt of simulated input signals.
2. To demonstrate operation of the reactor protection system block and permissive interlocks.
3. To demonstrate the functioning of the reactor protection system instrumentation inputs and BOP protection system instrumentation inputs to the SSPS including interlock and alarm functions.
4. To demonstrate operation of the reactor trip breakers and reactor trip bypass breakers, including breaker interlock, alarm, and tripping functions.
5. To demonstrate that the undervoltage coil and the shunt trip coil will function independently to trip the reactor trip breakers and bypass breakers following initiation of a manual reactor trip.
6. To demonstrate that reactor trip functions occur at design setpoints using simulated signals inserted at the NSSS and BOP process and protection rack inputs.

B. Prerequisites

1. Required component testing and instrument calibrations are complete.
2. Required electrical power supplies and control circuits are energized and operational.
3. Required slave output relay field wiring has been electrically isolated to prevent operation of components controlled by the relays except where specifically removed in the preoperational test procedure.
4. Plant system and components to be operated during testing are aligned prior to actuation of the particular system or component.

C. Test Method

1. Appropriate input signals are simulated and proper operation of the reactor protection logic matrix is verified by observing operation of reactor trip breakers and/or undervoltage coils.
2. Appropriate input signals are simulated and proper operation of the ESFAS logic matrix is verified by observing operation of the master relays and slave output relays.
3. Operation of the reactor protection system block and permissive interlocks is verified by observing proper response on receipt of appropriate input signals.
4. Operation of the reactor protection and the BOP protection panel inputs to the reaction protection system including setpoints, interlock, and alarm functions, are verified.
5. Operation of the reactor trip breakers and reactor trip bypass breakers and their associated trip, interlock, and alarm functions are verified.

D. Acceptance Criteria

1. The ESFAS and reactor protection systems operate as designed to initiate the appropriate reactor trip and safety actuation signals. (See NSSS Startup Manual, SU-2.7.2 and SU-2.7.3.)
2. Reactor protection system block and permissive interlocks operate in accordance with system design. (See drawings 1X6AA02-225, 1X6AA02-226, 1X6AA02-227, 1X6AA02-228, 1X6AA02-229, 1X6AA02-230, 1X6AA02-231, 1X6AA02-232, 1X6AA02-233, 1X6AA02-234, 1X6AA02-235, 1X6AA02-236, 1X6AA02-237, 1X6AA02-238, 1X6AA02-239, 1X6AA02-240, 1X6AA02-494, 1X6AA02-495, 1X6AA02-496, and 1X6AA02-519, table 7.2.1-2, and table 7.3.1-3.)
3. The reactor protection instrumentation inputs and BOP protection instrumentation inputs into the SSPS setpoint, interlock, and alarm functions, operate. (See paragraph 7.2.2.3.)
4. The reactor protection system logic trains respond during testing. (See paragraph 7.2.2.3).
5. The reactor trip breakers and reactor trip bypass breakers operate as designed to provide normal trip, interlock, and alarm functions. (See paragraph 7.2.2.3.)

14.2.8.1.85 Safeguards Test Cabinet (STC) Testing Capability

A. Objectives

1. To demonstrate that the ESFAS slave relays field wiring is terminated and that the ESFAS components actuate to the correct position.
2. To demonstrate that ESFAS slave relays can be operated from the safeguards test cabinet.

B. Prerequisites

1. Required component testing and instrument calibrations are complete.
2. Required electrical power supplies and control circuits are energized and operational.
3. Required slave output relay field wiring has been electrically isolated to prevent operation of components controlled by the relays except where specifically removed in the preoperational test procedure.
4. Plant system and components to be operated during testing are aligned prior to actuation of the particular system or component.
5. The reactor protection system logic trains and slave relays have been operationally checked with the reactor protection system test panels.

C. Test Method

1. Each ESFAS slave relay is actuated by the STC and relay operation is verified by contact operation, component actuation, or electrical test.
2. The ESFAS slave relays are actuated by the STC or simulated trip signals, and correct (proper positioning) component actuation and field wiring are verified by actual response or electrical checks (electrical checks are verifying a component breaker will change states, as required, when in the test position).

D. Acceptance Criteria

1. The ESFAS slave relay function is designed to actuate the ESFAS components. (See paragraph 7.3.1.1.1.1.)
2. The STC can actuate each slave relay individually.

14.2.8.1.86 Nuclear Instrumentation System Preoperational Test

A. Objective

To demonstrate the operability of the nuclear instrumentation source, intermediate, and power range channels, including their ability to supply signals for operating appropriate alarm and trip signals and indicating reactor power levels.

B. Prerequisites

1. Nuclear instrumentation system installation, calibration, and alignments are complete.
2. Required electrical power supplies are energized and operational.

3. The nuclear instrumentation system has been energized for at least 4 h.

C. Test Method

1. The ability of the source, intermediate, and power range nuclear instrumentation circuitry to respond to test signals is verified.
2. The source, intermediate, and power range instrumentation trip functions and alarm setpoints are verified.
3. Operation of the source range nuclear instrumentation audible count rate circuitry is verified.
4. Operation of the power range nuclear instrumentation comparator and rate circuitry is verified.

D. Acceptance Criteria

1. The source, intermediate, and power range nuclear instrumentation circuitry responds properly to test signals. (See NSSS Startup Manual, SU-2.9.1.)
2. The source, intermediate, and power range trip functions and alarm setpoints are within design specifications. (See NSSS Startup Manual, SU-2.9.1.)
3. The source range nuclear instrumentation audible count rate circuitry and power range nuclear instrumentation comparator and rate circuitry operate within design requirements. (See NSSS Startup Manual, SU-2.9.1.)

14.2.8.1.87 Process and Effluent Radiological Monitoring System Preoperational Test

A. Objective

To demonstrate operation of the process and effluent radiological monitoring system.

B. Prerequisites

The required portions of the following prerequisites are completed, as necessary, to support the preoperational test:

1. Construction acceptance testing is complete.
2. Component testing and instrument calibration are complete.
3. Test instrumentation is available and calibrated.
4. Suitable check sources are available.

C. Test Method

1. The operation of each monitor will be verified.
2. The control logic, annunciation, and power failure alarms of each monitor will be verified.

D. Acceptance Criterion

The process and effluent radiological monitoring system operates as described in section 11.5.

14.2.8.1.88 Incore Instrumentation System Preoperational Test

A. Objectives

1. To demonstrate operation of the incore movable detector system control equipment, drive assemblies, and transfer devices, including proper rotation, position indication, and limit switch actuation.
2. To demonstrate operation of the gas purge and leak detection systems associated with the incore movable detectors.

B. Prerequisites

1. The incore movable detector system installation is complete except for installation of the retractable guide thimbles.
2. Required component installation checks and instrument calibrations are complete.
3. Required electrical power supplies and control circuits are energized and operational.
4. A dummy cable is available for test performance.

C. Test Method

1. The functioning of the incore movable detector system control equipment drive assemblies and transfer devices is verified during operation in the various modes using a dummy cable.
2. Position indication and limit switch actuations are verified during test performance.
3. Operation of the gas purge and leak detection systems is verified by observing purge gas flow and observing drain system response to simulated high-level signal.

D. Acceptance Criteria

1. The drive assemblies insert and withdraw the detectors in the proper paths of the ten-path rotary transfer devices and operate in the high and low speeds in the Manual mode. Incore Instrumentation Manual Volume 1 1/2X6AB08-59.
2. The drive assemblies operate in the automatic mode for the Normal, Calibrate, Emergency, Common Group and Storage paths. Incore Instrumentation Manual Volume 1 1/2X6AB08-59.
3. The safety, withdraw, top, and bottom limit switches operate properly. Incore Instrumentation Manual Volume 1 1/2X6AB08-59.
4. The leak detection system is capable of detecting a leak. Incore Instrumentation Manual Volume 1 1/2X6AB08-59.

5. System limit switch setpoints are acceptable and in accordance with the manufacturer's instructions. (See NSSS Startup Manual, SU-2.9.3.)

14.2.8.1.89 Reactor Control, Rod Control, and Digital Rod Position Indication Preoperational Test

A. Objectives

1. To demonstrate the functioning of the rod control system and component cabinets.
2. To demonstrate operation of the rod control system in response to interlock signals.
3. To perform initial energization and calibration and demonstrate the functioning of the digital rod position indication system.

B. Prerequisites

1. Required component testing and instrument calibrations are complete.
2. Required electrical power supplies and control circuits are energized and operational.
3. The reactor vessel head is in place with all CRDM cables connected.
4. The integrated head and CRDM cooling system is in its normal operational alignment.

C. Test Method

1. The rod control system is initially operated, and functioning of the system and component cabinets is verified.
2. Rod control interlock actuation signals are simulated and response of the rod control system is verified.
3. The digital rod position indication system is initially energized and calibrated, and functioning of the system is verified through use of system-installed light-emitting diodes, annunciators, and a test fixture.

D. Acceptance Criteria

1. The rod control system and component cabinets operate as designed. (See NSSS Startup Manual, SU-2.5.2.)
2. The rod control system operates as designed in response to interlock signals (See table 7.7.1-1)
3. The digital rod position indication system operates as designed to provide rod position indication and alarms. (See section 7.7.1.3.2.)

14.2.8.1.90 CRDM Motor Generator Set Preoperational Test

A. Objectives

1. To demonstrate operation of the CRDM motor generator sets and system components including proper generator phasing for parallel operation.

2. To demonstrate operation of the motor generator set control system during motor generator starting, running, and parallel operation including verification of interlock and alarm functions.

B. Prerequisites

1. Required component testing and instrument calibrations are complete.
2. Required electrical power supplies are energized and operational.
3. A three-phase load bank is available for motor generator set testing under loaded conditions.

C. Test Method

1. CRDM motor generator set and system component control circuits including interlock and alarm functions are verified.
2. Generator phasing for parallel generator operation is verified.
3. Operation of the CRDM motor generator sets and control system is verified during starting, running, and parallel operation.

D. Acceptance Criteria

1. Generator phasing is proper for parallel motor generator set operation. (See Equipment Technical Manual.)
2. The CRDM motor generator sets and control system operates in accordance with design requirements during starting, running, and parallel operation. (See Equipment Technical Manual.)
3. The motor generator and control system alarms and interlocks function as designed. (See Equipment Technical Manual.)

14.2.8.1.91 CRDM Initial Timing Preoperational Test

A. Objectives

1. To demonstrate CRDM current command timing, rod speed, and bank overlap unit operation.
2. To demonstrate that the integrated head and CRDM cooling system maintain CRDM temperature within acceptable limits.

B. Prerequisites

1. Required component testing and instrument calibration are complete.
2. Required electrical power supplies and control circuits are energized and operational.
3. For the CRDM cooling checkout, the plant is at or near normal operating temperature, and pressure and hot functional testing is in progress.
4. The integrated head and CRDM cooling system is in its normal operational alignment.

C. Test Method

1. CRDM withdrawal and insertion are initiated; and proper current command sequence, timing, and rod speed signal voltages are verified.

2. Proper operation of the bank overlap unit to control rod bank sequence and movement is verified.

3. Adequacy of the integrated head and CRDM cooling system for maintaining CRDM temperature is verified by measuring CRDM coil resistances and calculating the coil temperatures.

D. Acceptance Criteria

1. The CRDMs operate as verified by current traces showing current command sequence and timing and measurement of rod speed signal voltages. (See NSSS Startup Manual, SU-2.5.1.)

2. The bank overlap unit operates to control rod movement and sequence. (See NSSS Startup Manual, SU-2.5.2.)

3. The integrated head and CRDM cooling system maintains CRDM coil temperatures within design limits. (See Equipment Technical Specification.)

14.2.8.1.92 Seismic Monitoring System Preoperational Test

A. Objective

To demonstrate operation of the seismic monitoring system.

B. Prerequisites

The required portions of the following prerequisites are completed, as necessary, to support the preoperational test:

1. Construction acceptance testing is complete.

2. Instrumentation calibration is complete.

3. Test instrumentation is available and calibrated.

C. Test Method

1. Verify operability of internal calibration devices by recording calibration records on all applicable sensors.

2. Verify system response to a simulated seismic event by actuating the appropriate trigger units, recording accelerograph outputs, and playing back the records for analysis.

3. Verify all alarms and indicators.

4. Verify installation and operation of the peak recording accelerographs.

D. Acceptance Criterion

The seismic monitoring system operates as described in subsection 3.7.4.

14.2.8.1.93 Nuclear Sampling System Preoperational Test

A. Objectives

1. To demonstrate the capability of the nuclear sampling system, liquid and gaseous, to collect representative samples from the RCS and auxiliary system process streams.
2. To verify the operation of system valves and control circuitry, including the response to safety signals.
3. To initially energize, calibrate, and verify the operation of the gross-failed fuel detector system.

B. Prerequisites

1. Required component testing and instrument calibration are complete.
2. ACCW is available for cooling the sample streams.
3. The systems to be sampled are at their normal pressure and temperature.
4. The CVCS is available to receive discharge from the nuclear sampling station.

C. Test Method

1. Samples are taken from each of the sample points; flows are adjusted; and flow, pressure, and temperature data are recorded.
2. The gross-failed fuel detector system is energized, and its operability is verified.
3. Control circuit and operability of the system valves are verified.

D. Acceptance Criteria

1. The sample system flows, pressures, and temperatures are within specifications such that normal samples can be taken and analyzed. (See Design Basis DC-1211, DC-1212.)
2. The system valve controls and interlocks function as designed. (See table 6.2.4-1.)

14.2.8.1.94 Miscellaneous Leak Detection System Preoperational Test

A. Objective

To demonstrate operation of the miscellaneous leak detection system.

B. Prerequisites

The required portions of the following prerequisites are completed, as necessary, to support the preoperational test:

1. Construction acceptance testing is complete.
2. Component testing and instrument calibration are complete.
3. Test instrumentation is available and calibrated.

4. Support systems are available.

C. Test Method

Verify alarms, indicating instruments, and status lights are functional.

D. Acceptance Criterion

The miscellaneous leak detection system operates as described in subsection 9.3.3.

14.2.8.1.95 Metal Impact Monitoring System Preoperational Test

A. Objectives

1. To perform initial calibration of the metal impact monitoring system.
2. To demonstrate operation of the metal impact monitoring system and establish the alarm signal level for use during preoperational testing.

B. Prerequisites

1. Metal impact monitoring system accelerometers have been mechanically fastened to the RCS.
2. Required electrical power supplies and control circuits are energized and operational.
3. For system operation demonstration portions of the test, the RCS hot functional test is in progress.

C. Test Method

1. The metal impact monitoring system initial calibration is performed in accordance with the manufacturer's instructions, including verification of accelerometer calibration.
2. During RCS hot functional testing, the metal impact monitoring system is set up for initial operation, and system operation is verified. The alert signal level for use during hot functional testing will be established.

D. Acceptance Criteria

1. Metal impact monitoring system calibration is acceptable and in accordance with the manufacturer's instructions. (See Equipment Technical Manual.)
2. System operates for RCS monitoring during hot functional testing.

14.2.8.1.96 Integrated Control Logic Safety Injection Preoperational Test

A. Objectives

1. To demonstrate that all safety-related equipment starts or repositions properly when the appropriate safeguard signals are actuated and that response is obtained when the actuation signals are reset.

2. To demonstrate operation of the offsite power distribution system and startup transformers when the safety-related components are actuated by safeguard signals.

3. To demonstrate the validity of the assumptions used and the analytical results of the class 1E electrical system voltage analyses (performed during HFT for Unit 2).

4. To demonstrate that the diesel generators start and accelerate to rated speed and voltage within the required time interval.

B. Prerequisites

1. Required preoperational testing, component testing, and instrument calibrations are complete.

2. Required electrical power supplies and control circuits are energized and operational.

3. The reactor vessel head has been removed, and the vessel and refueling cavity have been prepared to receive a large quantity of water.

4. The RWST contains an adequate supply of demineralized water for the performance of this test.

5. The condensate storage tanks contain an adequate supply of demineralized water for the performance of this test.

6. Plant systems and components to be operated during testing are aligned prior to actuation of the safeguard signals.

C. Test Method

1. Appropriate safeguard signals are initiated and equipment starting and repositioning are verified.

2. The time required for the diesel generators to reach rated speed and voltage following safeguards actuation is determined.

3. Operation of the offsite power distribution system and startup transformers are verified during the safeguard sequence.

4. Class 1E electrical system power and voltage drop data are taken during the safeguards sequence for Unit 1 and during HFT for Unit 2 and compared with analytical results of the voltage analyses.

5. Safety-related component response is verified when the appropriate safeguard signals are reset.

D. Acceptance Criteria

1. Safety-related equipment starts or repositions when appropriate safeguard signals are actuated, and equipment response is obtained when the actuation signals are reset. (See drawings 1X3D-AA-K02A, 2X3D-AA-K02A, 1X3D-AA-K02B, and 2X3D-AA-K02B.)

2. The offsite power distribution system and startup transformers operate during and following safeguard sequence loading.

3. The diesel generators start and accelerate to rated speed and voltage within the design specification time interval. (See Design Basis DC-1821.)

4. Voltage drop test results and the analytically derived voltage values provide good correlation to support the validity of the voltage analyses analytical results (performed during HFT for Unit 2). (See Design Basis DC-1821.)

14.2.8.1.97 Integrated Safeguards and Blackout Sequence Preoperational Test

A. Objectives

1. To demonstrate that the diesel generators start, offsite power is isolated, loads are shed, and the emergency bus loads are properly sequenced on the diesel generators; to verify voltage and frequency are maintained during sequencing; and to verify that proper train separation of safety-related components is maintained upon initiation of a safeguards actuation signal coincident with a simulated loss of offsite power.
2. To demonstrate that, upon interruption of the onsite source, the emergency bus loads are shed and that, upon restoration of the onsite source, subsequent loading is through the load sequencer.
3. To demonstrate that, upon a simulated loss of offsite power or upon a simulated loss of offsite power in conjunction with a safeguards actuation signal, the emergency buses are deenergized and loads are shed from the emergency buses.
4. To demonstrate that, upon a simulated loss of offsite power, the diesel generators start and emergency bus loads are sequenced on the diesel generators.
5. To demonstrate the functional capability of the diesel generators at full-load temperature conditions by simulating a loss of offsite power immediately following the full-load capability test.
6. To demonstrate that the continuous operation rating of the diesel generator is not exceeded, upon initiation of a simulated loss of offsite power or safeguards actuation coincident with a simulated loss of offsite power.
7. To demonstrate the capability of the diesel generators to reject a load equal to or greater than the largest single load associated with that diesel generator while maintaining speed and voltage, and the capability to reject load equal to its continuous rating without overspeed tripping of the diesel.
8. To demonstrate the capability of the diesel generators to synchronize with and transfer its loads to the offsite power source while the generator is loaded with its emergency bus loads.

B. Prerequisites

1. Required preoperational testing, component testing, and instrument calibrations are complete.
2. Required electrical power supplies and control circuits are energized and operational.

3. The reactor vessel head has been removed, and the vessel and refueling cavity have been prepared to receive a large quantity of water.

4. The RWST contains an adequate supply of demineralized water for the performance of this test.

5. The condensate storage tank contains an adequate supply of demineralized water for the performance of this test.

6. Plant systems and components to be operated during testing are aligned prior to actuation of the safeguard signals.

C. Test Method

1. Each train diesel generator and its emergency bus are functionally tested, with the opposite train ac and dc buses deenergized, by initiating a safeguards actuation signal coincident with a simulated loss of offsite power; load sequencing and train separation of safety-related components are verified; and the absence of voltage on other train ac and dc buses will be verified.

2. With the diesel generator supplying the emergency bus loads and offsite power not available, a loss of the diesel generator is simulated, and load shedding and subsequent loading by the load sequencer are verified.

3. Emergency bus deenergization and load shedding are verified on a simulated loss of offsite power and on a safeguards actuation signal in conjunction with a simulated loss of offsite power.

4. A loss of offsite power is simulated and diesel generator starting and emergency bus load sequencing are verified.

5. Immediately following the diesel generator full-load capability test, a loss of offsite power is simulated, and diesel generator functioning is verified.

6. Following initiation of a simulated loss of offsite power and a safeguards actuation coincident with a simulated loss of offsite power, diesel generator loading is checked to ensure that the diesel generator continuous operation rating is not exceeded.

7. A load equal to or greater than the largest single load associated with each diesel generator and a load equal to the diesel generator continuous rating are instantaneously removed from each diesel generator, and diesel generator load rejection capability is verified.

8. With each diesel generator loaded with the emergency bus loads, the capability to synchronize each diesel generator with the offsite source and to transfer load is verified.

D. Acceptance Criteria

1. Each train diesel generator functions as designed following initiation of a safeguards actuation signal coincident with a simulated loss of offsite power, emergency bus loads are sequenced on the bus, and train independence is verified. (See Design Basis DC-1821.)

2. On a simulated loss of the diesel generator with offsite power not available, the loads are shed from the emergency buses and subsequent loading is in accordance with design requirements. (See paragraph 8.3.1.1.3.)

3. The emergency buses are deenergized and loads are shed from the emergency buses upon a simulated loss of offsite power coincident with a safeguards actuation signal. (See paragraph 8.3.1.1.3.)
4. The diesel generators function as designed on a simulated loss of offsite power including a simultaneous start of the redundant units, and emergency bus loads are sequenced on the bus. (See Design Basis DC-1821.)
5. With the diesel generators at full-load temperature conditions, the diesel generators will function as designed following a simulated loss of offsite power. (See paragraph 8.3.1.1.3.)
6. The diesel generator loading is within design specifications following a simulated loss of offsite power or safeguards actuation coincident with a simulated loss of offsite power. (See paragraph 8.3.1.1.3.)
7. The diesel generator functions following rejection of a load equal to or greater than its largest associated load and rejection of load equal to its continuous rating. (See paragraph 8.3.1.1.3.)
8. With the diesel generators loaded with emergency bus loads, the diesel generators function properly during synchronizing and transferring loads to the offsite power source. (See paragraph 8.3.1.1.3.)

14.2.8.1.98 ESFAS Master and Slave Relay Preoperational Test

This test incorporated into 14.2.8.1.84, 14.2.8.1.85, and 14.2.8.1.106.

14.2.8.1.99 Containment Local Leakrate Preoperational Test

A. Objective

To determine the leakage rate of the containment penetrations and isolation valves.

B. Prerequisites

1. Containment isolation valves are closed by normal activation methods.
2. Associated piping is drained, and vent paths for leakage are established as required. (See table 6.2.6-1.)

C. Test Method

The containment penetrations and isolation valves are leak tested by performing type B and type C tests, in accordance with 10 CFR 50, Appendix J.

D. Acceptance Criterion

The combined leakage from containment penetrations and isolation valves is a 0.6 L where L is a leakage of 0.20 percent by weight of the containment air per 24 h at a pressure of not less than P (design accident pressure). (See subsection 6.2.6).

14.2.8.1.100 Reactor Containment Structural Integrity Test

A. Objective

To demonstrate the structural integrity of the reactor containment building.

B. Prerequisites

1. Containment penetrations are installed, and penetration leak tests are complete.
2. Containment penetrations, including equipment latches and personnel airlocks, are closed.

C. Test Method

The containment is pressurized to the test value and deflection measurements, and concrete crack inspections are made to determine that the actual structural response is within the limits predicted by the design analyses.

D. Acceptance Criterion

The containment structural response is within the limits predicted by design analyses (FSAR 3.8.1.7).

14.2.8.1.101 High-Efficiency Particulate Air Filters Preoperational Test

A. Objective

To demonstrate operation of the high-efficiency particulate air (HEPA) filters.

B. Prerequisites

The required portions of the following prerequisites are completed, as necessary, to support the preoperational test:

1. Construction acceptance testing is complete.
2. The ventilation systems containing HEPA filters and charcoal absorbers have been air balanced and are operational and available to support this test.
3. Component testing and instrument calibration is complete.
4. Test instrumentation is available and calibrated.
5. Support systems are available.

C. Test Method

1. HEPA filters and charcoal absorbers will be tested in place.
2. Verify alarms, indicating instruments, and status lights.
3. Verify design airflow.

D. Acceptance Criterion

The HEPA filters operate as described in subsection 6.5.1 and section 9.4.

14.2.8.1.102 Emergency Core Cooling System Sump Preoperational Test

A. Objectives

1. To demonstrate the operating characteristics of the RHR and containment spray pumps during the recirculation phase of safety injection.
2. To demonstrate the piping pressure drop from the containment sump to the suction of the RHR and containment spray pumps while in the recirculation mode.
3. To verify that vortex control and pressure drop across the sump screens were evaluated with a laboratory model study.

B. Prerequisites

1. Required preoperational testing, instrument calibration, and system flushing/cleaning are complete.
2. Required electrical power supplies and control circuits are energized and operational.
3. The RWST contains an adequate supply of demineralized water for performance of the test.
4. A temporary piping jumper connects each of the RHR and containment spray suction pipes at the containment sump.
5. Temporary piping is connected to the containment spray pump discharge lines to allow removal of the containment spray water from containment.

C. Test Method

1. Each RHR pump is started, taking a suction on the RWST through the temporary piping jumper to simulate the recirculation mode and pump operating characteristics; and piping pressure drops are recorded while discharging to the reactor vessel.
2. Each containment spray pump is started, taking a suction on the RWST through the temporary piping jumper to simulate the recirculation mode and pump operating characteristics; and piping pressure drops are recorded while discharging out the temporary discharge pipe.

D. Acceptance Criteria

1. The RHR and containment spray pumps operate as designed in the simulated recirculation mode. (See 6.2.2.2.2.3.2 and 6.3.2.2.4.)
2. The suction piping pressure drop for each pump is within the design limits. (See 6.2.2.2.2.3.2 and 6.3.2.2.4.)

14.2.8.1.103 Thermal Expansion Testing

A. Objectives

1. To verify that essential nuclear steam supply system and BOP components, piping, support, and restraint deflections are unobstructed and within design specifications.

2. To verify that thermal movements for safety-related snubbers, as delineated in the Technical Specifications and whose system operation temperature exceeds 250°F, are within design specifications.

B. Prerequisites

1. This test is conducted simultaneously with hot functional testing.
2. Supports, restraints, and hangers are installed, and reference points and predicted movements are established.
3. Temporary instrumentation is installed as required to monitor the expansion of the components under test.
4. A preservice examination has been performed on snubbers as defined in paragraph 14.2.8.1.103.A.2.

C. Test Method

1. During the reactor coolant system heatup and cooldown, deflection data are recorded.
2. Snubber thermal movements are verified by recording positions during initial system heatup and cooldown.
3. Snubber swing clearance is verified at specified heatup and cooldown intervals.

D. Acceptance Criteria

1. There shall be no evidence of blocking of the thermal expansion of any piping or components, other than by installed supports, restraints, and hangers (paragraph 3.9.B.2.1).
2. Spring hanger movements must remain within the hot and cold setpoints, and snubbers must not become fully retracted or extended.
3. Piping and components must return to their approximate baseline cold position.

14.2.8.1.104 Power Conversion and Emergency Core Cooling System Dynamics Test

A. Objective

To demonstrate during specified transients that the system's monitored parts respond in accordance with design calculations.

B. Prerequisites

1. Reference points for measurement of the systems are established, and required temporary instrumentation is installed and calibrated.
2. Hot functional testing is in progress for those portions of testing requiring elevated temperatures and pressure.
3. All subject systems are available for the specified dynamic operations.

C. Test Method

Deflection measurements are recorded during various plant transients.

D. Acceptance Criteria

1. The movements due to flow-induced loads shall not exceed design limits (subsection 3.9.2).

2. Flow-induced movements and loads will not cause malfunctions of plant equipment or instrumentation.

14.2.8.1.105 Remote Shutdown Preoperational Test

A. Objectives

1. To demonstrate the capability to cool down the plant from the hot standby condition to the cold shutdown condition using controls and instrumentation located outside the control room.

2. To demonstrate the capability to control plant parameters during a simulated loss of ac power using manual control and the steam-driven auxiliary feedwater pump.

B. Prerequisites

1. The controls and instrumentation associated with the remote shutdown panel are available.

2. Hot functional testing is in progress with the RCS temperature above that at which the RHR system is in operation.

C. Test Method

1. Transfer control from the control room to the remote shutdown panel.

2. Check the functioning of instrumentation, controls, alarms, and interlocks.

3. Remotely cool the plant down to the point of establishing RHR.

4. Remotely establish RHR, and reduce reactor coolant temperature approximately 50°F.

5. Demonstrate manual control of plant parameters using the atmospheric relief valves and the steam-driven auxiliary feedwater pump.

6. Transfer control back to the control room.

D. Acceptance Criteria

1. Transfer of control from the control room to the remote shutdown panel is achieved in accordance with design requirements. (See table 7.4.2-1.)

2. The potential for cooling the plant down from hot standby to cold shutdown has been demonstrated. (See Design Basis DC-1624.)

14.2.8.1.106 Reactor Trip System and ESFAS Response Time Test

A. Objective

1. To demonstrate that the reactor trip system and ESFAS response times are less than the maximum allowable response times as specified in the accident analysis.

2. To demonstrate that the containment isolation valves reposition in less than or equal to the required operating time.

B. Prerequisites

1. Required reactor protection system alignments, calibration, and testing are complete.
2. Required electrical power supplies and control circuits are energized and operational.
3. Special test circuitry test equipment and test instrumentation required for test performance are available.

C. Test Method

1. The response times of each primary sensor associated with the reactor trip system and the ESFAS will be obtained by testing with an appropriate test device.
2. Reactor trip system and ESFAS actuation signals are applied at the process sensor output, and the time interval is measured until loss of stationary gripper coil voltage and until the ESF equipment is capable of performing its safety function.
3. Total reactor trip system and ESFAS response time is computed by summing the results of the previous test methods.
4. Individual components and valve response times are accomplished in the system preoperational test procedures.

D. Acceptance Criterion

1. The total reactor trip system and total ESFAS response times do not exceed the allowable response times as specified.
2. Containment isolation valves reposition in less than or equal to the required time. (See table 6.2.4-1.)

14.2.8.1.107 Extraction Steam Test

A. Objectives

1. To verify proper operation and control of each extraction line isolation and drain valve.
2. To verify proper operation of each nonreturn valve.

B. Prerequisites

The required portions of the following prerequisites are completed as necessary to support the preoperational test:

1. Construction acceptance testing is complete.
2. Component testing and instrument calibration is complete.
3. Test instrumentation is available and calibrated.
4. Support systems are available.

C. Test Methods

1. Extraction line isolation and drain valves will be tested to verify the valves function in accordance with the control logic.

2. Each nonreturn valve will be observed to verify that the valve shuts following a turbine trip signal.

D. Acceptance Criteria

1. The extraction line isolation valves close and drain valves open following a turbine trip signal.

2. Each nonreturn valve closes following a turbine trip.

14.2.8.1.108 Condensate and Feedwater Chemical Injections System Test

A. Objective

To verify the operability of the condensate and feedwater chemical injection system.

B. Prerequisites

1. Required construction acceptance testing is complete.

2. Required system flushing/cleaning is complete.

3. Required electrical power supplies and control circuits are operational.

4. The condensate and feedwater systems are available.

C. Test Method

1. The operating parameters of the positive displacement chemical feed pumps and the wet layup pumps will be measured.

2. The mixing capability of the mixing pump will be verified.

D. Acceptance Criterion

1. The isolation valves operate as described in 10.4.7.2.2.5.

2. The automatic functions required to add chemicals performed as described in 10.4.7.5.

14.2.8.1.109 Proteus Computer Preoperational Test

A. Objectives

1. To verify computer hardware is operational.

2. To verify all digital inputs are conditioned correctly.

3. To verify computer software functions correctly.

B. Prerequisites

1. Required electrical power supplies are operational.

2. Test instrumentation is available and calibrated.

C. Test Method

1. Diagnostic programs are run on each section of the hardware.
2. Test signals are injected into the computer to simulate all digital inputs.
3. Software tests are run to verify operability of software.

D. Acceptance Criteria

1. The diagnostic programs or other functional tests are performed on computer hardware without error.
2. The computer conditions all digital inputs in agreement with the Proteus computer input/output list.
3. Applicable software routines run without error when subject to the applications program test procedures.

14.2.8.1.110 Equipment Building HVAC and Piping Penetration Preoperational Test

This test has been deleted.

14.2.8.1.111 Steam Generator Blowdown Processing System

A. Objective

To demonstrate that the steam generator blowdown processing system accepts water from each steam generator blowdown line, processes the blowdown as required, and delivers the processed water to the condensate system and the waste water retention basin.

B. Prerequisites

The required portions of the following prerequisites are completed, as necessary, to support the preoperational test:

1. Construction acceptance testing has been completed.
2. Component testing and instrument calibration has been completed.
3. Test instrumentation is available and has been calibrated.
4. Support systems are available.

C. Test Method

1. Verify manual and automatic system controls.
2. Verify flowrates, pressures, and temperatures.
3. Verify processed water quality.
4. Verify indications (flow, pressure, and temperature), status lights, and alarms.
5. Verify system isolation using simulated signals.
6. Verify flowpaths.

D. Acceptance Criterion

The steam generator processing system operates as described in subsection 10.4.8.

14.2.8.1.112 Main Turbine System Valves Preoperational Test

A. Objective

To demonstrate the functional performance of the main turbine system stop valves, including the following:

1. Operability of the main stop valves, control valves, intermediate stop valves, and intercept valves, including the actuation times from a turbine trip signal. Testing is to be performed in accordance with vendor requirements.

B. Prerequisites

The required portions of the following prerequisites are completed, as necessary, to support the preoperational test:

1. Construction acceptance testing completed.
2. Component testing and instrument calibration completed.
3. Test instrumentation available and calibrated.
4. Support systems available.

C. Test Method

1. The main stop valves, control valves, intermediate stop valves, and intercept valves will be tested in accordance with vendor requirements.
2. The various interfaces from control and trip signals will be simulated as required during testing.

D. Acceptance Criterion

The main turbine system stop valves operate as described in Section 10.2.

14.2.8.1.113 125-V-dc Class 1E Minimum Load Voltage Verification

A. Objectives

1. To measure the voltage drops, at nominal battery voltage, to 125-V-dc class 1E inverters and power-operated valves.
2. To determine the voltage which would be available at the 125-V-dc class 1E inverters and power-operated valves if the batteries were discharged to the minimum voltage limit.
3. To verify that the voltage available to 125-V-dc class 1E inverters and power-operated valves exceed the design minimum.

B. Prerequisites

1. Required construction acceptance testing is complete.

2. The 125-V-dc class 1E inverters and power-operated valves are operable.

3. Required load test devices are available.

C. Test Method

1. Each 125-V-dc class 1E inverter will be loaded to its design capacity and the voltage drop from the battery to the inverter input measured.

2. Each 125-V-dc class 1E power-operated valve will be operated and the voltage drop from the battery to the motor or solenoid measured.

3. The minimum available voltage at each 125-V-dc class 1E inverter and power-operated valve will be determined from the measured voltage drops and the battery minimum voltage limit.

D. Acceptance Criteria

1. The minimum available input voltage for the 125-V-dc class 1E inverters equals or exceeds 105 V-dc (paragraph 8.3.2.1.1).

2. The minimum available input voltage for the 125-V-dc class 1E power-operated valves equals or exceeds 100 V-dc or the equipment minimum-rated operating voltage, whichever is lower (paragraph 8.3.2.1.1).

14.2.8.1.114 Steady State Vibration Monitoring of Safety Related and High-Energy Piping

A. Objective

Demonstrate that steady state vibrations of safety- related and high-energy piping are within acceptable limits.

B. Prerequisites

1. The subject systems are available for operation at the specified conditions.

2. Portable instrumentation is available and calibrated.

C. Test Method

1. Specified safety-related and high-energy piping runs are screened qualitatively for perceptible vibration by visual inspection.

2. All piping observed to be vibrating is monitored with portable instrumentation. Results are assessed versus quantitative screening criteria or, if necessary, using standard stress evaluation techniques.

D. Acceptance Criterion

Steady state vibrations of safety-related and high- energy piping are within the allowable stress limits defined in 3.9.B.2.1-B.2.

14.2.8.1.115 AMSAC (ATWS Mitigating System Actuation Circuitry System)

A. Objectives

1. Demonstrate the capability of the AMSAC processing system to perform the required logic and timing functions.

2. Demonstrate that the AMSAC system produces the required output signals in response to specific input signals.

B. Prerequisites

7300 Process Protection and Control System is calibrated and available to the extent necessary to provide the required input signals to the AMSAC system.

C. Test Method

1. Internal AMSAC testing will be accomplished to verify input, output, setpoint, timer, and logic functions.

2. Overall tests will be accomplished to verify correct AMSAC output trip, and actuation signals are generated in response to specified input signal levels.

D. Acceptance Criteria

1. The AMSAC processing system performed the required logic and timing functions.

2. The AMSAC system produced the required output signals in response to specified input signals.

14.2.8.2 <u>Startup Test Procedures</u>

The following are the test abstracts for each startup test listed in table 14.2.1-2.

14.2.8.2.1 RCS Final Leak Test

A. Objective

To determine the amount of identified and unidentified leakage from the RCS and verify that the leakage is within allowable limits.

B. Prerequisites

1. Core loading is complete.

2. The RCS is at operating temperature and pressure.

C. Test Method

The identified and unidentified RCS leakage rates are determined by monitoring the system water inventory over a specified period of time.

D. Acceptance Criterion

Identified and unidentified RCS leakage is within Technical Specification limits.

14.2.8.2.2 Pressurizer Heater and Spray Capability and Continuous Spray Flow Verification Test (See Table 14.2.1-2 for Unit 2)

A. Objectives

1. To establish the optimum continuous spray flowrate.
2. To determine the effectiveness of the pressurizer heaters and normal control spray.
3. To demonstrate the depressurization rate by turning off pressurizer heaters and by using auxiliary spray.

B. Prerequisites

1. The RCS is at no-load operating temperature and pressure.
2. Reactor coolant pumps are operating.
3. Pressurizer heaters are operable.

C. Test Method

1. While maintaining constant pressurizer level, spray bypass valves are adjusted until a minimum flow is achieved which maintains the temperature difference between the spray line and the pressurizer within acceptable limits.
2. With the pressurizer spray valves closed, pressurizer heaters are energized, and the time to raise the pressurizer pressure a specified amount is recorded.
3. With the pressurizer heaters deenergized, both spray valves are fully opened, and the time to lower the pressurizer pressure a specified amount is recorded.
4. All except one reactor coolant pump are turned off, and the pressurizer heaters are turned off. The depressurization rate is then observed with its effect on margin to saturation temperature.
5. The pressurizer heaters are reestablished and auxiliary spray is initiated; charging and steam flow are then varied to observe the effects on RCS depressurization.

D. Acceptance Criteria

1. The spray bypass valves are throttled so that the minimum flow necessary to keep the spray line warm is achieved.
2. The pressurizer pressure response to the opening of the pressurizer spray valves and to the actuation of all pressurizer heaters is within the limits described in Section 5.4.10.

14.2.8.2.3 RCS Flow Measurement Test

A. Objectives

1. To obtain the necessary data to interrelate reactor coolant elbow tap differential pressure as a measurement of RCS flowrate, confirming

reactor coolant flow is equal to or greater than the design flow listed in table 5.1.2-1.

2. To perform a calorimetric flow measurement at less than 75-percent power, confirming RCS flow is equal to or greater than the design flow listed in table 5.1.2-1.

B. Prerequisites

1. Required instrument calibration is complete.
2. Required support systems are operational.
3. The reactor core is installed, and the plant is at normal operating temperature and pressure prior to initial criticality.

C. Test Method

1. Elbow tap differential pressure is taken prior to criticality and RCS flow is calculated.
2. A calorimetric is performed at the 50-percent power plateau and RCS flow is calculated.

D. Acceptance Criterion

Reactor coolant flowrate in each loop is equal to or greater than the design flow listed in table 5.1.2-1.

14.2.8.2.4 RTD Bypass Valve Flow Measurement Test

A. Objectives

1. To determine the flowrate necessary to achieve the design objectives for reactor coolant transport time in each resistance temperature detector (RTD) bypass loop.
2. To measure the flowrate in each RTD bypass loop and verify that the reactor coolant transport times in these loops (from the bypass loop connection on the main coolant loop to the last RTD) are acceptable.
3. To establish the low flow alarm setpoint for the total bypass loop flow for each reactor coolant loop.

B. Prerequisites

1. The installed pipe length measurements are made with the plant cold before insulation is installed.
2. The RCS is at operating temperature and pressure.

C. Test Method

1. The flowrate necessary to achieve the design reactor coolant transport time for each hot and cold leg bypass loop is calculated utilizing the piping length of each leg.
2. The hot and cold leg RTD bypass loop flow data are recorded at operating temperature and pressure.

D. Acceptance Criteria

1. Using actual piping lengths, the minimum flowrates were calculated to meet the transport time specified in the Westinghouse NSSS Startup Manual.

2. Actual flowrates in all RTD bypass manifolds exceed the minimum required flowrate required to achieve a 1.0.

3. Each RTD loop bypass low flow alarm has been verified to activate at the flow specified in the Westinghouse NSSS Startup Manual.

14.2.8.2.5 RCS Flow Coastdown Test

A. Objectives

1. To measure the rate at which reactor coolant flow changes, subsequent to tripping all reactor coolant pumps.

2. To measure various delay times associated with the loss of flow accident.

B. Prerequisites

1. Required component testing and instrument calibration are complete.

2. Required electrical power supplies and control circuits are operational.

3. The reactor core is installed, and the plant is at normal operating temperature and pressure with all reactor coolant pumps running.

C. Test Method

Flow coastdown stabilization and loss of coolant delay-time data are recorded while tripping all reactor coolant pumps.

D. Acceptance Criteria

1. The rate of change of reactor coolant flow for all reactor coolant pumps tripped is less than the rate of change assumed in section 15.0 within design specifications.

2. The RCS low-flow delay times are less than or equal to those assumed in the safety analysis for loss of flow in section 15.0.

14.2.8.2.6 Reactor Protection Test

A. Objectives

1. To verify that initial trip setpoint adjustments have been made prior to initial unit startup and to specify which trip setpoint adjustments will require readjustment during startup.

2. To obtain a record of all trip setpoints.

B. Prerequisites

1. Reactor trip instrumentation has been aligned and calibrated with setpoints adjusted to values given in the Technical Specifications or the unit test documents.

2. Reactor trip instrumentation has been energized for a time sufficient to achieve stability.

C. Test Method

1. Trip setpoints are reviewed and documented prior to criticality.
2. During startup and test operations, specific setpoints noted for readjustment on the data sheets are readjusted and final setpoint values recorded.

D. Acceptance Criteria

1. Initial reactor trip setpoints are verified to be within design criteria and in conformance with or more conservative than values in the Technical Specifications.
2. Setpoints readjusted during startup and testing are noted, and a final record of all setpoints is obtained.

14.2.8.2.7 Core Loading Instrumentation and Neutron Source Requirements Test

A. Objectives

1. To verify alignment, calibration, and neutron response of the temporary core loading instrumentation prior to the start of fuel loading.
2. To verify the neutron response of the nuclear instrumentation system (NIS) source range channels prior to the start of fuel loading.
3. To verify the neutron response of the temporary and NIS source range instrumentation prior to resumption of fuel loading following any delay of 8 h or more.

B. Prerequisites

1. Temporary core loading instrumentation package is available.
2. Operational alignment of the NIS that is to be performed prior to core loading is complete.
3. Plant and temporary instrumentation has been calibrated, and calibration data are available.

C. Test Method

1. A portable neutron source (1 to 5 Ci) and preshipment equipment checkout data are used to verify proper alignment, calibration, and neutron response of the temporary core loading instrumentation and the NIS source range instrumentation.
2. A portable neutron source (1 to 5 Ci) or movement of a source-bearing fuel element is used to produce the desired change in neutron level to verify the neutron response of the temporary core loading instrumentation and the NIS source range instrumentation prior to resumption of fuel loading, following any delay of 8 h or more.
3. A statistical evaluation of ten observations for each channel may be performed to verify operability of the equipment as an alternative to Test Method No. 2.

D. Acceptance Criterion

Neutron instrumentation is operational and calibrated and indicates the appropriate change in count rate as the neutron level is varied. (NIS source range instrumentation is calibrated in accordance with paragraph 14.2.8.2.10.)

14.2.8.2.8 Thermal Power Measurement and Statepoint Data Collection Test

A. Objectives

1. To determine reactor power by performing a heat balance at 30-, 50-, 75-, 90-, and 100-percent power.
2. To identify instrumentation for statepoint data collection.

B. Prerequisites

1. The following equipment has been installed and checked out and is operational: sensor for measuring steam generator feedwater temperature, differential pressure measuring devices for determining feedwater flow to each steam generator, and pressure gauges to measure steam pressure at steam generator outlets.
2. The following control systems are in automatic: pressurizer pressure and level and steam generator level.

C. Test Method

1. Collect data and calculate thermal power.
2. Obtain statepoint data, compute the average for each parameter measured, convert to the appropriate units, and summarize the data for each RCS loop.
3. Calibrate nuclear instruments to the power determined by the heat balance.

D. Acceptance Criterion

This test is for the collection of data.

14.2.8.2.9 Incore Detector Test

A. Objectives

1. To set up and demonstrate the operation of the incore instrumentation system.
2. To verify its adequacy for incore flux mapping.

B. Prerequisites

1. Reactor vessel upper internals are installed, and the reactor vessel head is installed with the studs tensioned.
2. The initial fuel loading has been completed.
3. Rotation and limit switch operation of the incore detector system has been verified.

4. Testing is performed at hot shutdown.

C. Test Method

1. A dummy cable will be inserted into each thimble to demonstrate system operation.

2. Detectors will be inserted into the thimbles to demonstrate system operation.

D. Acceptance Criterion

The system operates per its technical manual and is capable of taking a flux map.

14.2.8.2.10 Operational Alignment of Nuclear Instrumentation Test

A. Objectives

To establish and determine voltage settings, trip settings, operational settings, alarm settings, and overlap of channels on source range, intermediate range, and power range instrumentation from prior to initial criticality to at or near full reactor power.

B. Prerequisite

The NIS has been aligned in accordance with the Westinghouse Nuclear Instrumentation System Manual and the Precautions, Limitations, and Setpoints for Westinghouse NSSS.

C. Test Method

1. Functions will be calibrated, tested, and verified, utilizing permanently installed controls and adjustment mechanisms.

2. Operational modes of the source range, intermediate range, and power range channels will be set for their proper functions, in accordance with the test instructions.

D. Acceptance Criteria

1. The NIS demonstrates the ability to achieve the operational adjustments according to the Westinghouse Technical Manual.

2. The NIS demonstrates an overlap of indication between the source and intermediate ranges and the intermediate and power range instrumentation.

14.2.8.2.11 (Material deleted)

14.2.8.2.12 Rod Control System Test

A. Objective

To demonstrate and document that the rod control system performs the required control and indication functions just prior to initial criticality.

B. Prerequisites

1. RCS is at no-load operating temperature and pressure.

2. The NIS source range channels are aligned and operable.

C. Test Method

1. With the reactor at no-load temperature and pressure, just prior to initial criticality, the operation of the rod control system in various modes is verified.

2. The operation of status lights, alarms, and indicators is verified.

D. Acceptance Criteria

1. The rod control system will withdraw and insert each rod bank.

2. The rod position and indication system tracks each rod bank as it is being moved.

3. The control banks overlap system starts and stops rod movement at the designated bank positions.

14.2.8.2.13 CRDM Operational Test

A. Objectives

1. To demonstrate the operation of the CRDM under both cold and hot plant conditions.

2. To provide verification of slave cycler timing.

B. Prerequisites

1. The RCS is filled and vented at cold shutdown.

2. All rods are fully inserted.

3. Nuclear instrumentation channels are available.

4. A fast-speed oscillograph (Visicorder or equivalent) to monitor test parameters is available.

C. Test Method

1. With the reactor core installed and the reactor in the cold shutdown condition, confirm that the slave cycler devices supply operating signals to the CRDM stepping magnet coils.

2. Verify operation of all CRDMs under both cold and hot standby conditions. The CRDM magnet coil currents are recorded.

D. Acceptance Criterion

The CRDMs conform to the requirements for proper mechanism operation and timing as described in the magnetic CRDM instruction manual.

14.2.8.2.14 Rod Drop Time Measurement Test

A. Objectives

1. To determine the rod drop time of each RCCA under full-flow conditions, with the reactor at normal operating temperature and pressure.
2. To verify the operability of the control rod deceleration device.

B. Prerequisites

1. Initial core loading has been completed.
2. Source range channels are in operation.
3. All rods are fully inserted.

C. Test Method

1. Withdraw each RCCA, interrupt the electrical power to the associated CRDM, measure and record the rod drop time, and verify control rod deceleration.
2. Perform a minimum of three additional drops for each control rod whose drop time falls outside the two-sigma limit, as determined from the drop times obtained for each test condition.

D. Acceptance Criteria

1. The rod drop times are acceptable in accordance with plant Technical Specifications.
2. The control rod is slowed by the control rod deceleration device during rod drop testing.

14.2.8.2.15 Rod Position Indication Test

A. Objective

To verify that the rod position indication system satisfactorily performs required indication and alarm functions for each individual rod and that each rod operates satisfactorily over its entire range of travel.

B. Prerequisites

1. The RCS is at no-load operating temperature and pressure.
2. At least one reactor coolant pump in service with reactor coolant boron concentration not less than specified in the Technical Specifications for refueling shutdown.

C. Test Method

Rod banks are individually withdrawn from and reinserted into the core, according to the test procedure, while recording analog output voltage, control room position readout, and the group step counters.

D. Acceptance Criterion

The rod position indication system performs the required indication and alarm functions, as described in section 7.7.1.3, and each rod operates over its entire range of travel.

14.2.8.2.16 Operational Alignment of Process Temperature Instrumentation Test

A. Objective

To align ΔT and T_{avg} process instrumentation under isothermal conditions prior to criticality and at power.

B. Prerequisites

1. All reactor coolant pumps are operating.
2. The RCS average temperature must be at the hot no-load average temperature ±2°F.

C. Test Method

1. Align ΔT and T_{avg} according to test instructions at isothermal conditions prior to criticality and at approximately 75-percent power. Extrapolate the 75-percent data to determine ΔT and T_{avg} values for the 100-percent plateau.
2. At or near 100-percent power, check the alignment of the ΔT and T_{avg} channels for agreement with the results of the thermal power measurement.

D. Acceptance Criteria

1. The T_{avg} should be within 0.5°F of the value calculated from the T_{hot} and T_{cold} converter outputs.
2. The T_{hot} and T_{cold} measured values are within 1.2°F of the spare RTDs for each channel.
3. The measured ΔT for each channel is within 1 percent of the calorimetric power at power and 0.5 percent at isothermal conditions.

14.2.8.2.17 Startup Adjustments of RCS Test

A. Objective

To obtain the optimum plant efficiency.

B. Prerequisites

1. The RCS is at no-load operating temperature and pressure.
2. The RCS temperature is being controlled by the dump valves.

C. Test Method

1. Obtain system temperature and steam pressure data at steady-state conditions for zero power and at hold points during power escalations.

2. Evaluation of this data will provide the basis for adjustments to the RCS.

D. Acceptance Criteria

1. Steam generator pressure is within ±10 psi of design at 100-percent power.

2. The full load T_{avg} shall not exceed 588.5°F.

14.2.8.2.18 RCS Sampling for Core Loading Test

A. Objective

To verify correct and uniform boron concentration, prior to core loading in the RCS and directly connected auxiliary systems.

B. Prerequisites

1. Boric acid storage tanks, transfer pumps, and associated piping and equipment are available for use.

2. The RCS is filled with reactor grade water which has been borated to the concentration specified in the initial core loading procedure.

C. Test Method

1. The RCS will be filled and on recirculation to maintain a uniform boron concentration.

2. Four samples will be obtained at approximately four equidistant depths from the surface to the bottom of the reactor vessel. Also, a sample will be taken from the operating RHR loop.

3. These samples will be analyzed to verify that there is a uniform boron concentration in the RCS.

D. Acceptance Criteria

1. The samples obtained from the designated sample points shall have a minimum boron concentration of 2000 ppm.

2. The reactor vessel samples are within 50 ppm of each other.

14.2.8.2.19 Metal Impact Monitoring System Test

A. Objectives

1. To perform final calibration of the metal impact monitoring system.

2. The alarm signal level will be established for startup testing and operations.

B. Prerequisites

1. Required electrical power supplies and control circuits are energized and operational.

2. Metal impact monitoring system accelerometers have been installed at the proper locations.

C. Test Method

1. The metal impact monitoring system calibration is performed in accordance with the manufacturer's instructions including verification of accelerometer calibration.

2. After fuel loading, the RCS metal impact monitoring system operation is verified. The alert signal level is established.

D. Acceptance Criteria

1. A metal impact monitoring system calibration is acceptable and in accordance with the manufacturer's instructions.

2. The system operates for RCS monitoring during startup testing and power operations.

14.2.8.2.20 Initial Fuel Loading Test Sequence

A. Objective

1. To establish the conditions under which the initial fuel loading is to be accomplished.

2. To accomplish initial fuel loading in a safe and orderly manner.

B. Prerequisites

1. Testing prior to initial fuel loading is completed sufficiently to demonstrate the operability of required systems and components.

2. Temporary and permanent source range channels are operable.

3. At least one path for boron addition to the RCS is available.

4. Uniform boron concentration in the RCS is maintained by recirculation with at least one RHR pump and is sufficient to ensure required boron concentration in accordance with the Technical Specifications during fuel loading.

5. Containment integrity is established in accordance with applicable Technical Specifications.

C. Test Method

1. Fuel is inserted into the reactor vessel in accordance with the prespecified loading sequence.

2. Neutron count rate is monitored on temporary and permanent source range detectors.

3. Core reactivity is monitored through plots of inverse neutron count rate ratio.

D. Acceptance Criterion

The core is assembled in accordance with the prespecified configuration.

14.2.8.2.21 Inverse Count Rate Ratio Monitoring for Fuel Loading Test

A. Objective

To describe the frequency, core conditions, and methods for obtaining nuclear monitoring data during initial core loading.

B. Prerequisites

1. Temporary and plant source range nuclear instrumentation has been operational for a minimum of 60 min to allow the instruments to attain stable operating conditions.
2. The plant is prepared for initial core loading.

C. Test Method

1. Data from the temporary and permanent nuclear monitoring channels are used to assess the core multiplication factor during core loading operations.
2. The inverse count rate ratio is plotted and evaluated to prevent any unexpected deviation from the required core multiplication factor.

D. Acceptance Criteria

Acceptance criteria are not applicable to this test.

14.2.8.2.22 Determination of Core Power Range for Physics Testing

A. Objectives

1. To determine the reactor power level at which effects from fuel heating is detectable.
2. To establish the range of neutron flux in which zero power reactivity measurements are to be performed.

B. Prerequisites

1. The reactor is critical and stable in the intermediate range.
2. Control rods are sufficiently deep in the core to allow positive reactivity insertion by rod withdrawal.
3. Reactor coolant temperature is established at a value that minimizes the moderator temperature coefficient reactivity feedback.

C. Test Method

1. Withdraw control rod bank and allow the neutron flux level to increase until nuclear heating effects are indicated by the reactivity computer.
2. Record the reactivity computer picoammeter flux level and, if possible, the corresponding intermediate range channel currents at which nuclear heating occurs, to obtain zero power testing range.

D. Acceptance Criterion

This procedure is provided to collect data and to determine the power range at which zero power testing will be conducted.

14.2.8.2.23 Precritical Test Sequence

A. Objectives

To specify the sequence of events which constitute the precritical test program.

B. Prerequisite

Plant system conditions are established as required by the individual test instructions within this sequence.

C. Test Method

This instruction will establish the sequence for required testing after core loading, until the plant has completed all precritical testing and reached the hot shutdown condition.

D. Acceptance Criteria

Acceptance criteria are contained in the various individual tests conducted during this time.

14.2.8.2.24 Dynamic Automatic Steam Dump Control Test

A. Objective

To verify automatic operation of the T_{avg} steam dump control system, demonstrate controller setpoint adequacy, and obtain final settings from steam pressure control of the condenser dump valves.

B. Prerequisites

1. The steam dump control system has been aligned and calibrated to initial settings.
2. The plant is at no-load temperature and pressure.
3. The condenser vacuum has been established.
4. The reactor is critical.

C. Test Method

1. Reactor power is increased to less than 10 percent by rod withdrawal and steam dump to condenser to demonstrate setpoint adequacy.
2. Pressure controller setpoint is increased prior to switching to T_{avg} control, which rapidly modulates open condenser dump valves.
3. Simulate turbine operating conditions with reactor at power, then simulate a turbine trip resulting in the rapid opening of the steam dump valves.

D. Acceptance Criteria

1. The plant trip controller responds to maintain a stable T_{avg}. After steady state power is achieved, no divergent oscillations in temperature occur.
2. The loss of load controller responds properly to maintain a specified stable T_{avg}. After steady state power is achieved, no divergent oscillations in temperature occur.

3. The steam header pressure controller responds to maintain a stable pressure at normal no-load pressure.

14.2.8.2.25 Automatic Steam Generator Level Control Test

A. Objective

To verify the stability of the automatic steam generator level control system following simulated transients at low-power conditions and the operation of the variable speed feature of the feedwater pumps.

B. Prerequisites

1. The reactor is critical, and at less than 10-percent power.
2. The steam generator level control system has been checked and calibrated.
3. Steam generator alarm setpoints have been set for each generator.

C. Test Method

1. Induce simulated steam generator level transients to verify steam generator level control response.
2. Verify the variable speed features of the main feedwater pumps by manipulation of controllers and test input signals.

D. Acceptance Criteria

1. Steam generator level overshoot (undershoot) is less than 4 percent following a level change; the level will track to within 2 percent of the level setpoint within three reset time constants following a control transfer, level, or setpoint change.
2. Feedwater pump discharge pressure oscillations are less than 3 percent within three reset time constants following a steamflow change.
3. The main feedwater regulating valves open and stabilize in response to various steam flow conditions.

14.2.8.2.26 Power Coefficient Determination Test

A. Objective

To verify nuclear design predictions of the doppler-only power coefficient.

B. Prerequisites

1. Reactor power level is at the specified testing plateau (30-, 50-, 75-, and greater than 90-percent power). Reactor coolant temperature and pressure, and rod cluster control (RCC) bank configurations are at normal operating conditions and are stable.
2. All subsystems which affect overall plant transient response, except the rod control system, should, whenever possible, be in automatic control.

C. Test Method

1. Plant conditions are stabilized at selected power plateaus (30-, 50-, 75-, and greater than 90-percent). The plant load is varied, using the turbine-generator controller, in a manner approximating step changes of about 2- to 4-percent power.

2. The doppler-only power coefficient verification factor is determined from correlative measurements of reactor coolant system temperature and core thermal power output.

D. Acceptance Criterion

The doppler-only power coefficient verification factor agrees with the value given in the nuclear fuel design report.

14.2.8.2.27 Load Swing Test

A. Objective

To verify nuclear plant transient response, including automatic control system performance, when step-load changes are introduced to the turbine-generator at 30-, 75-, and 100-percent power levels.

B. Prerequisite

The plant is operating in a steady state condition at the desired power level.

C. Test Method

The turbine-generator output is changed as rapidly as possible to achieve a step load increase or decrease. Selected plant parameters are monitored and recorded during the load transients.

D. Acceptance Criterion

The primary and secondary control systems, with no manual intervention, maintain reactor power, RCS temperature, pressurizer pressure and level, steam generator levels and pressures within acceptable ranges during steady-state and transient operation.(3)

14.2.8.2.28 Process and Effluent Radiation Monitoring System Test

A. Objectives

1. For monitors which:

   a. Will be used for establishing conformance with the safety limits or limiting conditions for operation that will be included in the facility technical specifications, or,
   b. Are classified as engineered safety features or will be relied on to support or ensure operation of the engineered safety features within design limits, or,

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3 Control system response is reviewed and adjustments to the control systems are made, if necessary, prior to proceeding to the next power plateau.

c. Are assumed to function or for which credit is taken in the accident analysis of the facility, and,
d. Will be used to process, store, control, or limit the release of radioactive materials.

2. The objectives are:

a. To verify the calibration of the process and effluent radiation monitor against an acceptable standard.
b. To establish baseline activity and background levels.
c. To demonstrate that process and effluent radiation monitoring systems are responding correctly by performing independent laboratory or other analyses.

B. Prerequisites

1. The plant is stable at the desired power level.

2. The sampling systems for the process and effluent systems are operable.

C. Test Method

1. Calibrations are performed with the use of radioactive sources to verify proper operation of the monitors and detectors.

2. Samples are collected, analyzed with laboratory instruments, and compared with the process and effluent monitor to verify proper operation.

3. Background levels are established at low power (less than 5-percent power); background levels and baseline activity levels determined by sampling are established at 100-percent power to monitor the buildup of activity.

D. Acceptance Criteria

1. Process and effluent radiation monitors are calibrated against radioactive standards.

2. Baseline activities are established.

3. With the exception of ARE-0014, laboratory analyses agree satisfactorily, to the extent practical given sensitivity and energy response, with the process and effluent radiation monitors.

14.2.8.2.29 Axial Flux Difference Instrumentation Calibration Test

A. Objective

To derive calibration factors for overpower and overtemperature ΔT setpoints based on incore flux data, calorimetric data, and excore nuclear instrumentation detector currents.

B. Prerequisites

1. The NIS power range isolation amplifiers have been aligned.

2. The core average axial offset from the incore data is available from flux maps.

3. Average core thermal power from calometrics, performed during the flux mapping, is available.

C. Test Method

Collect data, as required by test instruction, at 75-percent power, perform FΔI calculations to the calibration factors, and extrapolate results for use at the 100-percent power plateau.

D. Acceptance Criterion

Calibration factors agree with Technical Specifications.

14.2.8.2.30 Flux Map Test

A. Objective

To determine the reactor core power distributions for various control rod configurations at 30-, 50-, 75-, and 100-percent power.

B. Prerequisites

1. Incore instrumentation and process computer are operable.
2. Reactor is critical, and power level is established as necessary.

C. Test Method

Use data collected from incore detectors to generate a flux map.

D. Acceptance Criterion

The core flux distribution indicated by the flux maps are acceptable and in accordance with plant Technical Specifications.

14.2.8.2.31 Automatic Reactor Control System Test

A. Objectives

To demonstrate the capability of the reactor control system to respond to input signals.

B. Prerequisites

1. The reactor is at equilibrium at the power level specified by the startup test program reference document.
2. Setpoints and controls for the pressurizer, steam generator, steam dump, and feedwater pump have been checked and set to proper values.

C. Test Method

T_{avg} is varied from the T_{ref} setpoint to verify the transient recovery capabilities of the automatic RCS.

D. Acceptance Criterion

T_{avg} should return to within $\pm$ 1.5°F of the T_{ref} setpoint.

14.2.8.2.32 (Material Deleted)

14.2.8.2.33 (Material Deleted)

14.2.8.2.34 Pseudo Rod Ejection Test (See Table 14.2.1-2 for Unit 2)

A. Objective

To verify that the power distribution resulting from a simulated, ejected RCCA is within acceptable limits.

B. Prerequisites

1. Excore and incore nuclear instrumentation and incore thermocouples are operable and have been calibrated.
2. The test is performed with the reactor critical at 0-percent power.

C. Test Method

1. Simulate rod ejection accident at the required power levels by withdrawing an RCCA.
2. Record data and calculate power distribution.

D. Acceptance Criterion

The power distribution resulting from the withdrawn RCCA is consistent with the power distribution assumed in the plant safety analysis.

14.2.8.2.35 RCCA Bank Worth Measurement at Zero Power Test (See Table 14.2.1-2 for Unit 2)

A. Objectives

1. To determine the differential and integral reactivity worth of RCCA banks.
2. To determine that the highest worth rod is less than or equal to the value used in the Nuclear Fuel Design Report.

B. Prerequisites

1. The reactor is critical at zero power, with no-load temperature and pressure stable.
2. The reactivity computer is installed and calibrated.

C. Test Method

RCCA bank worths are determined by constant addition and/or dilution of boron in the RCS, causing rod movement to compensate for the boron concentration changes. This rod movement will cause step changes in reactivity which are used to compute the bank worths.

D. Acceptance Criteria

1. The RCCA bank worths are in accordance with the Westinghouse Startup Program Reference Document.

2. The worth of the highest worth rod is less than or equal to the value used in the Nuclear Fuel Design Report.

14.2.8.2.36 Control Systems Checkout Test

This material is included in the Load Swing Test, 14.2.8.2.27; the System Generator Level Control Test, 14.2.8.2.25; and the Automatic Steam Dump Control Test, 14.2.8.2.24.

14.2.8.2.37 Boron Endpoint Determination Test

A. Objective

To determine the critical reactor coolant system boron concentration appropriate to an endpoint configuration (RCC configuration).

B. Prerequisites

1. The reactor is critical at zero power, with no-load temperature and pressure stable.

2. The reactivity computer has been installed.

C. Test Method

1. The control rods are positioned close to the desired endpoint configuration, with the RCS boron concentration stabilized.

2. The rods are quickly moved to the desired endpoint configuration without boron concentration adjustment.

3. The change in reactivity is measured, and this is converted to an equivalent change in boron concentration which will yield the endpoint for the given rod configuration.

D. Acceptance Criterion

The results of the boron endpoint calculations agree with the acceptance criteria of the Westinghouse Startup Program Reference Document.

14.2.8.2.38 Inverse Count Rate Ratio Monitoring for Approach to Initial Criticality Test

A. Objective

To describe methods for obtaining and evaluating data used as an indication of core reactivity during the approach to initial criticality.

B. Prerequisites

1. Both source range and intermediate range nuclear channels alarms, trip functions, and indicating devices have been checked out and calibrated.

2. Both source range and intermediate range nuclear channels have been energized a minimum of 60 min to ensure stable operation.

C. Test Method

1. Obtain baseline count rates prior to rod withdrawal and boron dilution. After each increment of rod withdrawal and periodically during boron dilution, count rates are obtained and inverse count rate ratio is evaluated.

2. Core reactivity is monitored during the approach to criticality.

D. Acceptance Criteria

Acceptance criteria are not applicable to this test.

14.2.8.2.39 Initial Criticality Test

A. Objective

To achieve initial criticality in a controlled manner.

B. Prerequisite

Plant system conditions are established as required in the plant Technical Specifications.

C. Test Method

1. At preselected points during rod withdrawal and/or boron dilution, data are taken and the inverse count rate ratio is plotted to enable an extrapolation to be carried out to predict the expected critical point.

2. Initial criticality may be achieved by boron dilution or, if desired, by rod withdrawal.

D. Acceptance Criterion

The reactor is critical with the flux level established at approximately 1×10^{-8} amps on the intermediate range nuclear instrumentation.

14.2.8.2.40 Isothermal Temperature Coefficient Measurement Test

A. Objective

To determine isothermal temperature coefficient and then derive the moderator temperature coefficient from the isothermal data.

B. Prerequisites

1. The reactor is critical at zero power, with no-load temperature and pressure stable.

2. The reactivity computer has been installed.

C. Test Method

The isothermal temperature coefficient is determined by heating/cooling the RCS at a constant rate and plotting temperature versus reactivity. The moderator temperature may be derived from isothermal data.

D. Acceptance Criterion

The average of the measured values of the isothermal and the derived moderator temperature coefficient agrees with acceptance criteria given in the Westinghouse Startup Program Reference Document.

14.2.8.2.41 Initial Criticality and Low-Power Test Sequence

A. Objective

To define the sequence of tests and operations, beginning with initial criticality, which constitutes the low-power testing program.

B. Prerequisite

Plant system conditions are established as required by the individual test instruction within this sequence.

C. Test Method

Individual test instruction will establish the plant conditions required for initial criticality and during the low-power testing program following initial criticality.

D. Acceptance Criterion

Relevant acceptance criteria are provided in each of the test procedures.

14.2.8.2.42 Biological Shield Survey Test (See Table 14.2.1-2 for Unit 2)

A. Objectives

1. To document the radiation levels in accessible locations of the plant outside of the biological shield while at power.
2. To obtain baseline radiation levels for comparison with future measurements of level buildup with operation.

B. Prerequisites

1. Radiation survey instruments are calibrated.
2. Background radiation levels are measured in designated locations prior to initial criticality.
3. The plant is stable at the applicable power level.

C. Test Method

Gamma and neutron radiation dose rates at designated locations will be measured at 30, 50, 75, and 100 percent of reactor power.

D. Acceptance Criterion

Radiation levels have been demonstrated acceptable for full-power operation and are within the limits described in section 12.3.

14.2.8.2.43 Dynamic Response Test

A. Objectives

1. To verify during power range testing that stress analysis of essential nuclear steam supply system (NSSS) and BOP components under transient conditions is in accordance with design.
2. Points will be tested to resolve discrepancies from hot functional testing, to test modifications made since hot functional testing, and to test systems not tested during hot functional testing, such as main feedwater.

B. Prerequisites

1. Temporary instrumentation is installed, as required, to monitor the deflections of components under test.
2. Points are monitored and baseline data are established.

C. Test Method

Deflection measurements are recorded during various plant transients.

D. Acceptance Criteria

1. The movements due to flow-induced loads shall not exceed the stress analysis of the monitored points.
2. Flow-induced movements and loads will not cause malfunctions of plant equipment or instrumentation.

14.2.8.2.44 Steady-State Core Performance Test

This material has been incorporated into 14.2.8.2.8 and 14.2.8.2.30.

14.2.8.2.45 Remote Shutdown Test

A. Objectives

1. To demonstrate the capability of performing a safe plant shutdown.
2. To maintain the plant in a hot standby condition from outside the control room.

B. Prerequisites

1. Reactor power is greater than 20 percent.
2. The equipment and instrumentation associated with the remote shutdown panel are available.
3. Plant systems are in the normal operating mode, with the turbine-generator in operation.

C. Test Method

1. Transfer control from the control room to the remote shutdown panel.
2. Check the instrumentation, control interlocks, and alarms.

3. Perform a safe shutdown from the remote shutdown panel.
4. Demonstrate the capability to achieve and maintain the plant in a hot standby condition from the remote shutdown panel for a minimum of 30 min.

D. Acceptance Criteria

1. Transfer of control from the control room to the remote shutdown panel is achieved.
2. Remote shutdown instrumentation, controls, and interlocks function.
3. The ability to perform a safe shutdown and to achieve and maintain hot standby conditions from the remote shutdown panel has been demonstrated.

14.2.8.2.46 Loss of Offsite Power at Greater Than 10-Percent Power Test

A. Objective

To demonstrate plant response following a plant trip with no offsite power available.

B. Prerequisites

1. The reactor will be operated at approximately 10-to 20-percent power with the reserve auxiliary transformers supplying station electrical load.
2. The test will occur prior to ascending from the nominal 20-percent power plateau.

C. Test Method

1. Manually start equipment, as required, for protection of nonsafety-related secondary equipment.
2. Isolate the reserve auxiliary transformers.
3. Verify reactor trip.
4. Verify automatic starting and automatic loading of diesel generators.
5. Verify that natural circulation conditions have been established.
6. Verify that the RCS can be maintained in a stable shutdown condition utilizing the power-operated atmospheric relief valves to remove decay heat.
7. Certain nonsafety-related, secondary plant equipment may be immediately restarted.

D. Acceptance Criteria

1. The automatic transfer from the reserve auxiliary transformers to the onsite standby power supply is demonstrated.
2. The primary plant can be maintained in a stable condition on natural circulation using the battery and diesel power supplies for 30 minutes.

14.2.8.2.47 Natural Circulation Test (See Table 14.2.1-2 for Unit 2)

A. Objectives

1. To demonstrate the capability to remove decay heat by natural circulation.

2. To demonstrate that primary parameters can be controlled within acceptable limits.

B. Prerequisites

1. Reactor coolant pumps are operating.
2. Primary system is at normal operating temperature and pressure.

C. Test Method

1. The test will be initiated by tripping off all reactor coolant pumps.
2. Natural circulation will be verified by observing the response of the hot leg and cold leg temperature instrumentation in each loop for natural circulation stabilization period and the ability to maintain the cooling mode.

3. Core exit thermocouples will be monitored to assess core flow distribution.

D. Acceptance Criteria

1. Natural circulation must occur in the primary system, and primary temperatures and pressures are within the design limits of subsection 5.1.2.
2. Delta T across the reactor core is less than the full-power delta T.
3. Decay heat removal capability is demonstrated by maintaining natural circulation conditions for a minimum of one hour.

14.2.8.2.48 Thermal Expansion Test

A. Objective

To demonstrate that essential NSSS and BOP components can expand without obstruction and that the expansion is in accordance with design. Also, during cooldown the components return to their approximate baseline cold position. Testing will be conducted to resolve discrepancies from hot functional testing and to test modifications made since hot functional testing was completed. Systems not tested during hot functional will be tested (e.g., main feedwater).

B. Prerequisite

Temporary instrumentation is installed, as required, to monitor the deflections for the components under test.

C. Test Method

For the components being tested the following will apply:

1. During plant heatup and cooldown, deflection data are recorded.

2. Snubber movements are verified by recording hot and cold positions.

D. Acceptance Criteria

For the components being tested, the following will apply:

1. There shall be no evidence of blocking of the thermal expansion of any piping or component, other than by installed supports, restraints, and hangers (paragraph 3.9.B.2.1).
2. Spring hanger movements must remain within the hot and cold setpoints, and snubbers must not become fully retracted or extended.
3. Piping and components must return to their approximate baseline cold position.

14.2.8.2.49 Primary and Secondary Chemistry Test

A. Objective

To verify proper water quality in the RCS and secondary coolant system.

B. Prerequisite

The plant is at steady state condition at approximately 0-, 30-, 50-, 75-, and 100-percent power.

C. Test Method

Samples will be taken and analyzed to determine the chemical and radiochemical concentrations.

D. Acceptance Criterion

The chemical and radiochemical control systems are capable of maintaining the water chemistry within limits specified by the manufacturer and listed in the Technical Specifications.

14.2.8.2.50 Power Ascension Test Sequence

A. Objective

To define the sequence of operations, beginning at approximately 5-percent power, which constitutes the power ascension testing program.

B. Prerequisite

Plant system conditions are established, as required, by the individual test instruction within this sequence.

C. Test Method

The sequence of operations and tests are presented along with detailed instructions, specific plant conditions, and test procedures.

D. Acceptance Criteria

Relevant acceptance criteria are provided in each of the test procedures.

14.2.8.2.51 (Material Deleted)

14.2.8.2.52 Large Load Reduction Test

A. Objectives

1. To demonstrate satisfactory plant transient response to various specified large-load reductions.
2. To monitor the RCS during these transients.
3. To optimize the RCS setpoints, if necessary.

B. Prerequisite

The plant is operating in a steady-state condition at the desired power level.

C. Test Method

1. The turbine-generator output is manually reduced to achieve an approximate 50-percent load reduction.
2. Selected plant parameters are monitored and recorded during the load transients.
3. If necessary, RCS setpoints are adjusted until optimal response is obtained.

D. Acceptance Criterion

The control systems, with no manual intervention, maintain reactor power, RCS temperature, pressurizer level and pressure, and steam generator levels and pressures without causing a reactor or turbine trip, or lift primary or secondary safety valves during, or following, the transient operation.

14.2.8.2.53 Plant Trip from 100-Percent Power Test

A. Objectives

1. To verify the ability of the plant automatic control systems to sustain a trip from 100-percent power and to bring the plant to stable conditions following the transient.
2. To determine the overall response time of the hot leg RTD.
3. To optimize the control systems setpoints, if necessary.

B. Prerequisite

The plant is operating in a steady-state condition at full power.

C. Test Method

1. The turbine-generator is tripped by opening the generator main breakers.
2. Selected plant parameters are monitored and recorded.
3. If necessary, the control systems setpoints are adjusted to obtain optimal response.

D. Acceptance Criteria

1. Following a 100-percent load rejection, primary and secondary control systems and operator actions can stabilize RCS temperature, pressurizer pressure and level, and steam generator levels to no-load operating temperature and pressure.

2. The steam dump control system operates to prevent opening of primary and secondary safety valves.

14.2.8.2.54 (This test deleted)

14.2.8.2.55 Plant Performance

A. Objectives

1. To monitor secondary plant systems under loaded conditions during power ascension testing.

2. To make adjustments to plant secondary systems, as necessary, to improve plant efficiency.

B. Prerequisites

1. The plant is at the specified power level.

2. The applicable secondary systems are in service.

C. Test Method

1. This procedure provides for the monitoring and collection of data to ensure that the plant is operated in the most efficient manner.

2. This procedure provides for minor adjustments to be made to secondary systems to obtain greater efficiency.

D. Acceptance Criteria

No acceptance criteria are required.

14.2.8.2.56 Nuclear Steam Supply System Acceptance Test

A. Objectives

1. To demonstrate the reliability of the NSSS by maintaining the plant at rated output (+0 percent, -5 percent) for 100 h without a load reduction resulting from an NSSS malfunction.

2. To measure the NSSS output and compare with its warranted rating.

B. Prerequisite

The plant is in normal operating conditions at full power (+0 percent, -5 percent).

C. Test Method

The plant will be maintained at steady-state conditions with NSSS thermal output between 95 percent and 100 percent. Selected plant parameters are monitored and recorded.

D. Acceptance Criteria

1. The reliability of the NSSS has been demonstrated by operating the plant at 95 to 100 percent of the rated thermal output for 100 h without a load reduction resulting from an NSSS malfunction.

2. The NSSS is capable of developing the warranted output.

14.2.8.2.57 At-Power Intercomparison of Reactor Protection System Inputs and Plant Computer Outputs Test

A. Objective

To demonstrate that the inputs to the reactor protection system and the plant computer are in satisfactory agreement with one another.

B. Prerequisites

1. The plant is at steady-state conditions at the applicable power level (30, 50, 75, and 100 percent).

2. Permanently installed instrumentation is calibrated and operable.

C. Test Method

1. Take simultaneous readings of the inputs to the reactor protection system, as read from the main control board or other accessible locations, and the output of the plant computer.

2. Evaluate the readings taken.

D. Acceptance Criterion

The inputs of the reactor protection system and outputs of the plant computer agree within the combined accuracy of the individual instruments.

14.2.8.2.58 Ventilation Capability Test

A. Objective

To verify that various HVAC systems for the containment and areas housing ESF continue to maintain design temperatures.

B. Prerequisite

The plant is operating at or near the desired power (0, 50, and 100 percent).

C. Test Method

1. Record temperature readings in specified areas while operating with normal ventilation lineups.

2. Record temperature readings in specified areas while operating the designed minimum number of HVAC components consistent with existing plant conditions.

3. Record surface concrete temperatures adjacent to the high temperature piping penetrations and at selected locations on the concrete shielding (at 100-percent power only).

4. Record ESF room cooler temperatures and compute the heat transfer capability (UA).

D. Acceptance Criteria

1. Temperature conditions are maintained in the containment, and ESF areas in accordance with table 9.4.1-2.

2. Concrete surface temperatures are maintained in accordance with paragraph 9.4.6.1.

3. Verify that the UA for each ESF room cooler has the required capacity.(4)

14.2.8.2.59 Gross Failed Fuel Detector Test (See Table 14.2.1-2 for Unit 2)

A. Objective

1. Calibrate the gross-failed fuel detector.

2. To establish baseline activity levels.

B. Prerequisites

1. Required electrical power supplies and control circuits are energized and operational.

2. A neutron source is available.

C. Test Method

1. Using a neutron source, the detector is calibrated and alarms checked.

2. At specified power levels (25 percent, 100 percent) baseline activity levels are recorded.

D. Acceptance Criteria

1. The gross-failed fuel detector is calibrated in accordance with the Westinghouse technical manual.

2. Base line activity levels are established.

---

[4] Should a cooling coil fall below rated capacity, water and/or air flow will be raised through the coil to required capacity.

14.2.8.2.60 Ultimate Heat Sink Heat Rejection Capability Test (See Table 14.2.1-2 for Unit 2)

A. Objectives

1. To determine the heat rejection capabilities of the NSCW cooling towers while operating under partial heat load conditions.
2. To evaluate the heat rejection capabilities of the NSCW cooling towers determined under test conditions and demonstrate that they meet the design requirements.

B. Prerequisites

1. The NSCW systems are operable.
2. The plant is operated in a manner to provide the NSCW system with a heat load.

C. Test Method

Heat rejection capability of NSCW cooling tower cells will be measured in accordance with CTI Bulletin ATC-105.

D. Acceptance Criteria

The heat rejection capability of the NSCW cooling towers, based on test conditions, is in accordance with design requirements.

TABLE 14.2.1-1 (SHEET 1 OF 7)

PREOPERATIONAL TEST PROCEDURES

Title	Test Abstract FSAR Paragraph
Main steam system	14.2.8.1.1(1)
Steam dump	14.2.8.1.2(2)
Condensate system	14.2.8.1.3(3)
Main feedwater system	14.2.8.1.4(1)
Motor-driven auxiliary feedwater (AFW) system	14.2.8.1.5
Turbine-driven AFW system	14.2.8.1.6
Reactor coolant system (RCS)	14.2.8.1.7
Pressure relief tank	14.2.8.1.8
RCS hydrostatic	14.2.8.1.9
RCS leak rate	14.2.8.1.10
Pressurizer pressure and level control	14.2.8.1.11
Reactor coolant pump initial operation	14.2.8.1.12
RCS hot functional	14.2.8.1.13(10)
Reactor internals and RCS vibration	14.2.8.1.14
Resistance temperature detectors/ thermocouple cross-calibration	14.2.8.1.15
Resistance temperature detector bypass flow measurements	14.2.8.1.16
Residual heat removal system	14.2.8.1.17
Chemical and volume control system	14.2.8.1.18
Letdown system	14.2.8.1.19
Boron thermal regeneration system	14.2.8.1.20(13)

TABLE 14.2.1-1 (SHEET 2 OF 7)

Title	Test Abstract FSAR Paragraph
Boric acid blender	14.2.8.1.21[(5)]
Safety injection system	14.2.8.1.22
Safety injection check valve	14.2.8.1.23
Safety injection accumulator	14.2.8.1.24
Boron Injection Tank and recirculation test (test deleted)	14.2.8.1.25
Containment spray system	14.2.8.1.26
Reactor makeup water storage tank and degasifier system	14.2.8.1.27
Refueling water storage tank	14.2.8.1.28
Condenser air ejection system	14.2.8.1.29[(3)]
Circulating water system	14.2.8.1.30[(3)]
Spent fuel pool cooling and cleanup system	14.2.8.1.31
Auxiliary component cooling water	14.2.8.1.32
Nuclear service cooling water	14.2.8.1.33
Component cooling water	14.2.8.1.34
AFW pumphouse heating, ventilation, and air-conditioning (HVAC)	14.2.8.1.35
Fuel handling building HVAC system	14.2.8.1.36[(4)]
Essential chilled water system	14.2.8.1.38
Control building HVAC	14.2.8.1.39[(5)]
Auxiliary building HVAC	14.2.8.1.40
Diesel generator building HVAC	14.2.8.1.41

TABLE 14.2.1-1 (SHEET 3 OF 7)

Title	Test Abstract FSAR Paragraph
Containment heat removal system	14.2.8.1.42
Containment integrated leak rate test	14.2.8.1.43(7)
Control rod drive mechanism (CRDM) cavity and vessel support cooling system	14.2.8.1.44
Post-loss-of-coolant accident purge exhaust	14.2.8.1.45
Hydrogen monitor and removal system	14.2.8.1.46
Containment air purification cleanup system	14.2.8.1.47
Gaseous waste processing system	14.2.8.1.48(5)
Liquid waste processing system	14.2.8.1.49(5)(14)
Waste evaporator	14.2.8.1.50(11)
Backflushable filter system	14.2.8.1.51
Boron recycle system	14.2.8.1.53(4)
Containment, auxiliary, control, and fuel handling buildings drains system	14.2.8.1.54(1)(4)(15)
Diesel generator fuel oil system	14.2.8.1.55
Service air system	14.2.8.1.56(2)
Instrument air system	14.2.8.1.57
Fire protection system	14.2.8.1.58(5)
Fuel handling and vessel servicing	14.2.8.1.59(5)
Fuel building hoists and elevator	14.2.8.1.60(4)(5)
Fuel transfer system	14.2.8.1.61
Refueling machine	14.2.8.1.62

TABLE 14.2.1-1 (SHEET 4 OF 7)

Title	Test Abstract FSAR Paragraph
Refueling machine indexing	14.2.8.1.63
Diesel generator	14.2.8.1.64
Auxiliary building flood retaining rooms alarm system	14.2.8.1.65
Main and unit auxiliary transformers	14.2.8.1.66(3)
Reserve auxiliary transformers	14.2.8.1.67(3)
Non-Class 1E ac distribution	14.2.8.1.68(3)
125 V dc (non-Class 1E)	14.2.8.1.69(3)
4.16 kV (Class 1E)	14.2.8.1.70
Class 1E standby power supply (4.16-kV diesel generator sequencer)	14.2.8.1.71
480 V (Class 1E) switchgear	14.2.8.1.72
480 V (Class 1E) motor control center	14.2.8.1.73
125 V dc Class 1E	14.2.8.1.74
Vital 120-V-ac Class 1E	14.2.8.1.75
Essential lighting system	14.2.8.1.76
Emergency lighting system	14.2.8.1.77
Offsite communication system	14.2.8.1.78(3)
Inplant communication system	14.2.8.1.79(3)
Heat tracing system	14.2.8.1.80(3)
(Material deleted)	14.2.8.1.81
Plant annunciator system	14.2.8.1.82
Emergency response facility and post-accident monitoring and sampling systems	14.2.8.1.83(1)(5)

TABLE 14.2.1-1 (SHEET 5 OF 7)

Title	Test Abstract FSAR Paragraph
Reactor protection system and engineered safety features actuation system (ESFAS) logic	14.2.8.1.84
Safeguards test cabinet testing capability	14.2.8.1.85
Nuclear instrumentation system	14.2.8.1.86
Process and effluent radiological monitoring system	14.2.8.1.87
Incore instrumentation system	14.2.8.1.88
Reactor control, rod control, and digital rod position indication	14.2.8.1.89
Control rod drive mechanism (CRDM) motor generator set	14.2.8.1.90
CRDM initial timing	14.2.8.1.91
Seismic monitoring system	14.2.8.1.92[(4)]
Nuclear sampling system	14.2.8.1.93
Miscellaneous leak detection system	14.2.8.1.94[(2)]
Metal impact monitoring system	14.2.8.1.95[(6)]
Integrated control logic safety injection	14.2.8.1.96[(10)]
Integrated safeguards and blackout sequence	14.2.8.1.97
ESFAS master and slave relay (test deleted)	14.2.8.1.98
Containment local leak rate	14.2.8.1.99
Reactor containment structural integrity test	14.2.8.1.100
High-efficiency particulate air filters	14.2.8.1.101[(1)]
Emergency core cooling system (ECCS) sump	14.2.8.1.102

TABLE 14.2.1-1 (SHEET 6 OF 7)

Title	Test Abstract FSAR Paragraph
Thermal expansion	14.2.8.1.103
Power conversion and ECCS dynamics	14.2.8.1.104
Remote shutdown	14.2.8.1.105
Reactor trip system and ESFAS response time	14.2.8.1.106
Extraction steam	14.2.8.1.107[(3)]
Condensate and feedwater chemical injection system	14.2.8.1.108[(2)]
Proteus computer	14.2.8.1.109[(9)]
Equipment building HVAC and piping penetration (test deleted)	14.2.8.1.110
Steam generator blowdown processing system	14.2.8.1.111
Main turbine system preoperational test	14.2.8.1.112
125-V dc, Class 1E minimum load voltage verification	14.2.8.1.113
Steady State Vibration Monitoring of Safety Related and High Energy Piping	14.2.8.1.114
AMSAC (ATWS Mitigating System Actuation Circuitry) System	14.2.8.1.115[(12)]

The notes below identify how the test abstracts are being implemented on Unit 2. These notes also identify where acceptance tests will be used on Unit 2 instead of a preoperational test as identified in chapter 14 and other chapters of the FSAR.

(1) This test abstract will be performed on Unit 2 via preoperational test(s) on the safety-related portions and acceptance tests(s) on the nonsafety related portions.

(2) This test abstract will be performed as an acceptance test on Unit 2. Any safety-related item(s) in this test abstract will be performed in another preoperational test.

TABLE 14.2.1-1 (SHEET 7 OF 7)

(3) This test abstract will be performed as an acceptance test on Unit 2.

(4) The system(s) covered by this test abstract are common to Units 1 & 2 and were tested during the Unit 1 preoperational test program. Interfaces, if any, with Unit 2 will be covered in the Unit 2 preoperational or acceptance test of the system that has the interface.

(5) The system(s) covered by this test abstract are common to Units 1 & 2. The portions of the system(s) tested during Unit 1 testing will not be repeated during the Unit 2 preoperational test.

(6) Parts of this test abstract will be performed as part of 14.2.8.1.13, RCS Hot Functional Preoperational Test Procedure, and will not have a unique preoperational test as on Unit 1. The initial calibration of the metal impact monitor, items A.1., C.1, and D.1 of 14.2.8.1.95, in preparation for hot functional testing, will be accomplished via vendor checkout.

(7) This test abstract may be performed as a surveillance procedure under the controls of the preoperational test program.

(8) The NPSH test and ultimate heat sink rejection capability test required by this test abstract do not have to be performed on Unit 2.

(9) The portions of this test abstract concerning the Internal Application Programs will not be performed on Unit 2 for those programs that were successfully debugged and unchanged between Units. (i.e., image tape of Unit 1 master tape used on Unit 2).

(10) The testing described in items A3, C4, and D4 of 14.2.8.1.96 will be performed during conduct of 14.2.8.1.13 HFT for Unit 2.

(11) The waste evaporator test will be performed prior to equipment use, but is not scheduled prior to Unit 2 OL.

(12) Since this design modification was not incorporated into Unit 1 prior to fuel load, this test abstract is applicable to Unit 2 only.

(13) On Unit 2, the Units 1 and 2 common chiller will be used to verify system performance during HFT. The Unit 2 chiller performance will be verified by use of an external heat source to simulate HFT condition.

(14) The portion of the preoperational test procedure which demonstrates the capability to transfer resin will not be performed on Unit 2 since the capability to transfer resin on similar equipment was demonstrated on Unit 1.

(15) Flow-path verification will not be accomplished for flow discharge paths that result in discharging to a common system that is contaminated and could result in increasing contamination in Unit 2.

TABLE 14.2.1-2 (SHEET 1 OF 4)

STARTUP TEST PROCEDURES

Title	Test Abstract FSAR Paragraph
RCS final leak test	14.2.8.2.1
Pressurizer heater and spray capability and continuous spray flow verification test	14.2.8.2.2[(2)]
RCS flow measurement	14.2.8.2.3
Resistance temperature detector (RTD) bypass valve flow measurement	14.2.8.2.4
RCS flow coastdown	14.2.8.2.5
Reactor protection	14.2.8.2.6
Core loading instrumentation and neutron source requirements	14.2.8.2.7
Thermal power measurement and statepoint data collection	14.2.8.2.8
Incore detector test	14.2.8.2.9
Operational alignment of nuclear instrumentation	14.2.8.2.10
Material deleted	14.2.8.2.11
Rod control system	14.2.8.2.12
CRDM operational test	14.2.8.2.13
Rod drop time measurement	14.2.8.2.14
Rod position indication	14.2.8.2.15
Operational alignment of process temperature instrumentation	14.2.8.2.16

TABLE 14.2.1-2 (SHEET 2 OF 4)

Title	Test Abstract FSAR Paragraph
Startup adjustments of RCS	14.2.8.2.17
RCS sampling for core loading	14.2.8.2.18
Metal impact monitoring system	14.2.8.2.19
Initial fuel loading	14.2.8.2.20
Inverse count rate ratio monitoring for fuel loading	14.2.8.2.21
Determination of core power range for physics testing	14.2.8.2.22
Precritical test sequence	14.2.8.2.23
Dynamic automatic steam dump control	14.2.8.2.24
Automatic steam generator level control	14.2.8.2.25
Power coefficient determination	14.2.8.2.26
Load swing	14.2.8.2.27
Process and effluent radiation monitoring	14.2.8.2.28
Axial flux difference instrumentation calibration	14.2.8.2.29
Flux map	14.2.8.2.30
Automatic reactor control system	14.2.8.2.31
(Material deleted)	14.2.8.2.32
(Material deleted)	14.2.8.2.33
Pseudo rod ejection	14.2.8.2.34(3)
RCCA and bank worth measurement at zero power	14.2.8.2.35(4)

TABLE 14.2.1-2 (SHEET 3 OF 4)

Title	Test Abstract FSAR Paragraph
Control system checkout (test deleted)	14.2.8.2.36
Boron endpoint determination	14.2.8.2.37
Inverse count rate ratio monitoring for the approach to initial criticality	14.2.8.2.38
Initial criticality	14.2.8.2.39
Isothermal temperature coefficient measurement	14.2.8.2.40
Initial criticality and low-power test sequence	14.2.8.2.41
Biological shield survey	14.2.8.2.42(5)
Dynamic response	14.2.8.2.43
Steady-state core performance test (test deleted)	14.2.8.2.44
Remote shutdown	14.2.8.2.45
Loss of offsite power at greater than 10-percent power	14.2.8.2.46
Natural circulation	14.2.8.2.47(3)
Thermal expansion	14.2.8.2.48
Primary and secondary chemistry	14.2.8.2.49
Power ascension test sequence	14.2.8.2.50
Material deleted	14.2.8.2.51
Large load reduction	14.2.8.2.52
Plant trip from 100-percent power	14.2.8.2.53
Steam generator moisture carryover	14.2.8.2.54

TABLE 14.2.1-2 (SHEET 4 OF 4)

Title	Test Abstract FSAR Paragraph
Plant performance	14.2.8.2.55
Nuclear steam supply system acceptance test	14.2.8.2.56
At-power intercomparison of reactor protection system inputs and plant computer output test	14.2.8.2.57
Ventilation capability test	14.2.8.2.58(7)
Gross failed fuel detector test	14.2.8.2.59(6)
Ultimate heat sink heat rejection capability test	14.2.8.2.60(1)

(1) The NPSHS test and ultimate heat sink rejection capability test required by this test abstract do not have to be performed on Unit 2

(2) Units 1 and 2 are essentially identical plants and it is not necessary to repeat all of this test abstract on Unit 2 when no significant new data or test results would be obtained. The portions (A.3, C.4, and C.5) relating to natural circulation will not be performed on Unit 2.

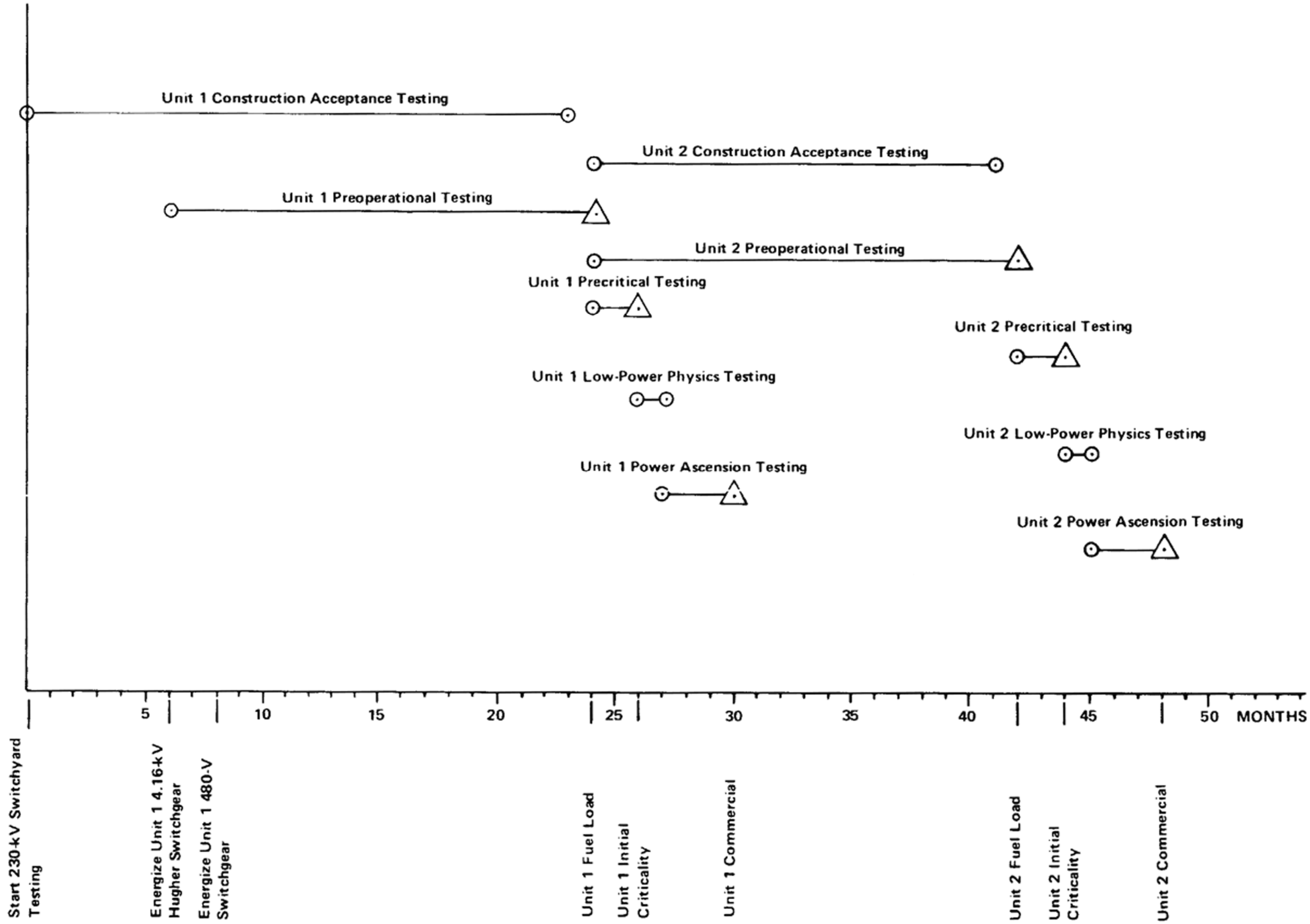
(3) Units 1 and 2 are essentially identical plants and the extremely close agreement between the predicted data and the Unit 1 test results demonstrates that the Unit 2 behavior can be accurately predicted. Performing this test abstract will not result in obtaining any significant new data or test results. This test abstract will not be performed on Unit 2.

(4) Units 1 and 2 are essentially identical plants and the extremely close agreement between the predicted data and the Unit 1 test results demonstrates that the Unit 2 behavior can be accurately predicted. Performing this test abstract will not result in obtaining any significant new data or test results. Acceptance criterion D.2 concerning pseudo-ejected rod will not be performed on Unit 2.

(5) Radiation surveys are required at the 50 % and 100 % power only. Radiation surveys at the 30 % and 75 % power plateau will not be performed on Unit 2.

(6) Calibration with a neutron source will not be performed for Unit 2. This is accomplished per preoperational test abstract 14.2.8.1.93. A calibration check will be performed instead.

(7) Test method C.4 and acceptance criteria D.3 will not be performed on Unit 2. The design similarity does not require that the UA be reverified.



REV 14 10/07

SOUTHERN COMPANY
Energy to Serve Your World®

VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

INITIAL TEST
PROGRAM MILESTONE SCHEDULE

FIGURE 14.2.7–1

15.0 ACCIDENT ANALYSES

15.0.1 CLASSIFICATION OF PLANT CONDITIONS

Since 1970 the American Nuclear Society classification of plant conditions has been used; this classification divides plant conditions into four categories according to anticipated frequency of occurrence and potential radiological consequences to the public. The four categories[1] are as follows:

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

The basic principle applied in relating design requirements to each of the conditions is that the most probable occurrences should yield the least radiological risk to the public, and those extreme situations having the potential for the greatest risk to the public should be those least likely to occur. Where applicable, reactor trip and engineered safeguards functioning is assumed to the extent allowed by considerations such as the single-failure criterion in fulfilling this principle; i.e., only Seismic Category 1, Class 1E, and qualified equipment, instrumentation, and components are used in the ultimate mitigation of the consequences of faulted Conditions II, III, and IV events. Step-by-step sequence of events diagrams are provided for each transient in figures 15.0.1-1 through 15.0.1-24. Figure 15.0.1-25 provides the legend used in these diagrams. The accident analysis radiological consequences evaluation models and parameters are discussed in appendix 15A. The analyses presented in this chapter are based on reload core transition to VANTAGE 5 fuel at uprated power.

15.0.1.1 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 which 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 described is generally based on a conservative set of initial conditions corresponding to adverse conditions which can occur during Condition I operation.

A typical list of Condition I events follows:

A. Steady-State and Shutdown Operations

1. Power operation (>5 to 100 percent of rated thermal power).

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[1] . For the definition of Conditions I, II, III, and IV events, refer to American National Standards Institute N18.2, Nuclear Safety Criteria for the Design of Stationary PWR Plants, 1973.

2. Startup ($k_{eff} \geq 0.99$, ≤5 percent of rated thermal power).
3. Hot standby (subcritical, residual heat removal (RHR) system isolated, $T_{avg} \geq 350°F$).
4. Hot shutdown (subcritical, RHR system in operation, $200°F < T_{avg} < 350°F$).
5. Cold shutdown (subcritical, RHR system in operation, $T_{avg} \leq 200°F$).
6. Refueling (subcritical, RHR system in operation, $T_{avg} \leq 140°F$).

B. Operation with Permissible Deviations

Various deviations that may occur during continued operation as permitted by the plant technical specifications must be considered in conjunction with other operational modes. These include:

1. Operation with components or systems out of service (such as an inoperable rod cluster control assembly (RCCA)).
2. Leakage from fuel with limited clad defects.
3. Excessive radioactivity in the reactor coolant:
   a. Fission products.
   b. Corrosion products.
   c. Tritium.
4. Operation with steam generator tube leaks.
5. Testing.

C. Operational Transients

1. Plant heatup and cooldown.
2. Step load changes (up to ±10 percent).
3. Ramp load changes (up to 5 percent/min).
4. Load rejection up to and including design full load rejection transient.

15.0.1.2 CONDITION II: FAULTS 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 rod failures, reactor coolant system failures, or secondary system overpressurization.

For the purposes of this report, the following faults are included in this category:

- Feedwater system malfunctions that result in a decrease in feedwater temperature (subsection 15.1.1).
- Feedwater system malfunctions that result in an increase in feedwater flow (subsection 15.1.2).

- Excessive increase in secondary steam flow (subsection 15.1.3).
- Inadvertent opening of a steam generator relief or safety valve (subsection 15.1.4).
- Loss of external electrical load (subsection 15.2.2).
- Turbine trip (subsection 15.2.3).
- Inadvertent closure of main steam isolation valves (subsection 15.2.4).
- Loss of condenser vacuum and other events resulting in turbine trip (subsection 15.2.5).
- Loss of nonemergency ac power to the station auxiliaries (subsection 15.2.6).
- Loss of normal feedwater flow (subsection 15.2.7).
- Partial loss of forced reactor coolant flow (subsection 15.3.1).
- Uncontrolled RCCA bank withdrawal from a subcritical or low power startup condition (subsection 15.4.1).
- Uncontrolled RCCA bank withdrawal at power (subsection 15.4.2).
- RCCA misalignment (dropped full-length assembly, dropped full-length assembly bank, or statically misaligned assembly) (subsection 15.4.3).
- Startup of an inactive reactor coolant pump at an incorrect temperature (subsection 15.4.4).
- Chemical and volume control system malfunction that results in a decrease in the boron concentration in the reactor coolant (subsection 15.4.6).
- Inadvertent operation of the emergency core cooling system (ECCS) during power operation (subsection 15.5.1).
- Chemical and volume control system malfunction that increases reactor coolant inventory (subsection 15.5.2).
- Inadvertent opening of a pressurizer safety or relief valve (subsection 15.6.1).
- Break in instrument line or other lines from reactor coolant pressure boundary that penetrate containment (subsection 15.6.2).

15.0.1.3 CONDITION III: INFREQUENT FAULTS

Condition III events are faults which may occur infrequently during the life of the plant. They may result in the failure of only a small fraction of the fuel rods. 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 containment barriers. The following faults are included in this category:

- Steam system piping failure (minor) (subsection 15.1.5).
- Complete loss of forced reactor coolant flow (subsection 15.3.2).
- RCCA misalignment (single RCCA withdrawal at full power) (subsection 15.4.3).
- Inadvertent loading and operation of a fuel assembly in an improper position (subsection 15.4.7).
- Loss-of-coolant accidents (LOCAs) resulting from a spectrum of postulated piping breaks within the reactor coolant pressure boundary (small break) (subsection 15.6.5).
- Radioactive gas waste system leak or failure (subsection 15.7.1).
- Radioactive liquid waste system leak or failure (subsection 15.7.2).
- Postulated radioactive releases due to liquid tank failures (subsection 15.7.3).
- Spent fuel cask drop accidents (subsection l5.7.5).

15.0.1.4 CONDITION IV: LIMITING FAULTS

Condition IV events are faults which are not expected to take place but are postulated because their consequences would include the potential of the release of significant amounts of radioactive material. They are the most drastic faults which must be designed against and they represent limiting design cases. Condition IV faults are not to cause a fission product release to the environment resulting in doses in excess of guideline values of l0 CFR 100. A single Condition IV event is not to cause a consequential loss of required functions of systems needed to cope with the fault, including those of the ECCS and the containment. For the purposes of this report the following faults have been classified in this category:

- Steam system piping failure (major) (subsection 15.1.5).
- Feedwater system pipe break (subsection 15.2.8).
- Reactor coolant pump shaft seizure (locked rotor) (subsection 15.3.3).
- Reactor coolant pump shaft break (subsection 15.3.4).
- Spectrum of RCCA ejection accidents (subsection 15.4.8).
- Steam generator tube failure (subsection 15.6.3).
- LOCAs resulting from the spectrum of postulated piping breaks within the reactor coolant pressure boundary (large break) (subsection 15.6.5).
- Design basis fuel handling accidents (subsection 15.7.4)

15.0.1.5 TMI ACTION PLAN (NUREG-0737) IMPLEMENTATION

In response to NUREG-0737 Item II.K.2.17 (Potential for Voiding in the RCS During Transients) the Westinghouse Owners Group has submitted a study to the Nuclear Regulatory Commission (NRC) which addresses the potential for void formation during natural circulation/cooldown transients. This study is applicable to VEGP. In addition, generic guidelines to preclude void formation in the upperhead were developed by Westinghouse Owners Group and submitted to the NRC. These guidelines will be used in the development of the VEGP specific procedures.

Following generic review, the NRC has determined that the concerns expressed in NUREG-0737 Item II.K.2.19 (Sequential Auxiliary Feedwater Flow Analysis) are not applicable to nuclear steam system suppliers with inverted U-tube steam generators. VEGP has these types of steam generators and therefore Item II.K.2.19 is not applicable to VEGP.

Westinghouse has submitted a new small break LOCA model to the NRC in response to NUREG-0737 Item II.K.3.30. This model is under NRC review and when approved by the NRC will complete Item II.K.3.30. Item II.K.3.31 (Plant Specific Analysis) will be addressed following completion of NRC review and approval of the Westinghouse new small break model that was submitted in response to NUREG-0737 Item II.K.3.30.

15.0.2 OPTIMIZATION OF CONTROL SYSTEMS

A control system setpoint study was performed prior to operation to simulate performance of the reactor control and protection systems. In this study, emphasis is placed on the development of a control and protection system which will automatically maintain prescribed conditions in the plant even under the most adverse set of anticipated plant operating transients with respect to both system stability and equipment performance. For each mode of plant operation, a group of optimum controller setpoints is determined. The control system setpoints developed are consistent with the nominal protection system setpoints used for accident analysis assumption.

The OTDT and OPDT reference temperature may be set higher than the indicated temperatures when considering instrumentation process errors.

The OTDT, OPDT, and control system reference temperatures are not permitted to exceed 588.4 °F. Instrumentation errors are calculated to be consistent with the method used in the accident analysis. These errors are applied in an adverse direction with respect to maintaining system stability and transient performance. The accident analysis and setpoint study combined produces a consistent set of control system parameters which show that the plant can be operated and meet both safety and operability requirements throughout the core life and for various levels of power operation.

The study is comprised of an analysis of the following control systems:

- Rod cluster control assembly,
- Steam dump,
- Steam generator level,
- Pressurizer pressure, and
- Pressurizer level.

The safety analyses and setpoints account for process effects on the hot leg temperature. The effects considered are:

- Burndown,
- Upper plenum flow anomaly,
- Loop-to-loop asymmetry, and
- A bias between the OTDT/OPDT reference temperature and the loop average temperatures.

The full power vessel temperature rise (ΔT_O) for each loop is used in the OTDT and OPDT setpoint equations. Due to process effects, the indicated ΔT can decrease or increase during the operating cycle. From a protection system point of view, the direction of nonconservatism that should be addressed is a decrease in the indicated ΔT. Changes in the increasing direction do not have to be addressed. However, such an increase can affect operational margin to trip. The analyses and setpoints account for a decrease of up to 3.4 °F. In addition, the setpoint uncertainty analyses assumed a minimum indicated full power loop ΔT of 55.8 °F. This parameter is an explicit input that determines the magnitude of the uncertainties in % ΔT span. If the minimum full power loop ΔT is found to be less than 55.8 °F, the uncertainty calculations should be evaluated. Indicated loop ΔT values greater than 55.8 °F at full power are conservative and do not require additional evaluation. The process effects and full power ΔT in each loop should be periodically monitored to ensure that the analysis assumptions are met and ΔT_O renormalized if necessary. An interval of 92 EFPD is recommended (Reference 1).

Similarly, the process effects can impact the indicated loop average temperatures ($T_{avg.}$). The full power loop average temperatures are used to determine the OTDT and OPDT protection reference temperatures (T’ and T” respectively) as well as the rod control system reference temperature. The safety analyses and setpoints account for a difference between the protection reference temperatures and the corresponding loop T_{avg} of up to 6.1 °F. The OTDT, OPDT, and control system reference temperatures are not permitted to exceed 588.4 °F. Within these limits, it is permissible to establish a common value for the individual loop reference temperatures for OTDT. Likewise, it is permissible to establish a common value for the individual loop reference temperatures for OPDT. The OTDT and OPDT reference temperatures can be set up to 2 °F above the control system reference temperature. The process effects and full power T_{avg} in each loop should be periodically monitored to ensure that the analysis assumptions are met and T’ and T” renormalized if necessary. An interval of 92 EFPD is recommended (Reference 1).

T_{avg} control for full power valves wide open (VWO) operations allows for the periodic adjustment of T_{avg}, within its analyzed range during the operating cycle. Unit startup from a refueling outage is accomplished on a preestablished beginning-of-cycle temperature program. Once full power conditions are established, if the control valves are not at VWO for 100% RTP, then valves may be gradually opened to VWO position while allowing RCS temperature to decrease within established temperature boundary limits. If the control valves are at VWO and 100% RTP cannot be achieved, then RCS temperature may be adjusted up within temperature boundary limits to ensure 100% RTP. At the point of VWO with RCS temperature established for 100% RTP, the control and protection systems must be recalibrated to establish this new VWO temperature program and ensure proper function within the analyzed plant response requirements. Steam dump controls (plant trip controller settings, load rejection controller settings, and trip open bistable settings) and the pressurizer spray line low temperature alarm settings should be conservatively reset to values corresponding to the lowest intended full load reference temperature for the cycle as defined by the T_{avg} low design limit provided in the cycle reload evaluation. The high auctioneered HI T_{avg} alarm setting should be reset to the alarm value corresponding to the new VWO T_{avg}. The control reference temperature used by the control system as well as the protection system reference temperatures T', T'', and ΔT_o are to be renormalized as necessary for VWO T_{avg}, and within the above described requirements.

Coastdowns at the end of an operating cycle may be performed by a power reduction on the normal temperature program (power coastdown), or by a combination of RCS temperature reduction (temperature coastdown) followed by a power coastdown. In the latter case, RCS temperature and power may initially be reduced by maintaining a constant turbine control valve position, and allowing the temperature feedback of the reactor core to control the rate of temperature and power reduction. If the valves are not at the valves wide open position when depletion of reactivity is reached at the end of an operating cycle, the valves can be gradually opened as temperature and power begin to decrease.

In order to perform a power coastdown on the normal temperature program, no specific adjustments to the control or protection system settings are required.

For the combination of a temperature coastdown followed by a power coastdown, the steam dump load rejection controller must be reset with trip open bistable and gain settings corresponding to a full-load reference temperature of no greater than the lowest intended full-load reference temperature. This ensures that the steam dumps provide adequate heat removal for load rejections as described in sections 7.7 and 10.4, and that the requirements of TMI Action III.K.3.10 are met for a turbine trip without a reactor trip below the P-9 setpoint as described in paragraph 7.2.1.1.2. This method of steam dump control may be used for any combination of temperature coastdown followed by a power coastdown within the analyzed range of temperature programs. No changes to the plant trip controller settings are required. To improve the reactor control system response to transients, and to provide the operators with a target temperature for manual control and trip recovery, the programmed reference temperature should be reset periodically during the temperature coastdown. Once a final temperature program is reached, no further changes are required during the subsequent power coastdown. The overtemperature (OTDT) and overpower (OPDT) setpoint reference temperatures may remain at their corresponding pre-coastdown settings for the duration of the coastdown.

15.0.2.1 REFERENCE

1&2X6AA02-00562, “Margin Recovery Technical Report.”

15.0.3 PLANT CHARACTERISTICS AND INITIAL CONDITIONS ASSUMED IN THE ACCIDENT ANALYSES

15.0.3.1 DESIGN PLANT CONDITIONS

Table 15.0.3-1 lists the principal power rating values assumed in analyses performed in this report. Two ratings are given:

A. The design nuclear steam supply system (NSSS) thermal power output. This power output includes the thermal power generated by the reactor coolant pumps.

B. The design core thermal power output. The NSSS is designed for thermal power higher than the design core value. This higher thermal power value is designated as the NSSS design rating. Most of the accident analyses assume the NSSS thermal power output; the remaining accident analyses assume the design core thermal power. Where initial power operating conditions are assumed in accident analyses, the appropriate thermal power output plus allowance for errors in steady-state power determination is assumed. The thermal power values used for each transient analyzed are given in table 15.0.3-2. In all cases where the NSSS rating is used in an analysis, the resulting transients and consequences are conservative compared to using the design core rating.

The values of other pertinent plant parameters utilized in the accident analyses are given in table 15.0.3-3.

15.0.3.2 INITIAL CONDITIONS

The analyses of departure from nucleate boiling (DNB) accidents use the Revised Thermal Design Procedure (RTDP) to define the initial conditions. Initial conditions for other accidents are obtained by applying the maximum steady-state errors to rated values (this procedure is commonly known as Standard Thermal Design Procedure or STDP). The following steady-state errors are considered in the analyses:

- Core power — 1.7% measurement uncertainty recapture power uprate and +0.3% power uncertainty
- Average reactor coolant system temperature — ±6°F allowance for deviation from programmed T_{avg} (includes measurement error)
- Pressurizer pressure — ±50 psi allowance for steady-state fluctuations and measurement errors

Accidents employing RTDP assume minimum measured flow (MMF); accidents employing STDP assume thermal design flow (TDF). Table 15.0.3-2 summarizes the initial conditions and computer codes used in the accident analyses.

The values of other pertinent plant parameters used in the accident analyses are given in table 15.0.3-3.

15.0.3.3 POWER DISTRIBUTION

The transient response of the reactor system is dependent on the initial power distribution. The nuclear design of the reactor core minimizes adverse power distribution through the placement of fuel assemblies and control rods. Power distribution may be characterized by the nuclear enthalpy rise hot channel factor $F_{\Delta}H$ and the total peaking factor (F_q). Unless specifically noted otherwise, the peaking factors used in the accident analyses are those presented in chapter 4.

For transients which may be departure from nucleate boiling (DNB) limited, the radial peaking factor is important. The radial peaking factor increases with decreasing power level due to rod insertion. This increase in $F_{\Delta}H$ is included in the core limits illustrated in figure 15.0.6-1. All transients that may be DNB limited are assumed to begin with an $F_{\Delta}H$ consistent with the initial power level defined in the Technical Specifications. The axial power shape used in the DNB calculation is the 1.55 chopped cosine, as discussed in paragraph 4.4.4.3 for transients analyzed at full power and the most limiting power shape calculated or allowed for accidents initiated at nonfull power or asymmetric rod cluster control assembly (RCCA) conditions.

The radial and axial power distributions described above are input to the VIPRE-01 code as described in paragraph 4.4.4.5.

For transients which may be overpower limited, the total peaking factor (F_q) is important. All transients that may be overpower limited are assumed to begin with plant conditions, including power distributions, which are consistent with reactor operation as defined in the Technical Specifications.

For overpower transients that are slow with respect to the fuel rod thermal time constant (for example, the chemical and volume control system malfunction that results in a slow decrease in the boron concentration in the reactor coolant system as well as an excessive increase in secondary steam flow) and that may reach equilibrium without causing a reactor trip, the fuel rod thermal evaluations are performed as discussed in subsection 4.4.4. For overpower transients that are fast with respect to the fuel rod thermal time constant (for example, the uncontrolled RCCA bank withdrawal from subcritical or low power startup and RCCA ejection incident, both of which result in a large power rise over a few seconds), a detailed fuel transient heat transfer calculation is performed.

15.0.4 REACTIVITY COEFFICIENTS ASSUMED IN THE ACCIDENT ANALYSES

The transient response of the reactor coolant system (RCS) depends on reactivity feedback effects, particularly the moderator temperature coefficient and the Doppler power coefficient. These reactivity coefficients are discussed in detail in paragraph 4.3.2.3.

In the analysis of certain events, conservatism requires the use of large reactivity coefficient values, whereas in the analysis of other events, conservatism requires the use of small reactivity coefficient values. Some analyses, such as loss of reactor coolant from cracks or ruptures in the reactor coolant system, do not depend on reactivity feedback effects. The values used are given in figure 15.0.4-1, which shows the upper and lower bound Doppler power coefficients as a function of power, used in the transient analysis. Figure 15.0.4-2 shows the moderator density coefficient (minimum and maximum), as a function of temperature, used in the transient analysis. The justification for use of conservatively large versus small reactivity coefficient values is treated on an event-by-event basis. In some cases conservative combinations of parameters are used to bound the effects of core life. For example, in a load increase transient it is conservative to use a small Doppler defect and a small moderator coefficient.

15.0.5 ROD CLUSTER CONTROL ASSEMBLY INSERTION CHARACTERISTICS

The negative reactivity insertion following a reactor trip is a function of the acceleration of the rod cluster control assemblies (RCCAs) and the variation in rod worth as a function of rod position.

With respect to accident analyses, the RCCA insertion from the start of insertion up to the dashpot entry, approximately 85 percent of the rod cluster travel, is assumed to be 2.7 seconds. The RCCA position versus time is shown on figure 15.0.5-1.

Figure 15.0.5-2 shows the fraction of total negative reactivity insertion versus normalized rod insertion. There is inherent conservatism in the use of this curve in that its basis is a bottomskewed axial power distribution. For cases other than those associated with axial xenon oscillations, the more favorable axial power distribution existing before trip results in significant negative reactivity to be inserted.

The normalized RCCA negative reactivity insertion versus time used in the safety analysis is shown on figure 15.0.5-3. The curve shown on this figure results from the combination of figure 15.0.5-1 and figure 15.0.5-2. The transient analyses, except where specifically noted otherwise, assume a total negative reactivity insertion following a trip of 4.0 percent $\Delta k/k$. This assumption is verified to be conservative with respect to the core design.

For analyses requiring the use of a multidimensional spatial neutron kinetics code (TWINKLE), the code directly calculates the negative reactivity insertion resulting from reactor trip which is not separable from other reactivity feedback effects. In this case, the code models the RCCA position versus time of figure 15.0.5-1.

15.0.6 TRIP POINTS AND TIME DELAYS TO TRIP ASSUMED IN ACCIDENT ANALYSES

A reactor trip signal opens two trip breakers connected in series, which feeds power to the control rod drive mechanisms. The loss of power to the mechanism coils causes the mechanisms to release the RCCAs which then fall by gravity into the core. There are various instrumentation delays associated with each trip function, including delays in signal actuation, in opening the trip breakers, and in the release of the rods by the mechanisms. The time delay from the time that the reactor reaches trip setpoint conditions to the time the rods are free to fall defines the total delay to trip. Limiting trip setpoints assumed in accident analyses and the time delay assumed for each trip function are given in table 15.0.6-1.

Table 15.0.6-1 refers to the overtemperature ΔT (OTΔT) and the overpower ΔT (OPΔT) reactor trip setpoints shown on figure 15.0.6-1. These trip setpoints bound the transition cores and a full core of VANTAGE 5 fuel. The associated OTΔT f(ΔI) penalty is shown on figure 15.0.6-2.

For all the reactor trips, the difference between the trip setpoints assumed in the analysis and the nominal trip setpoints account for instrumentation channel error and setpoint error. The plant Technical Specifications specify the nominal trip setpoints. The calibration of protection system channels and the determination of instrument response times are in accordance with the plant Technical Specifications.

15.0.7 INSTRUMENTATION DRIFT AND CALORIMETRIC ERRORS, POWER RANGE NEUTRON FLUX

The instrumentation drift and calorimetric errors used in establishing the power range high neutron flux setpoint are presented in table 15.0.7-1.

The calorimetric error is the error assumed in the determination of core thermal power as obtained from secondary plant measurements. The total ion chamber current (sum of the top and bottom sections) is calibrated (set equal) to this measured power on a periodic basis.

The secondary power is obtained from measurement of feedwater flow, feedwater inlet temperature to the steam generators, and steam pressure. High accuracy instrumentation is provided for these measurements with accuracy tolerances much tighter than those which would be required to control feedwater flow.

15.0.8 PLANT SYSTEMS AND COMPONENTS AVAILABLE FOR MITIGATION OF ACCIDENT EFFECTS

The plant is designed to afford proper protection against the possible effects of natural phenomena, postulated environmental conditions, and dynamic effects of the postulated accidents. In addition, the design incorporates features which minimize the probability and effects of fires and explosions. Chapter 17 discusses the quality assurance program which has been implemented to assure that the plant systems satisfactorily perform their assigned safety functions. The incorporation of these features in the plant, coupled with the reliability of the design, ensures that the normally operating systems and components listed in table 15.0.8-1 are available for mitigation of the events discussed in chapter 15. In determining which systems are necessary to mitigate the effects of these postulated events, the classification system of ANSI NI8.2-1973 is utilized. The design of systems important to safety (including protection systems) is consistent with IEEE Standard 379-1972 and Regulatory Guide 1.53 in the application of the single-failure criterion. Conformance to Regulatory Guide 1.53 is summarized in section 1.9.

In the analysis of the chapter 15 events, control system action is considered only if that action results in more severe accident results. No credit is taken for control system operation if that operation mitigates the results of an accident. For some accidents, the analysis is performed both with and without control system operation to determine the worst case. The pressurizer heaters are generally assumed not to be energized for the analysis of the chapter 15 events.

Operation of the pressurizer heaters as a result of normal control action or a single failure will be less conservative or have negligible effects for most analyses. Therefore, unless it is shown that such a control action results in more limiting results or more severe consequences, the control action of the pressurizer heaters is not modeled for the analyses performed in chapter 15.

In cases where loss of offsite power is postulated, a 12-s diesel start time is assumed.

Fission product barriers credited in the analyses of the events discussed are listed in table 15.0.8-2.

15.0.9 FISSION PRODUCT INVENTORIES

The fission product inventories considered to be available for release from the fuel in the event of fuel damage are identified in Appendix 15A.

15.0.10 RESIDUAL DECAY HEAT

15.0.10.1 TOTAL RESIDUAL HEAT

Residual heat in a subcritical core is calculated for the loss- of-coolant accident (LOCA) per the requirements of Appendix K of 10 CFR 50.46, as described in references 1 and 2. These requirements include assuming infinite irradiation time before the core goes subcritical to determine fission product decay energy. For all other accidents, the same models are used except that fission product decay energy is based on core average exposure at the end of the equilibrium cycle.

15.0.10.2 DISTRIBUTION OF DECAY HEAT FOLLOWING LOSS-OF-COOLANT ACCIDENT

During a LOCA, the core is rapidly shut down by void formation or rod cluster control assembly insertion, or both; and a large fraction of the heat generation to be considered comes from fission product decay gamma rays. This heat is not distributed in the same manner as steady-state fission power. Local peaking effects, which are important for the neutron-dependent part of the heat generation, do not apply to the gamma ray contribution. The steady-state factor of 97.4 percent, which represents the fraction of heat generated within the clad and pellet, drops to 95 percent for the hot rod in a LOCA.

For example, consider the transient resulting from the postulated double-ended break of the largest reactor coolant system pipe; 1/2 s after the rupture about 30 percent of the heat generated in the fuel rods is from gamma ray absorption. The gamma power shape is less peaked than the steady-state fission power shape, reducing the energy deposited in the hot rod at the expense of adjacent colder rods. A conservative estimate of this effect is a reduction of l0 percent of the gamma ray contribution or 3 percent of the total. Since the water density is considerably reduced at this time, an average of 98 percent of the available heat is deposited in the fuel rods; the remaining 2 percent is absorbed by water, thimbles, sleeves, and grids. The net effect is a factor of 0.95, rather than 0.974, to be applied to the heat production in the hot rod.

15.0.10.3 REFERENCES

1. Bordelon, F. M., et al., "SATAN-VI Program: Comprehensive Space-Time Dependent Analysis of Loss-of-Coolant," WCAP-8302 (Proprietary) and WCAP-8306 (Nonproprietary), June 1974.

2. Bordelon, F. M., et al., "LOCTA-IV Program: Loss-of-Coolant Transient Analysis," WCAP-8301 (Proprietary) and WCAP-8305 (Nonproprietary), June 1974.

15.0.11 COMPUTER CODES UTILIZED

Summaries of some of the principal computer codes used in transient analyses are given below. Other codes--in particular, very specialized codes in which the modeling has been developed to simulate one given accident, such as those used in the analysis of the reactor coolant system pipe rupture (section 15.6)--are summarized in their respective accident analyses sections. The codes used in the analyses of each transient have been listed in table 15.0.3-2.

15.0.11.1 FACTRAN COMPUTER CODE

FACTRAN calculates the transient temperature distribution in a cross-section of a metal clad UO_2 fuel rod and the transient heat flux at the surface of the clad, using as input the nuclear power and the time-dependent coolant parameters (pressure, flow, temperature, and density). The code uses a fuel model which contains the following features:

A. A sufficiently large number of radial space increments to handle fast transients such as rod ejection accidents.

B. Material properties which are functions of temperature and a sophisticated fuel-to-clad gap heat transfer calculation.

C. The necessary calculations to handle post-departure from nucleate boiling transients: film boiling heat transfer correlations, Zircaloy-water reaction, and partial melting of the fuel.

FACTRAN is further discussed in reference 1.

15.0.11.2 LOFTRAN COMPUTER CODE

The LOFTRAN program is used for studies of transient response of a pressurized water reactor system to specified perturbations in process parameters. LOFTRAN simulates a multiloop system by a model containing reactor vessel, hot and cold leg piping, steam generator (tube and shell sides), and pressurizer. The pressurizer heaters, spray, relief, and safety valves are also considered in the program. Point model neutron kinetics, and reactivity effects of the moderator, fuel, boron, and rods are included. The secondary side of the steam generator utilizes a homogeneous, saturated mixture for the thermal transients and a water level correlation for indication and control. The reactor protection system is simulated to include reactor trips on high neutron flux, overtemperature ΔT, overpower ΔT, high and low pressure, low flow, and high pressurizer level. Control systems are also simulated including rod control, steam dump, feedwater control, and pressurizer pressure control. The emergency core cooling system, including the accumulators, is also modeled.

LOFTRAN is a versatile program which is suited to both accident evaluation and control studies as well as parameter sizing.

LOFTRAN also has the capability of calculating the transient value of departure from nucleate boiling ratio (DNBR) based on the input from the core limits illustrated in figure 15.0.6-1. The core limits represent the minimum value of DNBR as calculated for typical or thimble cell.

LOFTRAN is further discussed in reference 2.

LOFTTR2 is a modified version of LOFTRAN with a more realistic break flow model, a 2-region SG secondary side, and an improved capability to simulate operator actions during a SGTR event. LOFTTR2 is further discussed in reference 4.

15.0.11.3 CROSS-SECTION GENERATION COMPUTER CODE

The lattice codes which have been used for the generation of group constants needed in the spatial, two-group diffusion codes are described in chapter 4.

15.0.11.4 SPATIAL TWO-GROUP DIFFUSION CALCULATION CODE

Spatial few-group diffusion calculations are described in chapter 4.

15.0.11.5 TWINKLE COMPUTER CODE

The TWINKLE program is a multidimensional spatial neutron kinetics code. The code uses an implicit finite-difference method to solve the two-group transient neutron diffusion equations in one, two, and three dimensions. The code uses six delayed neutron groups and contains a detailed multiregion fuel-clad-coolant heat transfer model for calculating pointwise Doppler and moderator feedback effects. The code handles up to 2000 spatial points and performs its own steady-state initialization. Aside from basic cross-section data and thermal-hydraulic parameters, the code accepts as input basic driving functions such as inlet temperature, pressure, flow, boron concentration, control rod motion, and others. The code provides various edits; e.g., channelwise power, axial offset, enthalpy, volumetric surge, pointwise power, and fuel temperatures.

The TWINKLE code predicts the kinetic behavior of a reactor for transients which cause a major perturbation in the spatial neutron flux distribution.

TWINKLE is further described in reference 3.

15.0.11.6 VIPRE-01 COMPUTER CODE

The VIPRE-01 code is described in chapter 4.

15.0.11.7 REFERENCES

1. Hargrove, H. G., "FACTRAN - A FORTRAN-IV Code for Thermal Transients in a UO_2 Fuel Rod," WCAP-7908-A, December 1989.
2. Burnett, T. W. T., et al., "LOFTRAN Code Description," WCAP-7907-A, April 1984.
3. Risher, D. H., Jr., and Barry, R. F., "TWINKLE - A Multi-Dimensional Neutron Kinetics Computer Code," WCAP-7979-P-A (Proprietary) and WCAP-8028-A (Nonproprietary), January 1975.
4. Lewis, R. N., et al., "SGTR Analysis Methodology to Determine the Margin to Steam Generator Overfill," WCAP-10750, August 1985 (Nonproprietary).
5. Lewis, R. N., et al., "SGTR Analysis Methodology to Determine the Margin to Steam Generator Overfill," WCAP-10750-A, August 1987 (Nonproprietary).

15.0.12 LIMITING SINGLE FAILURES

The most limiting single failure (where one exists), as described in section 3.1 of safety-related equipment, is identified in each analysis description; the consequences of this failure are described therein. In some instances, because of redundancy in protection equipment, no single failure which could adversely affect the consequences of the transient has been identified. The failure assumed in each analysis is listed in table 15.0.12-1.

15.0.13 OPERATOR ACTIONS

For most of the events analyzed in chapter 15, the plant will be in a safe and stable hot standby condition following the automatic actuation of reactor trip. This condition will, in fact, be similar to plant conditions following any normal, orderly shutdown of the reactor. At this point, the actions taken by the operator would be no different from normal operating procedures. The exact actions taken and the time at which these actions occur will depend on what systems are available (e.g., turbine bypass system, main feedwater system, etc.) and the plans for further plant operation. As a minimum, to maintain the stabilized hot standby condition, decay heat must be removed via the steam generators. The main feedwater system and the steam dump or atmospheric relief system could be used for this purpose. Alternatively, the auxiliary feedwater system and the steam generator atmospheric relief valves may be used, both of which are safety grade systems. Although the auxiliary system may be started manually, it will be automatically actuated, if needed, by one of the signals shown on drawings 1X6AA02-225, 1X6AA02-226, 1X6AA02-227, 1X6AA02-228, 1X6AA02-229, 1X6AA02-230, 1X6AA02-231, 1X6AA02-232, 1X6AA02-233, 1X6AA02-234, 1X6AA02-235, 1X6AA02-236, 1X6AA02-237, 1X6AA02-238, 1X6AA02-239, 1X6AA02-240, 1X6AA02-494, 1X6AA02-495, 1X6AA02-496, and 1X6AA02-519, such as low-low steam generator water level. Also, if the hot standby condition is maintained for an extended period of time (greater than approximately 18 h), operator action may be required to add boric acid via the chemical and volume control system to compensate for xenon decay and maintain shutdown margin.

Where a stabilized condition is reached automatically following a reactor trip and only actions typical of normal operation are required, this has been stated in the text of the chapter 15 events. For several events involving breaks in the reactor coolant system (RCS) or secondary system piping, additional requirements for operator action are identified.

Following the postulated main steam line break, a steam line isolation signal will be generated almost immediately, causing the main steam line isolation valves (MSIVs) to close within a few seconds. If the break is downstream of the MSIVs, all of which subsequently close, the break will be isolated. If the break is upstream of the MSIVs, the break will be isolated to three steam generators while the affected steam generator will continue to blow down. Only the case in which one steam generator continues to blow down is discussed here, since the break followed by isolation of all steam generators will terminate the transient.

Steam pressure from the steam generators is relieved by the turbine bypass system, secondary system atmospheric relief valves, or secondary system safety valves. The operator is instructed to terminate auxiliary feedwater flow to the affected steam generator as soon as he determines which steam generator is affected. As soon as an indicated water level returns to the pressurizer and pressure is no longer decreasing, the operator is instructed to terminate the charging pump flow to limit system repressurization.

For long-term cooling following a steam line break, the operator is instructed to use the intact steam generators to remove decay heat and plant stored energy. This is done by feeding the steam generators with auxiliary feedwater to maintain an indicated water level in the steam generator narrow-range span.

A safety injection signal (generated a few seconds after the break on low steam line pressure) will cause main feedwater isolation to occur. The only source of water available to the affected steam generator is then the auxiliary feedwater system. Following steam line isolation, steam pressure in the steam line with the affected steam generator will continue to fall rapidly, while the pressure stabilizes in the remaining three steam lines. The indication of the different steam pressures will be available to the operator within a few seconds of steam line isolation. This will provide the information necessary to identify the affected steam generator so that auxiliary feedwater to it can be isolated. Manual controls are provided in the control room for start and stop of the auxiliary feedwater pumps and for the control valves associated with the auxiliary feedwater system. Detecting the affected steam generator and isolating auxiliary feedwater to it requires only the use of safety grade equipment available following the break. The removal of decay heat in the long term (following the initial cooldown) using the remaining steam generators, requires only the auxiliary feedwater system as a water source and the secondary system relief valves to relieve steam.

The operator has available in the control room an indication of pressurizer water level from the reactor protection system instrumentation. Indicated water level returns to the pressurizer in approximately 5 to 7 min following the steam line break. To maintain the indicated water level, the operator can start and stop the charging pumps as necessary. The pressurizer level instrumentation and manual controls for the operation of the charging pumps meet the required standards for safety systems. The steam line break analysis does not model operator action to control pressurizer level by manipulating the safety injection flow. Minimum safety injection flow from one safeguards train is conservatively assumed throughout the transient.

As indicated, the information for terminating auxiliary feedwater to the affected steam generator is available to the operator within 1 min of the break. The requirements to terminate auxiliary feedwater flow to the affected steam generator can be met by switch actions by the operators, i.e., closing auxiliary feed discharge valve. Thus, the required actions to limit auxiliary feedwater delivery to the affected steam generator can be recognized, planned, and performed within 30 min. For decay heat removal and plant cooldown, the operator has a considerably longer time to respond because of the large initial cooldown associated with a steam line break transient.

For a feedwater line break, the required operator actions and times are discussed in subsection 15.2.8 and table 15.2.3-1. Auxiliary feedwater flow is initiated automatically, as is safety injection. As in the steam line break, the operator is expected to terminate auxiliary feedwater flow to the affected steam generator as soon as he determines which unit is affected, using safety grade equipment. Where possible, the operator should also increase auxiliary feedwater flow to the intact steam generators in order to shorten the time until primary temperatures begin to decrease. The analysis presented in subsection 15.2.8 does not model any specific delays associated with these actions and, therefore, does not directly impose any response time requirements on the operator.

As soon as primary temperature begins to decrease, the operator can use the steam dump system or the steam generator atmospheric relief valves to begin a controlled cooldown. In addition, if the U-tubes of the intact steam generators are covered with water as indicated by post-accident monitoring system (PAMS) steam generator water level instrumentation (chapter 7), the operator can modulate the high-head charging pumps, so that the primary pressure decreases while ensuring that voiding does not occur within the RCS. The primary pressure-temperature relationship can be monitored by the operator via the PAMS wide-range RCS pressure and temperature instruments.

Using the above-mentioned PAMS indications, the operator can maintain the plant in a hot shutdown condition for an extended period of time or can proceed to a cold shutdown condition as desired.

The safety-related indicators for steam line pressure and pressurizer water level noted above are further discussed in section 7.5.

Tables 15.0.13-1 and 15.0.13-2 list the short-term operator actions required to bring the plant to a stable condition for the loss-of-coolant accident (LOCA) and steam generator tube rupture (SGTR). Further information (including alarms which alert the operator) on operator action for these two accidents is given in paragraph 6.3.2.8 for the LOCA and subsection 15.6.3 for the SGTR.

Process information available to the operator in the control room following either of these accidents (LOCA or SGTR) is given in section 7.5.

Instrumentation and controls provided to allow the operator to complete required manual actions are classified as Class 1E. Electrical components are also classified as Class 1E. Mechanical components are classified as Safety Class 1, 2, or 3.

Safety systems required for accident mitigation are designed to function after the occurrence of the worst postulated single failure. There are no adverse impacts as a result of these actions.

TABLE 15.0.3-1

NUCLEAR STEAM SUPPLY SYSTEM POWER RATINGS

Parameter	Value
Design NSSS thermal power output (MWt)	3643
Minimum thermal power generated by the RCPs (MWt)	17
Maximum thermal power generated by the RCPs (MWt)	20
Maximum NSSS thermal power output (MWt)	3646
Design core thermal power (MWt)	3626

TABLE 15.0.3-2 (SHEET 1 OF 3)

SUMMARY OF INITIAL CONDITIONS AND COMPUTER CODES USED

Section	Faults	Computer Codes Utilized	Reactivity Coefficients Assumed: Moderator Density ($\Delta k/gm/cm^3$)	Reactivity Coefficients Assumed: Moderator Temperature (pcm/°F)[g]	Reactivity Coefficients Assumed: Doppler	Initial NSSS Thermal Power Output Assumed (MWt)
15.1	Increase in heat removal by the secondary system					
	Feedwater system malfunctions that result in an increase in feedwater flow	LOFTRAN, VIPRE (Also refer to subsection 15.4.1)	0.50	-	Lower (see figure 15.0.4-1)	0 and 3643[a,f]
	Excessive increase in secondary steam flow	LOFTRAN	0.0 and 0.50	-	Lower and upper (See figure 15.0.4-1)	3643[a,f]
	Inadvertent opening of a steam generator relief or safety valve	LOFTRAN	Function of moderator density (see figure15.1.4-1)	-	-2.9 pcm/°F	0 (subcritical)
	Steam system piping failure	LOFTRAN, VIPRE	Function of moderator density (see figure 15.1.4-1) and 0.50	-	See subsection 15.1.5 and lower (See figure 15.0.4-1)	0 (subcritical) and 3651[a,d,f]
15.2	Decrease in heat removal by the secondary system					
	Loss of external electrical load and/or turbine trip	LOFTRAN	Lower figure 15.0.4-2 and 0.50	+7	Lower and upper (see figure 15.0.4-1)	3651[a,d,f]
	Loss of nonemergency ac power to the station auxiliaries	LOFTRAN	Lower figure 15.0.4-2	+7	Upper (see figure 15.0.4-1)	3653[a,i]
	Loss of normal feedwater flow	LOFTRAN	Lower figure 15.0.4-2	+7	Upper (see figure 15.0.4-1)	3653[a,i]
	Feedwater system pipe break	LOFTRAN	Lower figure 15.0.4-2	+7	Lower (see figure 15.0.4-1)	3657[c,d,e]

TABLE 15.0.3-2 (SHEET 2 OF 3)

Section	Faults	Computer Codes Utilized	Reactivity Coefficients Assumed: Moderator Density ($\Delta k/gm/cm^3$)	Reactivity Coefficients Assumed: Moderator Temperature (pcm/°F)[g]	Reactivity Coefficients Assumed: Doppler	Initial NSSS Thermal Power Output Assumed (MWt)
15.3	Decrease in reactor coolant system flowrate					
	Partial and complete loss of forced reactor coolant flow	LOFTRAN, FACTRAN, VIPRE	Lower figure 15.0.4-2	+7	Upper (see figure 15.0.4-1)	3646[c,f]
	Reactor coolant pump shaft seizure (locked rotor)	LOFTRAN, FACTRAN	Lower figure 15.0.4-2	+7	Upper (see figure 15.0.4-1)	3657[c,d,e]
	Locked rotor rods in DNB	LOFTRAN, FACTRAN, VIPRE	0	0	Upper (see figure 15.0.4-1)	3646[c,f]
15.4	Reactivity and power distribution anomalies					
	Uncontrolled RCCA bank withdrawal from a subcritical or low power startup condition	TWINKLE, FACTRAN, VIPRE	Refer to subsection15.4.1.2	+7	Coefficient is consistent with a Doppler Defect of -0.998%Δk	0
	Uncontrolled RCCA bank withdrawal at power	LOFTRAN, VIPRE	Lower and upper figure 15.0.4-2	+7	Lower and upper (see figure 15.0.4-1)	358.5, 2151, 3646[c,f]
	RCCA misalignment	VIPRE, ANC	NA	-	NA	3626[b]
	Startup of an inactive reactor coolant pump at an incorrect temperature	LOFTRAN, FACTRAN, THINC	Upper figure 15.0.4-2	-	Lower (see figure 15.0.4-1)	2611[a,e]
	Chemical and volume control system malfunction that results in a decrease in the boron concentration in the reactor coolant	NA	NA	-	NA	0 and 3626[b]
	Inadvertent loading and operation of a fuel assembly in an improper position	LEOPARD, TURTLE	NA	-	NA	3579

TABLE 15.0.3-2 (SHEET 3 OF 3)

Section	Faults	Computer Codes Utilized	Reactivity Coefficients Assumed: Moderator Density (Δk/gm/cm^3)	Moderator Temperature (pcm/°F)[g]	Doppler	Initial NSSS Thermal Power Output Assumed (MWt)
	Spectrum of RCCA ejection accidents	TWINKLE, FACTRAN THINC	Refer to subsection 15.4.8.	Refer to subsection 15.4.8.	Coefficient consistent with Doppler Defect of -1.00%Δk	0 and 3636[d]
	Steamline break with coincidental rod withdrawal at power[h]	LOFTRAN	0.50	-	Lower (see figure 15.0.4-1).	3646[c,f]
15.5	Increase in reactor coolant Inventory					
	Inadvertent operation of the emergency core cooling system during power operation	LOFTRAN	Figure 15.0.4-2	+7	Lower (see figure 15.0.4-1).	3657[b,c,f]
15.6	Decrease in reactor coolant inventory					
	Inadvertent opening of a pressurizer safety or relief valve	LOFTRAN	Figure 15.0.4-2	+7	Upper (see figure 15.0.4-1).	3643[a,f]
	Steam generator tube failure	LOFTTR2	Figure 15.0.4-2	-	Upper (see figure 15.0.4-1).	3653 [a,i]
	LOCAs resulting from the spectrum of postulated piping breaks within the reactor coolant pressure boundary	SATAN-VI, NOTRUMP WREFLOOD, COCO, LOCTA-IV BART,BASH, LOCBART	See subsection15.6.5 references.[b]	-	See subsection 15.6.5 references.	3636[b,d]

a. Minimum pump heat of 17 MW_t assumed.

b. No pump heat assumed.

c. Maximum pump heat of 20 MW_t assumed.

d. 2% of margin applied.

e. Standard Thermal Design Procedure (STDP) with a Thermal Design Flow (TDF) of 93,600 gpm/loop assumed.

f. Revised Thermal Design Procedure (RTDP) with a Minimum Measured Flow (MMF) of 95,600 gpm/loop assumed.

g. The moderator density coefficients for these accidents were translated into moderator temperature coefficients.

h. The automatic rod withdrawal capability of the rod control system has been disabled. This eliminates the possibility that a steamline break event will result in a consequential and coincidental rod withdrawal.

i. 1.7% measurement uncertainty recapture power uprate and +0.3% power uncertainty applied.

TABLE 15.0.3-3

NOMINAL VALUES OF PERTINENT PLANT PARAMETERS
UTILIZED IN THE ACCIDENT ANALYSES

Parameter	STDP Values	RTDP Values
Thermal output of NSSS (including minimum thermal power generated by the RCPs, MWt)	3643	3643
Core inlet temperature (°F)	556.2	556.9
Vessel average temperature (°F)	588.4	588.4
RCS pressure (psia)	2250	2250
Reactor coolant flow per loop (gal/min)	93,600[a]	95,600[b]
Steam flow from NSSS (lb/h)	16,310,000	16,310,000
Steam pressure at steam generator outlet (psia)	941	941
Maximum steam moisture content (%)	0.25	0.25
Assumed feedwater temperature at steam generator inlet (°F)	448.7	448.7
Average core heat flux (Btu/h-ft^2)	209,612	209,612

a. Thermal design flow (TDF)
b. Minimum measured flow (MMF)

TABLE 15.0.6-1 (SHEET 1 OF 2)

TRIP POINTS AND TIME DELAYS TO TRIP ASSUMED
IN ACCIDENT ANALYSES

Trip Function	Limiting Trip Point Assumed In Analyses	Time Delays (s)
Power range high neutron flux, high setting	118%	0.5
Power range high neutron flux, low setting	35%	0.5
Power range high positive neutron flux rate	9% of RTP 2 seconds lag time constant	0.65
High neutron flux, P-8	84%	0.5
Source range neutron flux	NA	0.5
Overtemperature ΔT	Variable (See figure 15.0.6-1.)	(a)
Overpower ΔT	Variable (See figure 15.0.6-1.)	(a)
High pressurizer pressure	2425 psig	2.0
Low pressurizer pressure	1920 psig	2.0
Low reactor coolant flow (from loop flow detectors)	87% loop flow	1.0
Reactor coolant pump undervoltage trip	68% nominal	1.5
Turbine trip	NA	2.0
Low-low steam generator level	27% of narrow range level span[e]	2.0
	0% of narrow range level span[d]	

TABLE 15.0.6-1 (SHEET 2 OF 2)

Trip Function	Limiting Trip Point Assumed In Analyses	Time Delays (s)
High steam generator level trip of the feedwater pumps and closure of feedwater system valves and turbine trip	100.0% of narrow range level span	N/A[b] 7.0[c]

a. The response time test criteria provided in chapter 16 are based on evaluations of the FSAR chapter 15 analyses which model the RTD time constant as a 4-s first-order lag and include a 2-s filter on temperature measurement with the 2-s electronics delay (4-s first-order lag and a 4-s pure delay). The electronics delay is the trip circuit channel electronics delay plus the time for the reactor breakers to open and the time for the CRDM stationary grippers to disengage (gripper release time).

The model previously used in the FSAR analyses combined the 4-s RTD response time and 2-s filter on measured ΔT as a single first-order lag. The 2-s electronics delay was modeled as a pure delay.

b. Turbine trip is not modeled as a primary actuation signal in the safety analysis.

c. From time setpoint is reached to feedwater isolation.

d. Low-low level setpoint assumed for feedwater malfunction (subsection 15.1.2) and feedwater system pipe break (subsection 15.2.8); environmental errors are included.

e. Low-low level setpoint assumed for loss of nonemergency ac power to the station auxiliaries (subsection 15.2.6) and loss of normal feedwater (subsection 15.2.7); no environmental errors are included.

TABLE 15.0.7-1

DETERMINATION OF MAXIMUM OVERPOWER TRIP POINT, POWER RANGE NEUTRON FLUX CHANNEL, BASED ON NOMINAL SETPOINT CONSIDERING INHERENT INSTRUMENTATION ERRORS

Nominal setpoint (% of rated power)	109	
Calorimetric errors in the measurement of secondary system thermal power:		

Variable	Accuracy of Measurement of Variable	Estimated Effect on Thermal Power Determination (% error)
Feedwater temperature	±10°F	
Feedwater pressure (Small correction on enthalpy)	±60 psi	0.3
Steam pressure (Small correction on enthalpy)	±34.3 psi	
Feedwater flow	+3.3, -2.8% of full power flow	1.25
Assumed calorimetric error		2 (a)*
Axial power distribution effects on total ion chamber current		
Estimated error (% of rated power)	3	
Assumed error (% of rated power)		5 (b)*
Instrumentation channel drift and setpoint reproducibility		
Estimated error (% of rated power)	1	
Assumed error (% of rated power)		2 (c)*
*Total assumed error in setpoint (% of rated power): (a) + (b) + (c)		+9
Maximum overpower trip point assuming all individual errors are simultaneously in the most adverse direction (% of rated power)		118

TABLE 15.0.8-1 (SHEET 1 OF 3)

PLANT SYSTEMS AND EQUIPMENT AVAILABLE FOR TRANSIENT AND ACCIDENT CONDITIONS

FSAR Section	Incident	Reactor Trip Functions	ESF Actuation Functions	ESF Equipment	Other Equipment
15.1	Increase in heat removed by the secondary system				
	Feedwater system malfunctions that result in an increase in feedwater flow	Power range high flux, turbine trip, manual	High-high steam generator level produced feedwater isolation and turbine trip	-	Feedwater isolation valves
	Excessive increase in secondary steam flow	Power range high flux, overtemperature ΔT, overpower ΔT, manual	-	-	Pressurizer self-actuated safety valves, steam generator safety valves
	Inadvertent opening of a steam generator relief or safety valve	Low pressurizer pressure, manual, SIS	Low pressurizer pressure, low compensated steam line pressure	Auxiliary feedwater system, safety injection system (SIS)	Feedwater isolation valves, steamline stop valves
	Steam system piping failure	SIS, low pressurizer pressure, manual	Low pressurizer pressure, low compensated steamline pressure, high-1 containment pressure, manual	Auxiliary feedwater system, SIS	Feedwater isolation valves, main steamline isolation valves (MSIVs), residual heat removal, containment spray
15.2	Decrease in heat removal by the secondary system				
	Loss of external load /turbine trip	High pressurizer pressure, overtemperature ΔT, high pressurizer water level, low-low steam generator water level, manual	-	-	Pressurizer relief and safety valves, steam generator safety valves
	Loss of nonemergency ac power to the station auxiliaries	Steam generator low-low level, manual	Steam generator low-low level	Auxiliary feedwater system	Steam generator safety valves, pressurizer relief and safety valves
	Loss of normal feedwater flow	Steam generator low-low level, manual	Steam generator low-low level	Auxiliary feedwater system	Steam generator safety valves, pressurizer relief and safety valves
	Feedwater system pipe break	Steam generator low-low level, high pressure, SIS, manual	High-1 containment pressure, high-2 containment pressure steam generator low-low water level, low compensated steamline pressure	Auxiliary feedwater system, SIS	MSIVs, feedline isolation, pressurizer self-actuated safety valves, steam generator safety valves

TABLE 15.0.8-1 (SHEET 2 OF 3)

FSAR Section	Incident	Reactor Trip Functions	ESF Actuation Functions	ESF Equipment	Other Equipment
15.3	Decrease in reactor coolant system flowrate				
	Partial and complete loss of forced reactor coolant flow	Low flow, undervoltage, underfrequency, manual	-	-	Steam generator safety valves
	Reactor coolant pump (RCP) shaft seizure (locked rotor)	Low flow, manual high pressurizer pressure	-	-	Pressurizer safety valves, steam generator safety valves
15.4	Reactivity and power distribution anomalies				
	Uncontrolled RCCA bank withdrawal from a subcritical or low power startup condition	Power range high flux (low setpoint), source range high flux, intermediate range high flux, manual	-	-	-
	Uncontrolled RCCA bank withdrawal at power	Power range high flux overtemperature ΔT, high pressurizer pressure, manual power range high positive neutron flux rate	-	-	Pressurizer safety valves, steam generator safety valves
	RCCA misalignment	Overtemperature ΔT, manual	-	-	-
	Startup of an inactive reactor coolant loop at an incorrect temperature	Power range high flux, low flow (P-8 interlock), manual	-	-	Low insertion limit annunciators for boration
	CVCS malfunction that results in a decrease in boron concentration in the reactor coolant	Source range high flux, power range high flux, overtemperature ΔT, manual	-	-	Low insertion limit annunciators for boration
	Spectrum of RCCA ejection accidents	Power range high flux, high positive flux rate, manual	-	-	-
	Steamline break with coincidental rod withdrawal at power	Overpower and overtemperature ΔT, manual	Low pressurizer pressure, low compensated steamline pressure, high-1 containment pressure, manual	Auxiliary feedwater system, SIS	Feedwater Isolation valves, MSIVs, RHR, containment spray, pressurizer safety valves
15.5	Increase in reactor coolant inventory				
	Inadvertent operation of the ECCS during power operation	Low pressurizer pressure, manual, safety injection trip	-	SIS	-

TABLE 15.0.8-1 (SHEET 3 OF 3)

FSAR Section	Incident	Reactor Trip Functions	ESF Actuation Functions	ESF Equipment	Other Equipment
15.6	Decrease in reactor coolant inventory				
	Inadvertent opening of a pressurizer, safety, or relief valve	Low pressurizer pressure, overtemperature ΔT, manual	-	-	-
	Steam generator tube failure	Low pressurizer pressure, overtemperature ΔT, SIS manual	Low pressurizer pressure	ECCS, auxiliary feedwater system, emergency power system	Nuclear service cooling water (NSCW) system, component cooling water system, steam generator shell-side fluid operating system, steam generator safety and/ or relief valves, MSIVs pressurizer PORVs, radiation monitors (air ejector, streamline and SG blowdown)
	LOCAs resulting from the spectrum of postulated piping breaks within the reactor coolant pressure boundary	Low pressurizer pressure, SIS, manual	High-1 containment pressure, low pressurizer pressure	ECCS, auxiliary feedwater system, containment heat removal system, emergency power system	NSCW system, component cooling water system, steam generator safety and/or relief valves

TABLE 15.0.8-2 (SHEET 1 OF 6)

FISSION PRODUCT BARRIERS FOR TRANSIENT AND ACCIDENT ANALYSES

FSAR Section	Event Description	Fission Product Barrier	Design Basis Limit	References [1]
2.4.13	Accidental Releases of Liquid Effluents in Ground and Surface Water	Fuel Cladding, RCS Pressure Boundary, Containment	None credited	N/A
5.2.2.10	RCS Pressure Control During Low-Temperature Operation	Fuel Cladding	None credited	N/A
		RCS Pressure Boundary	RCS within ASME stress limits (Appendix G, heatup/cooldown)	FSAR 5.2.2.10, 1.9.65.2, 1.9.99.2, PTLR
		Containment	None credited	N/A
15.1.1	Decreased Feedwater Temperature	Fuel Cladding, RCS Pressure Boundary, Containment	Bounded by 15.1.3	FSAR 15.1.1.2.2
15.1.2	Increased Feedwater Flow	Fuel Cladding	DNBR bounded by 15.4.1	FSAR 15.1.2.3
			DNBR limit ≥ 1.22-1.24	FSAR 4.4.1.1.2
			Fuel centerline < 4700 °F	FSAR 15.1.2.2.2, Table 4.4-1.
		RCS Pressure Boundary	RCS Pressure < 110% of design = 2750 psia	Implied in FSAR 15.0.1.2, Table 5.3.3-1
		Containment	None credited	N/A
15.1.3	Increased Steam Flow	Fuel Cladding	DNBR limit ≥ 1.22-1.24	FSAR 15.1.3.3, 4.4.1.1.2
			Fuel centerline < 4700 °F	FSAR 15.1.3.2.2, 4.2.1.2 Table 4.4-1
			PCT ≤ 2200 ºF	FSAR 15.1.3.2.2, 15.6.5.3.3.1
		RCS Pressure Boundary	RCS pressure < 110% of design = 2750 psia	Implied in FSAR 15.0.1.2, Table 5.3.3-1
		Containment	None credited	N/A
15.1.4	SG SRV Opening	Fuel Cladding	DNBR limit ≥ 1.22-1.24	FSAR 15.1.4.3, 4.4.1.1.2, 15.1.4.1, Table 4.4-1
			Fuel centerline < 4700 °F	FSAR 15.1.4.1, 4.2.1.2, 15.6.5.3.3.1, Table 4.4-1
			PCT ≤ 2200 °F	FSAR 6.3.1
		RCS Pressure Boundary	RCS pressure < 110% of design = 2750 psia	FSAR 15.1.4.1, Table 5.3.3-1
		Containment	None credited	N/A

TABLE 15.0.8-2 (SHEET 2 OF 6)

FSAR Section	Event Description	Fission Product Barrier	Design Basis Limit	References
15.1.5	Steam Line Break	Fuel Cladding	DNBR limit ≥ 1.22-1.24	FSAR 15.1.5.2.4, 4.4.1.1.2
			PCT ≤ 2200 °F	FSAR 6.3.1, 15.1.5.1, 15.6.5.3.3.1
			I_2 spike ≤ 60 μCi/gm and 500 x normal appearance rate	FSAR 15.1.5.3.1.1A, Table 15.1.5-2
		RCS Pressure Boundary	RCS pressure < 110% of design = 2750 psia	FSAR 15.1.5.1, Table 5.3.3-1
			RCS-to-SG leak ≤1gpm, and < 500 gal/day to any one SG	FSAR 15.1.5.3.2C
		Containment	None credited	N/A
15.2.2	Loss of External Electrical Load	Fuel Cladding	Bounded by 15.2.3	FSAR 15.2.2.3
		RCS Pressure Boundary	RCS pressure ≤ 2750 psia	FSAR 15.2.2.1
		Containment	None credited	N/A
15.2.3	Turbine Trip	Fuel Cladding	DNBR limit ≥ 1.22-1.24	FSAR 15.2.3.3, 4.4.1.1.2
		RCS Pressure Boundary	RCS pressure ≤ 2750 psia	FSAR 15.2.3.2.2, Table 5.3.3-1
		Containment	None credited	N/A
15.2.4	MSIV Closure	Fuel Cladding, RCS Pressure Boundary, Containment	Bounded by 15.2.3	FSAR 15.2.4
15.2.5	Loss of Condenser Vacuum	Fuel Cladding, RCS Pressure Boundary, Containment	Bounded by 15.2.3	FSAR 15.2.5
15.2.6	Loss of Offsite Power	Fuel Cladding	DNBR ≥ 1.22-1.24	Implied in FSAR 15.0.1.2, 4.4.1.1.2
			I_2 spike ≤ 60 μCi/gm and 500 x normal appearance rate	FSAR 15.2.6.3.2B, Table 15.2.6-1
		RCS Pressure Boundary	RCS pressure ≤ 2750 psia	FSAR 15.2.6.2.1D, Table 5.3.3-1
			Pressurizer H_2O vol. < 1812.77 ft^3	FSAR 15.2.6.2.2, Figure 15.2.6-1
			RCS-to-SG leak ≤1gpm	FSAR 15.2.6.3.2D, Table 15.2.6-1
		Containment	None credited	N/A
15.2.7	Loss of Feedwater	Fuel Cladding	DNBR ≥ 1.22-1.24	Implied in FSAR 15.0.1.2, 4.4.1.1.2, SER 15.2.7
		RCS Pressure Boundary	RCS pressure ≤ 2750 psia	FSAR 15.2.7.2.1, Table 5.3.3-1
			Pressurizer H_2O vol. < 1812.77 ft^3	FSAR 15.2.7.2.2, Figure 15.2.7-1
		Containment	None credited	N/A
15.2.8	Feedwater Line Break	Fuel Cladding	DNBR ≥ 1.22-1.24	Implied in FSAR 15.0.1.2, 4.4.1.1.2, SER 15.2.8
			PCT ≤ 2200 °F	FSAR 6.3.1

FSAR Section	Event Description	Fission Product Barrier	Design Basis Limit	References
		RCS Pressure Boundary	RCS pressure ≤ 2750 psia	FSAR 15.2.8.2.2, Table 5.3.3-1
		Containment	None credited	N/A
15.3.1	Partial Loss of Forced Reactor Coolant Flow	Fuel Cladding	DNBR limit ≥ 1.22-1.24	FSAR 15.3.1.3, 4.4.1.1.2
		RCS Pressure Boundary	RCS pressure ≤ 2750 psia	Implied in FSAR 15.0.1.2, Table 5.3.3-1, SER 15.3.1
		Containment	None credited	N/A
15.3.2	Complete Loss of RCS Flow	Fuel Cladding	DNBR limit ≥ 1.22-1.24	FSAR 15.3.3, 4.4.1.1.2
		RCS Pressure Boundary	RCS pressure ≤ 2750 psia	Implied in FSAR 15.0.1.3, Table 5.3.3-1, SER 15.3.1
		Containment	None credited	N/A
15.3.3	Locked Rotor	Fuel Cladding	≤ 5% fuel rods exceed DNBR	FSAR 15.3.3.3.1.1.C
			I_2 spike ≤ 60 μCi/gm	FSAR 15.3.3.3.2, Table 15.3.3-2
			PCT ≤ 2700 °F (ZIRLO) PCT ≤ 2375 °F (Optimized ZIRLO)	FSAR 15.3.3.2.7, 15.4.8 (Ref. 9)
		RCS Pressure Boundary	RCS stress ≤ ASME limits in Tables 3.9.B.3-2 through 6	Implied in FSAR 15.0.1.4, SER 15.3.3
			RCS-to-SG leak ≤1gpm	FSAR 15.3.3.3.2D, Table 15.3.3-2
		Containment	None credited	N/A
15.3.4	RCP Shaft Break	Fuel Cladding	Bounded by FSAR 15.3.3.2.7	FSAR 15.3.4.2
		RCS Pressure Boundary	Bounded by FSAR 15.3.3.2.7	FSAR 15.3.4.2
		Containment	None credited	N/A
15.4.1	Low Power RCCA Withdrawal	Fuel Cladding	DNBR limit ≥ 1.22-1.24	FSAR 15.4.1.3, 4.4.1.1.2
			Fuel centerline ≤ 4700 °F	FSAR 4.2.1.2, Table 4.4-1, SER 15.4.1
		RCS Pressure Boundary	RCS stress ≤ ASME limits in Tables 3.9.B.3-2 through 6	FSAR 15.4.1.3
		Containment	None credited	N/A
15.4.2	At Power RCCA Withdrawal	Fuel Cladding	DNBR limit ≥ 1.22-1.24	FSAR 15.4.2.2.2, 4.4.1.1.2
			Linear heat rate ≤ 22.4 kw/ft	FSAR 15.4.2.1, 4.4.2.11.6
			Fuel centerline ≤ 4700 °F	FSAR 15.4.2.2.2, 4.2.1.2, Table 4.4-1
			PCT < 2200 °F	FSAR 15.4.2.2.2,
			Cladding strain < 1%	FSAR 4.2.1.1B.2, SER 15.4.2

TABLE 15.0.8-2 (SHEET 4 OF 6)

FSAR Section	Event Description	Fission Product Barrier	Design Basis Limit	References
		RCS Pressure Boundary	RCS pressure ≤ 2750 psia	FSAR 15.4.2
		Containment	None credited	N/A
15.4.3	RCCA Misalignment	Fuel Cladding	DNBR limit ≥ 1.22-1.24, or ≤ 5% fuel rods exceed DNBR Linear heat rate ≤ 22.4 kw/ft Cladding strain < 1% Fuel centerline ≤ 4700 °F	FSAR 15.4.3.3, 4.4.1.1.2 FSAR 15.4.3.2.1.2D.1 FSAR 15.4.3.2.1.2C, 4.4.2.11.6 FSAR 15.4.3.1, 4.2.1.1B.2 FSAR 15.4.3.1, 4.2.1.2, Table 4.4-1
		RCS Pressure Boundary	RCS stress within ASME limits (Appendix G, heatup/cooldown)	FSAR 15.4.3.2.1.2D, 5.3.2.1, PTLR
		Containment	None credited	N/A
15.4.4	Startup of Inactive RCS Loop	Fuel Cladding	DNBR limit ≥ 1.22-1.24	FSAR 15.4.4.3, 4.4.1.1.2
		RCS Pressure Boundary	RCS pressure ≤ 2750 psia	Implied in FSAR 15.0.1.2, Table 5.3.3-1, SER 15.4.4
		Containment	None credited	N/A
15.4.6	Boron Dilution	Fuel Cladding	DNBR limit ≥ 1.22-1.24	Implied in FSAR 15.0.1.2, 4.4.1.1.2
		RCS Pressure Boundary	RCS pressure ≤ 2750 psia	Implied in FSAR 15.0.1.2, Table 5.3.3-1
		Containment	None credited	N/A
15.4.7	Misloaded Fuel Assembly	Fuel Cladding, RCS Pressure Boundary, Containment	None credited	N/A
15.4.8	RCCA Ejection	Fuel Cladding	Avg. fuel enthalpy < 200 cal/g ≤ 10% fuel rods exceed DNBR ≤10% fuel centerline ≥ 4700 °F I_2 spike ≤ 60 µCi/gm	FSAR 15.4.8.1.2A, 1.9.77 FSAR 15.4.8.3.1.1C FSAR 15.4.8.1.2C FSAR 15.4.8.3.1.1A
		RCS Pressure Boundary	RCS stress ≤ ASME limits in Tables 3.9.B.3-2 through 6 RCS-to-SG leakage ≤ 1 gpm	FSAR 15.4.8.1.2B, 1.9.77 FSAR 15.4.8.3.2D, Table 15.4.8-2
		Containment	Containment leakage ≤ 0.2%/day	FSAR Table 15.4.8-2
15.4.9	MSLB w/ RCCA Withdrawl	Fuel Cladding, RCS Pressure Boundary, Containment	Bounded by 15.1.5	FSAR 15.4.9.3
15.5.1	Inadvertent ECCS Operation	Fuel Cladding	DNBR limit ≥ 1.22-1.24	FSAR 15.5.1.2.1B, 4.4.1.1.2
		RCS Pressure Boundary	RCS pressure ≤ 2750 psia PSV water relief ≤ 3 cycles	FSAR 15.5.1.2.1A, Table 5.3.3-1 FSAR 15.5.1.3, Table 15.5.1-1
		Containment	None credited	N/A

TABLE 15.0.8-2 (SHEET 5 OF 6)

FSAR Section	Event Description	Fission Product Barrier	Design Basis Limit	References
15.5.2	Increased RCS Inventory	Fuel Cladding, RCS Pressure Boundary, Containment	Bounded by 15.4.6 and 15.5.1	FSAR 15.5.2
15.6.1	Inadvertent Opening of a PORV/SRV	Fuel Cladding	DNBR limit ≥ 1.22-1.24	FSAR 15.6.1.2.2, 4.4.1.1.2
		RCS Pressure Boundary	None credited	N/A
		Containment	None credited	N/A
15.6.2	Small RCS Line Break	Fuel Cladding	DNBR ≥ 1.22-1.24	Implied in FSAR 15.0.1.2, 4.4.1.1.2
			I_2 spike ≤ 60 µCi/gm	FSAR 15.6.2.1A
		RCS Pressure Boundary	None credited	N/A
		Containment	None credited	FSAR 15.6.2.1H, 15.6.2.2F
15.6.3	SGTR	Fuel Cladding	DNBR ≥ 1.22-1.24	Implied by SRP 15.6.3, and FSAR 15.6.3.4.2, 6.3.1, 4.4.1.1.2
			I_2 spike ≤ 60 µCi/gm and 500 x normal appearance rate	FSAR 15.6.3.4.2.a
			PCT ≤ 2200 °F	FSAR 6.3.1
		RCS Pressure Boundary	RCS stress ≤ ASME limits in Tables 3.9.B.3-2 through 6	FSAR Figure 15.6.3-2
			RCS-to-intact SG leak ≤ 0.7 gpm	FSAR 15.6.3.4.3.c, Table 15.6.3-4
		Containment	None credited	N/A
15.6.5	LOCA	Fuel Cladding	PCT ≤ 2200 °F	FSAR 15.6.5.1A
			Clad oxidation ≤ 17% locally	FSAR 15.6.5.1C
		RCS Pressure Boundary	ECCS leakage outside containment is ≤ 2 gpm	FSAR Table 15.6.5-4
			RCS meets ASME limits in Tables 3.9.B.3-2 through 6	FSAR 15.6.5.1E, 3.9.N.1.4.8
		Containment	Stress ≤ ASME CC-3400 limits (including pressure ≤ 52 psig and temperature ≤ 381 °F)	FSAR 3.8.1.5 FSAR Table 6.2.1-1 (including footnote "a")
			Containment leakage ≤ 0.2%/day	FSAR 15.6.5.4.5C, Table 15.6.5-4
			Containment $H_2 \leq 4\%$	FSAR 6.2.5.3.1.6
15.7.1	Gas Decay Tank Failure	Fuel Cladding, RCS Pressure Boundary, Containment	None credited	N/A

TABLE 15.0.8-2 (SHEET 6 OF 6)

FSAR Section	Event Description	Fission Product Barrier	Design Basis Limit	References
15.7.2	Radwaste Liquid Leak	Fuel Cladding, RCS Pressure Boundary, Containment	None credited	N/A
15.7.3	Radwaste Liquid Tank Failure	Fuel Cladding, RCS Pressure Boundary, Containment	None credited	N/A
15.7.4	FHA	Fuel Cladding	Max. drop height = 13.5 feet	Implied by assemblies affected value in FSAR Table 15.7.4-1
			Pool water depth = 35 feet	Encl. to LCV-0828A, FSAR 9.1.3.5
		RCS Pressure Boundary	None credited	N/A
		Containment	No credit taken; containment integrity (ability to close containment) is retained as "defense-in-depth."	FSAR 3.8, 15.7.4.5.1.2, Table 15.7.4-1, 15.7.4-2
15.7.5	Spent Fuel Cask Drop	Fuel Cladding	Single failure proof crane limits per NUREG-0554	FSAR 15.7.5
		RCS Pressure Boundary	None credited	N/A
		Containment	None credited	N/A

1. References to SER sections refer to NUREG-1137 and supplements 1-9. References to SRP sections refer to NUREG-0800.

TABLE 15.0.9-1
Deleted

TABLE 15.0.12-1 (SHEET 1 OF 2)

SINGLE FAILURES ASSUMED IN ACCIDENT ANALYSES

Event Description	Worst Failure Assumed
Feedwater temperature reduction[(a)]	-
Excessive feedwater flow	One protection train
Excessive steam flow[(a)]	-
Inadvertent secondary depressurization	One safety injection train
Steam system piping failure	One safety injection train
Steam pressure regulator malfunction[(b)]	-
Loss of external load	One protection train
Turbine trip	One protection train
Inadvertent closure of main steam isolation valve	One protection train
Loss of condenser vacuum	One protection train
Loss of ac power	One auxiliary feedwater pump
Loss of normal feedwater	One auxiliary feedwater pump
Feedwater system pipe break	One protection train
Partial loss of forced reactor coolant flow	One protection train
Complete loss of forced reactor coolant flow	One protection train
Reactor coolant pump locked rotor	One protection train
Reactor coolant pump shaft break	One protection train
Rod cluster control (RCCA) bank withdrawal from subcritical	One protection train
RCCA bank withdrawal at power	One protection train
Dropped RCCA, dropped RCCA bank	One nuclear instrumentation system channel
Statically misaligned RCCA[(c)]	-
Single RCCA withdrawal	One protection train
Inactive reactor coolant pump startup	One protection train
Flow controller malfunction[(b)]	-
Uncontrolled boron dilution	One protection train
Improper fuel loading[(c)]	-
RCCA ejection	One protection train

TABLE 15.0.12-1 (SHEET 2 OF 2)

Event Description	Worst Failure Assumed
Inadvertent emergency core cooling system operation at power	One protection train
Increase in reactor coolant system inventory	One protection train
Boiling water reactor transients[b]	-
Inadvertent reactor coolant system depressurization	One protection train
Failure of small lines carrying primary coolant outside containment[c]	-
Boiling water reactor piping failures[b]	-
Spectrum of loss-of-coolant accident small breaks	One safety injection train
Large breaks	One residual heat removal pump

a. No protection action required.

b. Not applicable to VEGP.

c. No transient analysis involved.

TABLE 15.0.13-1

OPERATOR ACTIONS[a] REQUIRED FOR SMALL AND LARGE LOCAs

Time	Operator Action
Reactor trip signal is actuated[b]	None
Safety injection signal is actuated[b]	None
Prior to generation of refueling water storage tank low-low-level signal	Reset safeguards actuation signal. Check sump water level indicator.
Generation of refueling water storage tank low-low level signal	Verify opening of the residual heat removal containment emergency sump valves. Perform the additional valve alignments required for switchover to recirculation. .[c]
Switchover to hot leg recirculation	Perform operations necessary to switch to simultaneous hot and cold leg recirculation
To final stabilized condition	Monitor system pressure and temperature. Control pressurizer water level with safety injection system (SIS)

a. Actions associated with primary system protection.

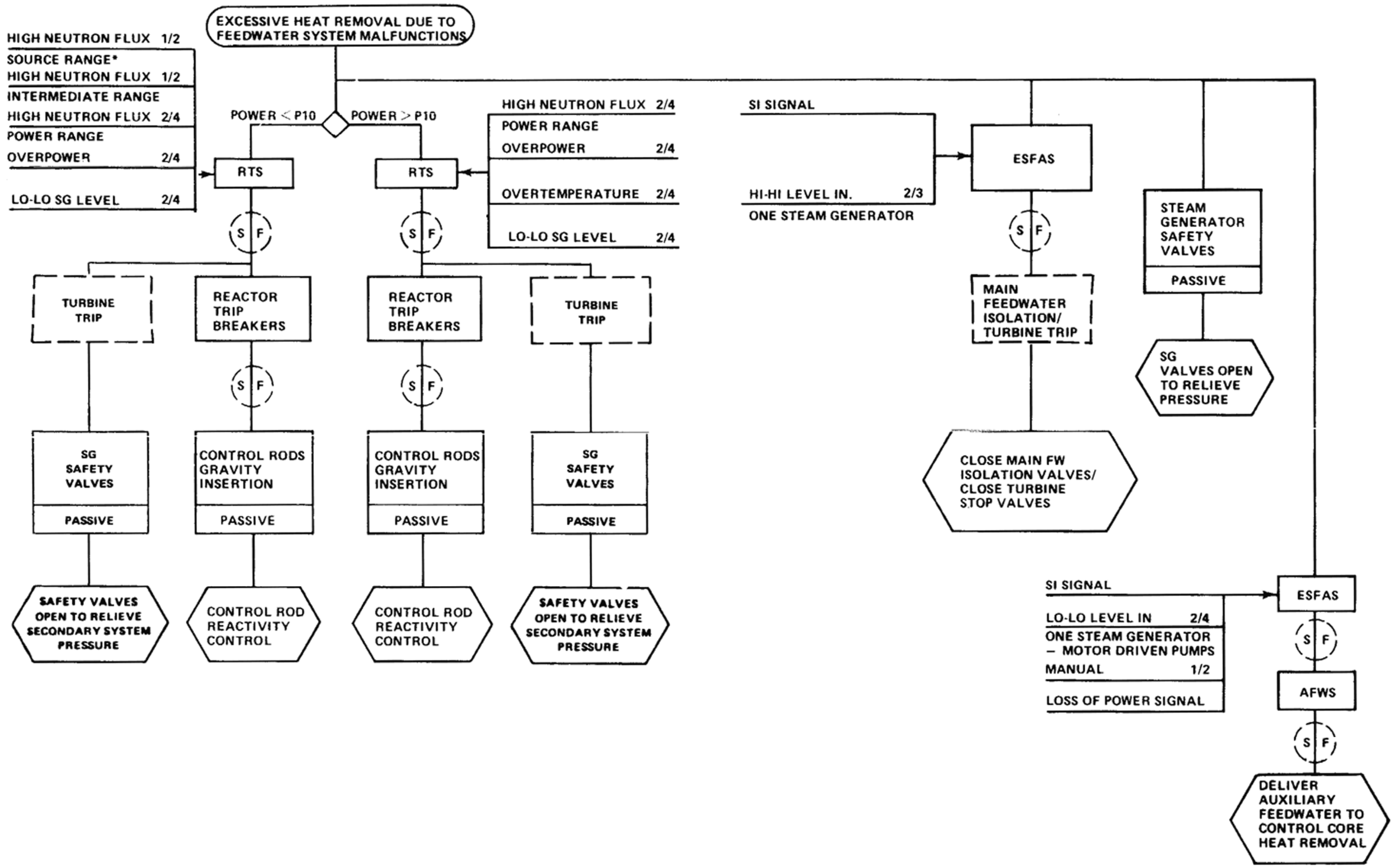
b. These times can be found in the sequence of events tables in subsection 15.6.5.

c. See paragraph 6.3.2.8 for the manual actions required for completion of switchover. Operation actions associated with containment protection are discussed in section 6.2.

TABLE 15.0.13-2

SHORT-TERM OPERATOR ACTIONS REQUIRED FOR STEAM GENERATOR TUBE RUPTURE

Time	System	Operator Action
Reactor trip signal actuated	Reactor trip system	None
Safety injection signal actuated	SIS	None
Post-safety injection signal generation to plant stabilization	SIS, auxiliary feedwater system, steam dump system	Perform operations necessary to isolate affected steam generator. Observe pressurizer water level controlling with SIS. Reduce system temperature and pressure within limits described by pressure-temperature limit curves. Proceed with normal plant cooldown, while monitoring RCS pressure, temperature, and boron concentration.



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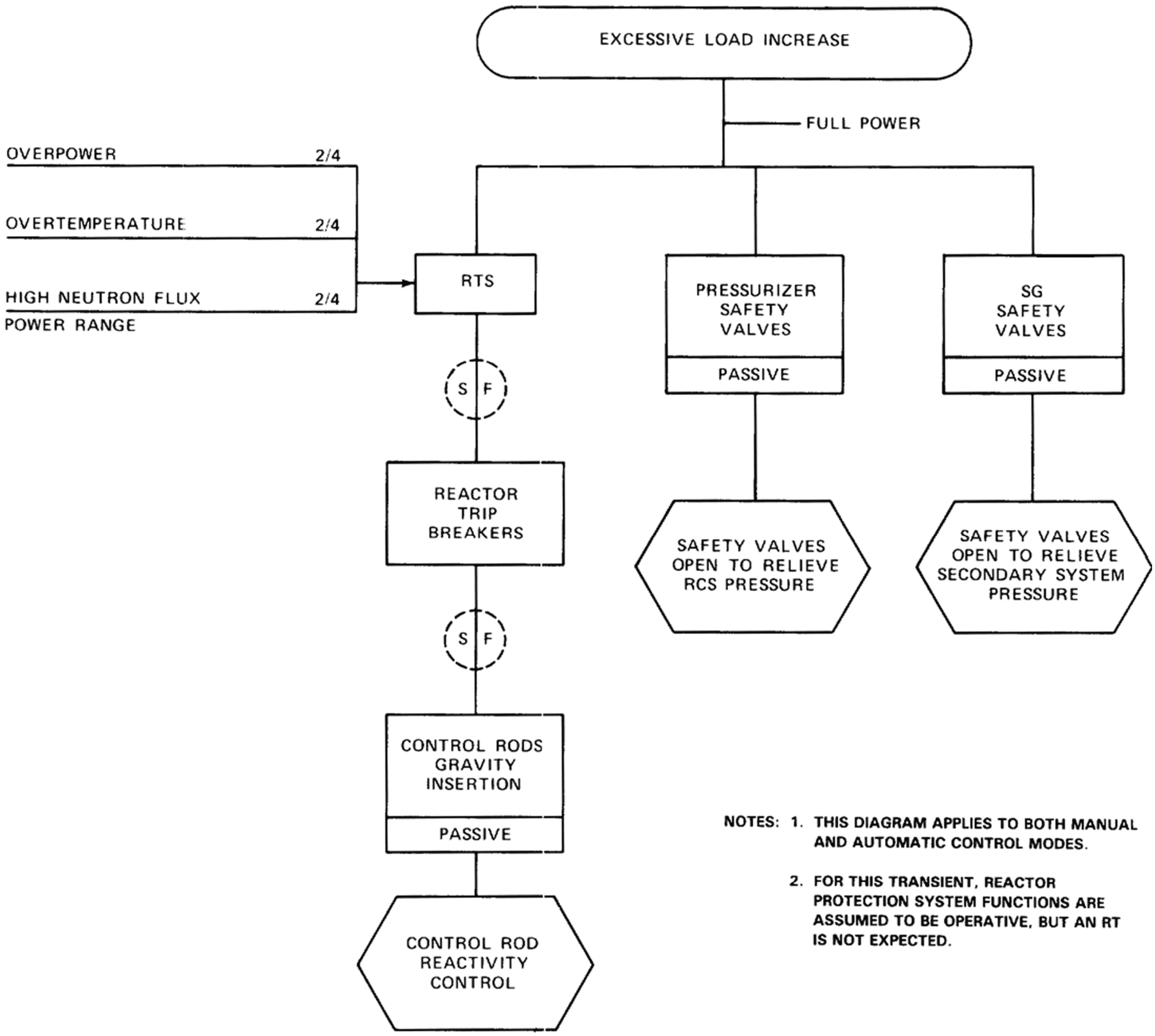
UNIT 1 AND UNIT 2

EXCESSIVE HEAT REMOVAL DUE TO FEEDWATER SYSTEMS MALFUNCTION

FIGURE 15.0.1–1



NOTES: 1. THIS DIAGRAM APPLIES TO BOTH MANUAL AND AUTOMATIC CONTROL MODES.

2. FOR THIS TRANSIENT, REACTOR PROTECTION SYSTEM FUNCTIONS ARE ASSUMED TO BE OPERATIVE, BUT AN RT IS NOT EXPECTED.

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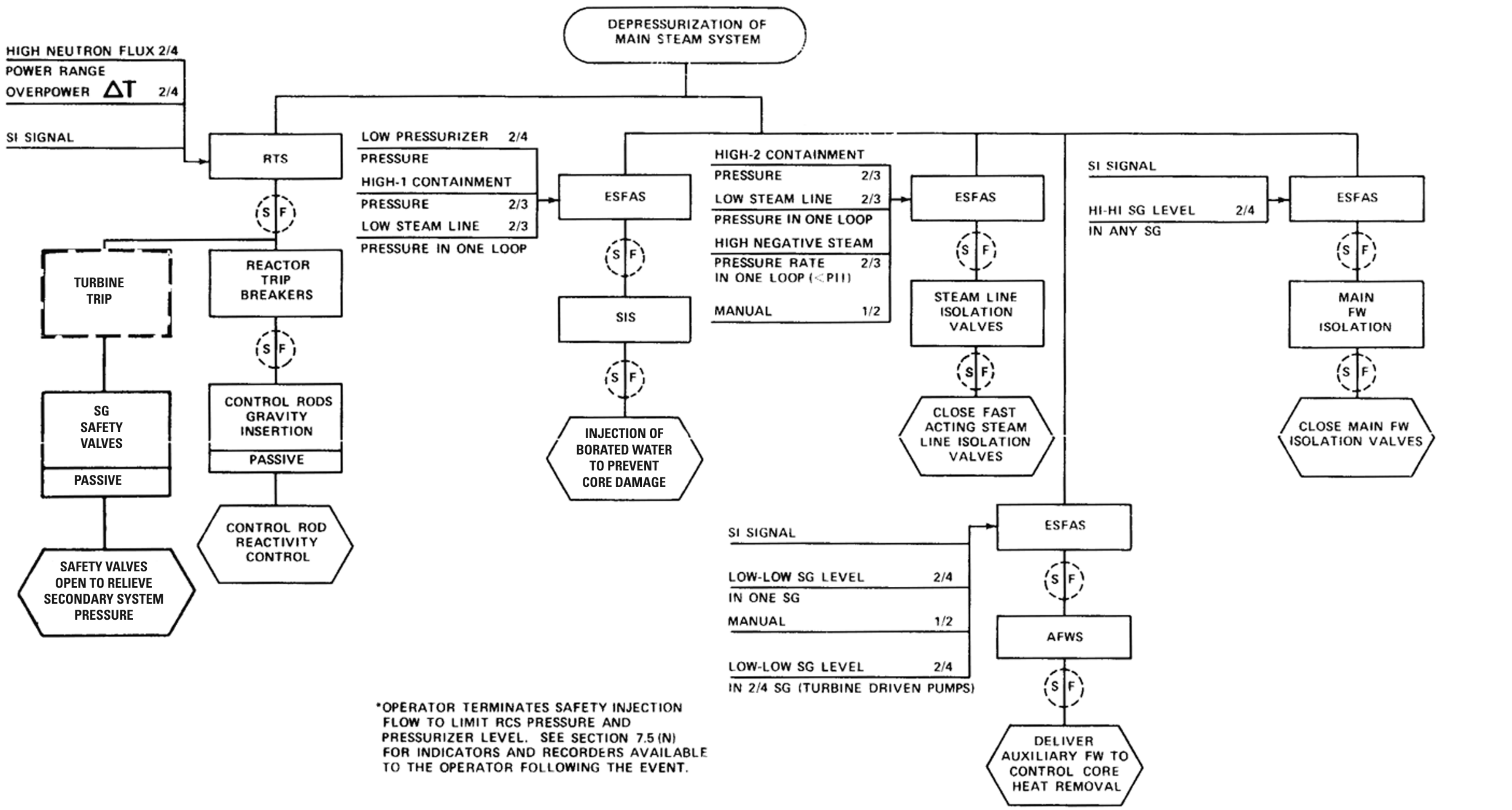
SOUTHERN COMPANY *Energy to Serve Your World®*	VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2	EXCESSIVE LOAD INCREASE FIGURE 15.0.1–2



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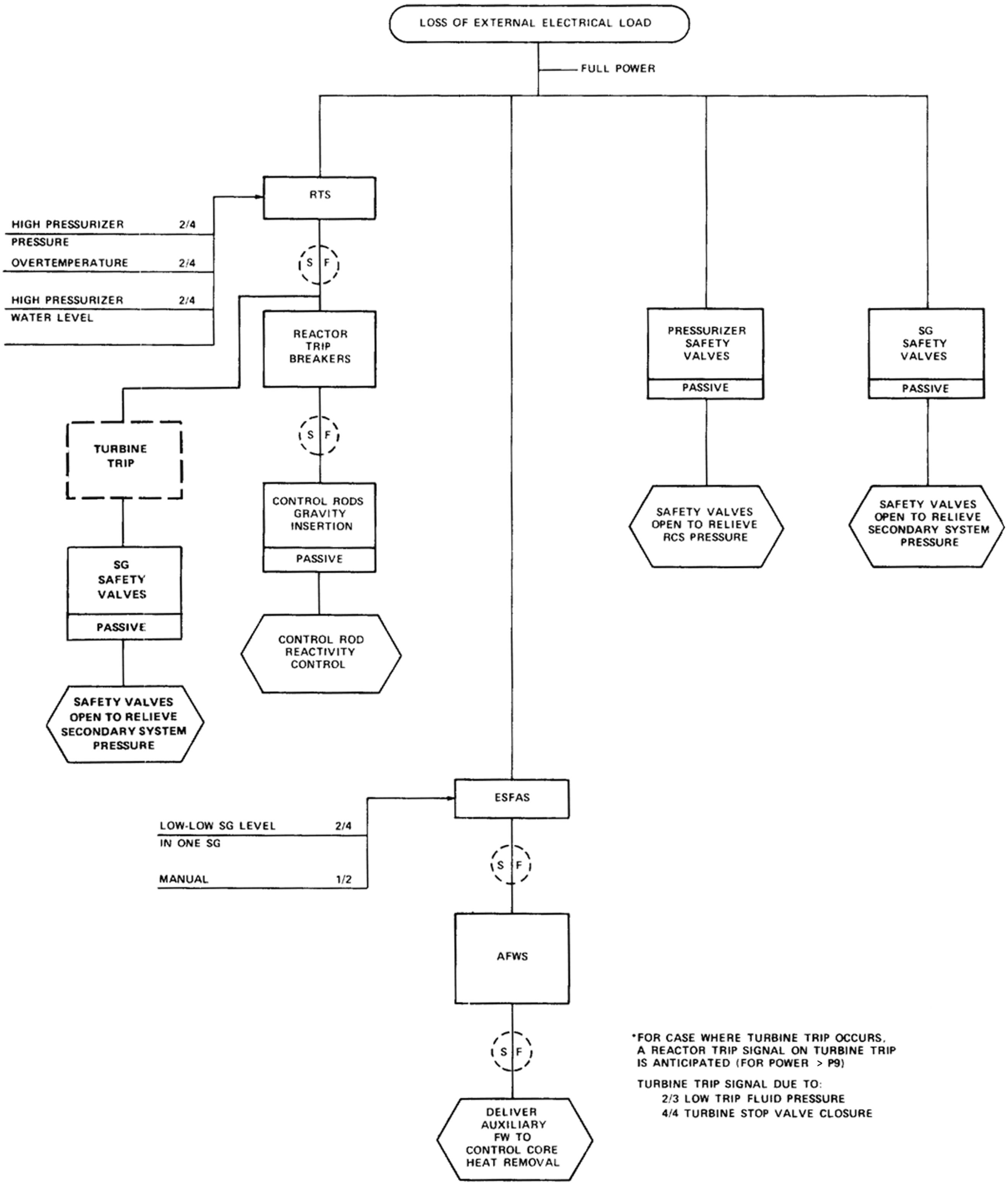
SOUTHERN COMPANY *Energy to Serve Your World®*	VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2	DEPRESSURIZATION OF MAIN STEAM SYSTEM
		FIGURE 15.0.1–3



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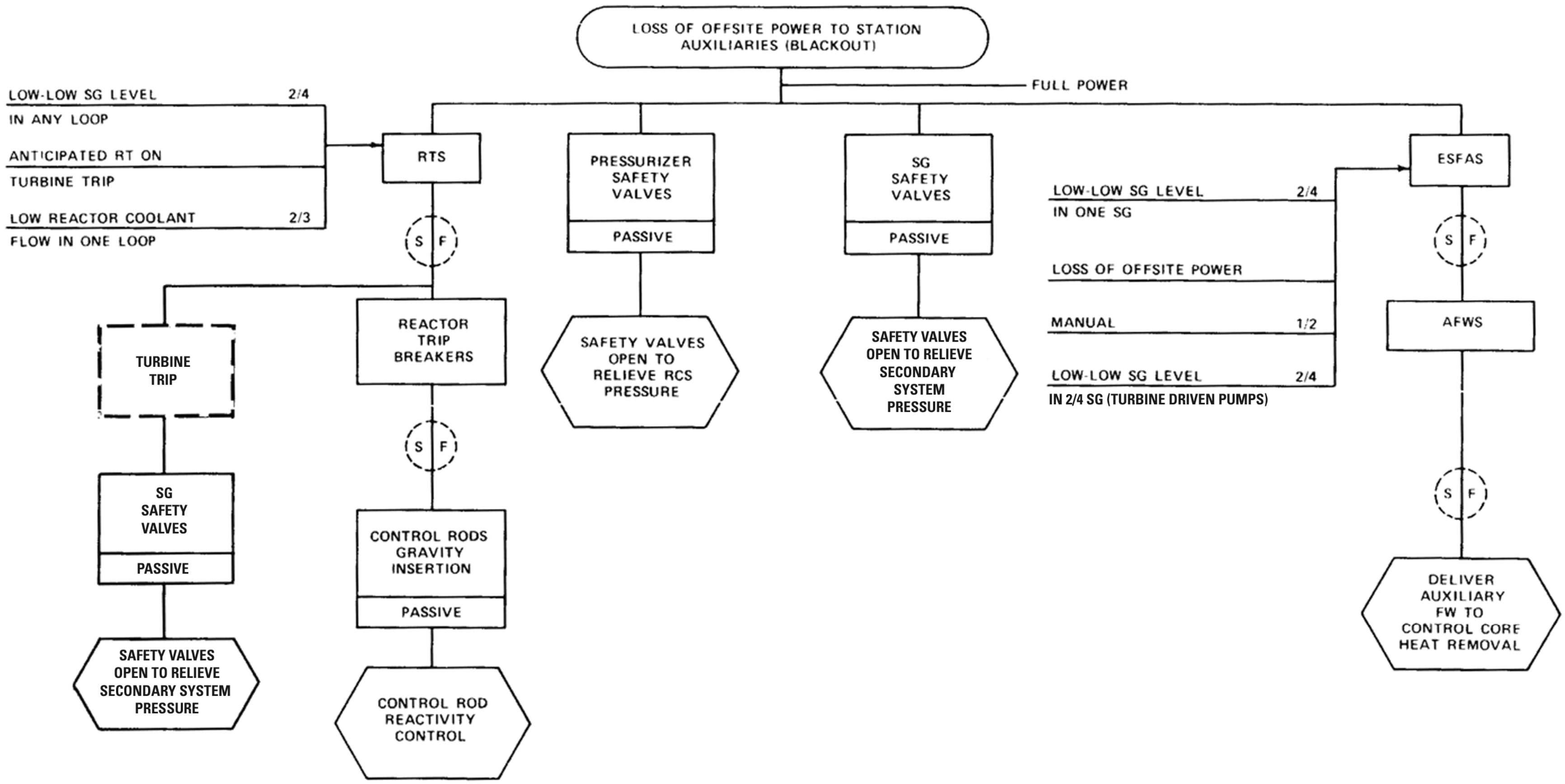
LOSS OF EXTERNAL LOAD

FIGURE 15.0.1–4



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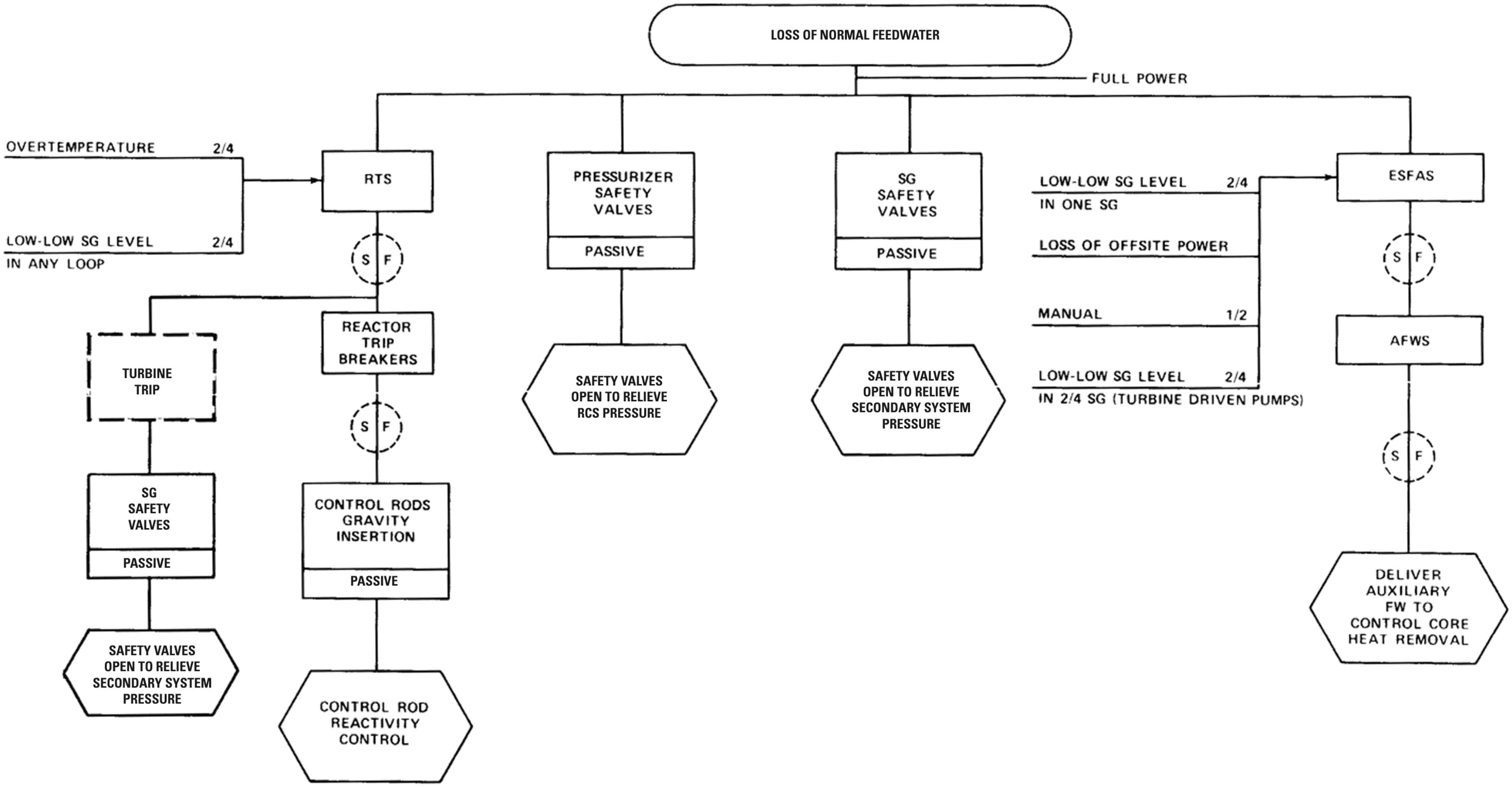
UNIT 1 AND UNIT 2

LOSS OF OFFSITE POWER TO STATION AUXILIARIES (BLACKOUT)

FIGURE 15.0.1–5



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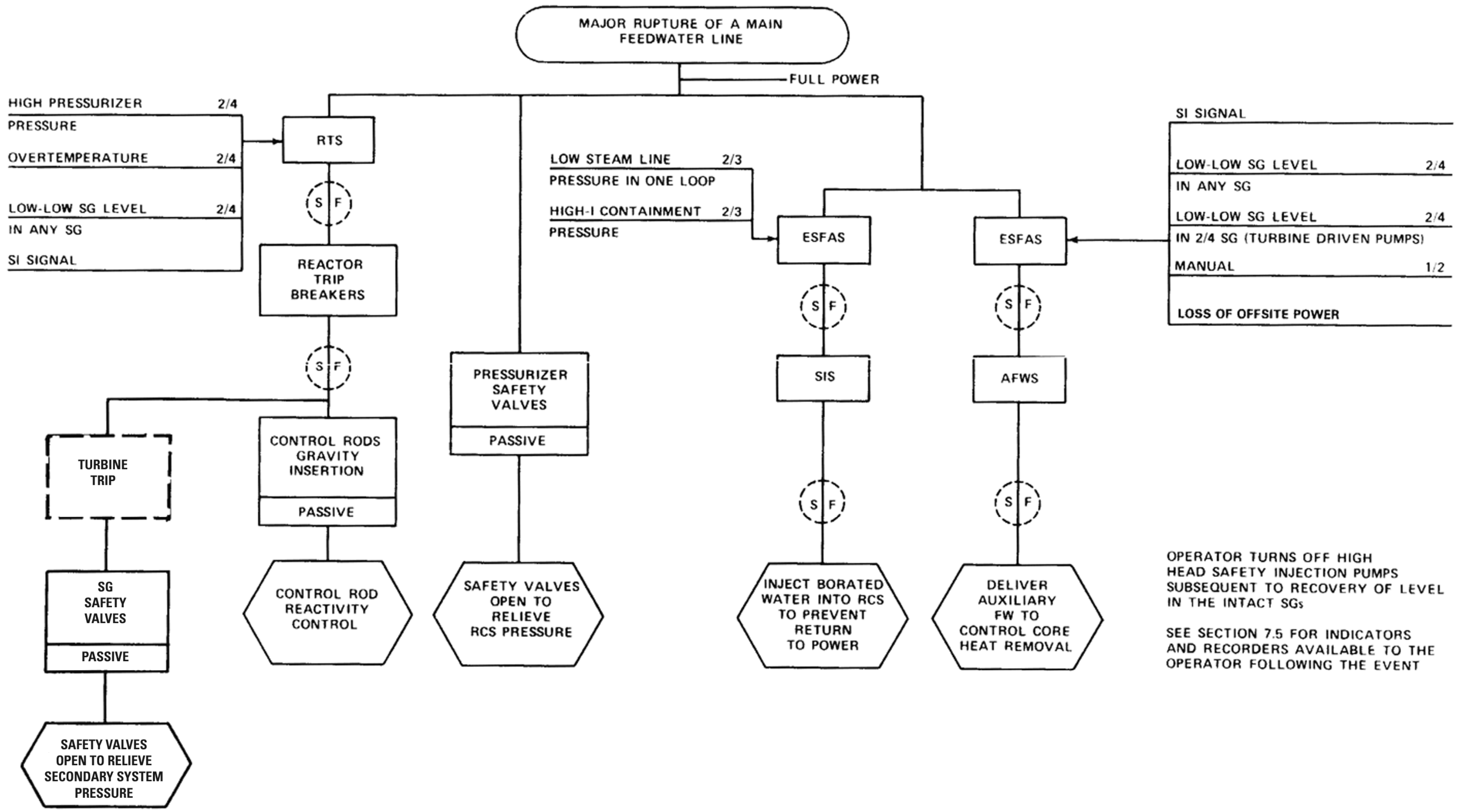
SOUTHERN COMPANY *Energy to Serve Your World®* VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2	LOSS OF NORMAL FEEDWATER
	FIGURE 15.0.1–6



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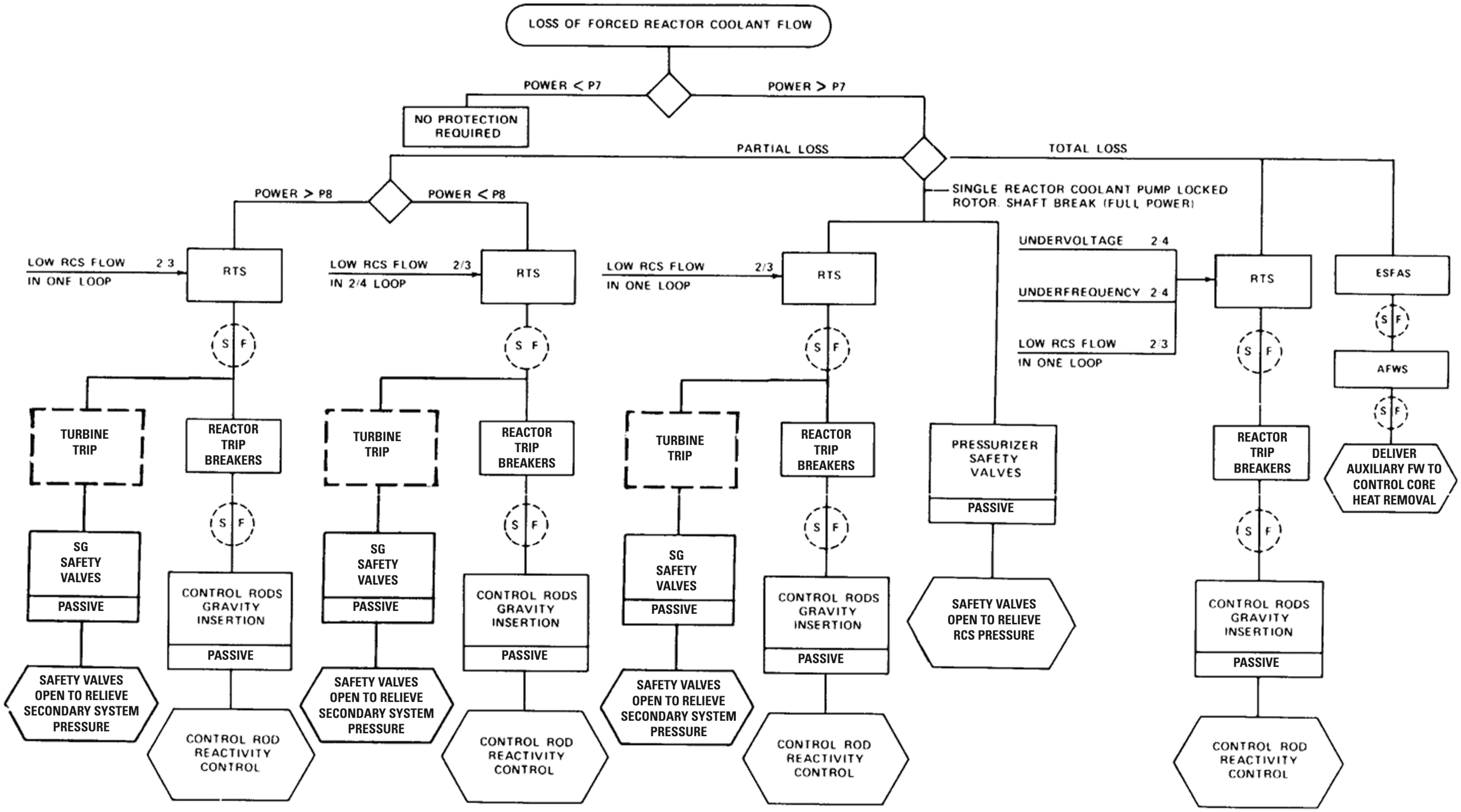
UNIT 1 AND UNIT 2

MAJOR RUPTURE OF A MAIN FEEDWATER LINE

FIGURE 15.0.1–7



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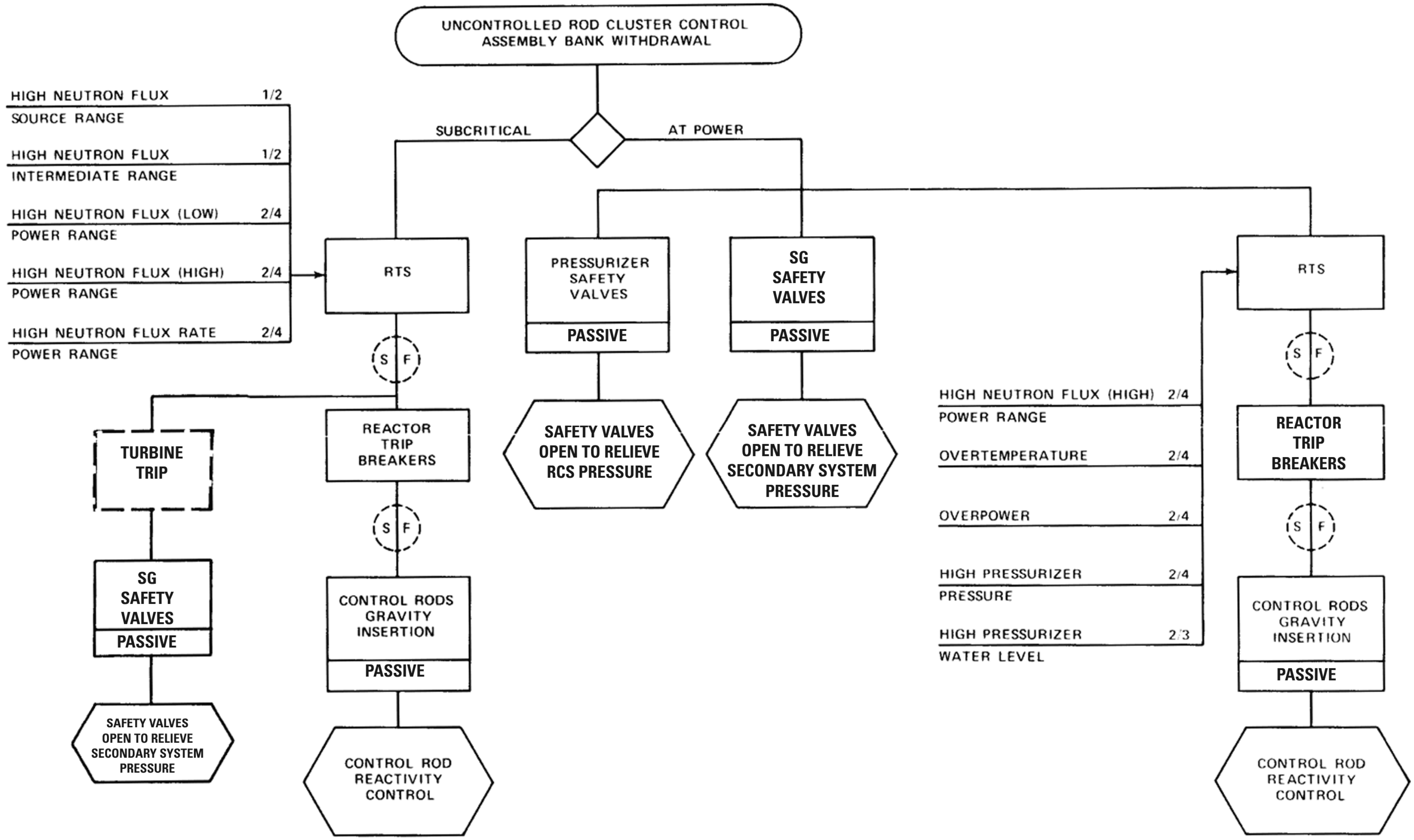
UNIT 1 AND UNIT 2

LOSS OF FORCED REACTOR COOLANT FLOW

FIGURE 15.0.1–8



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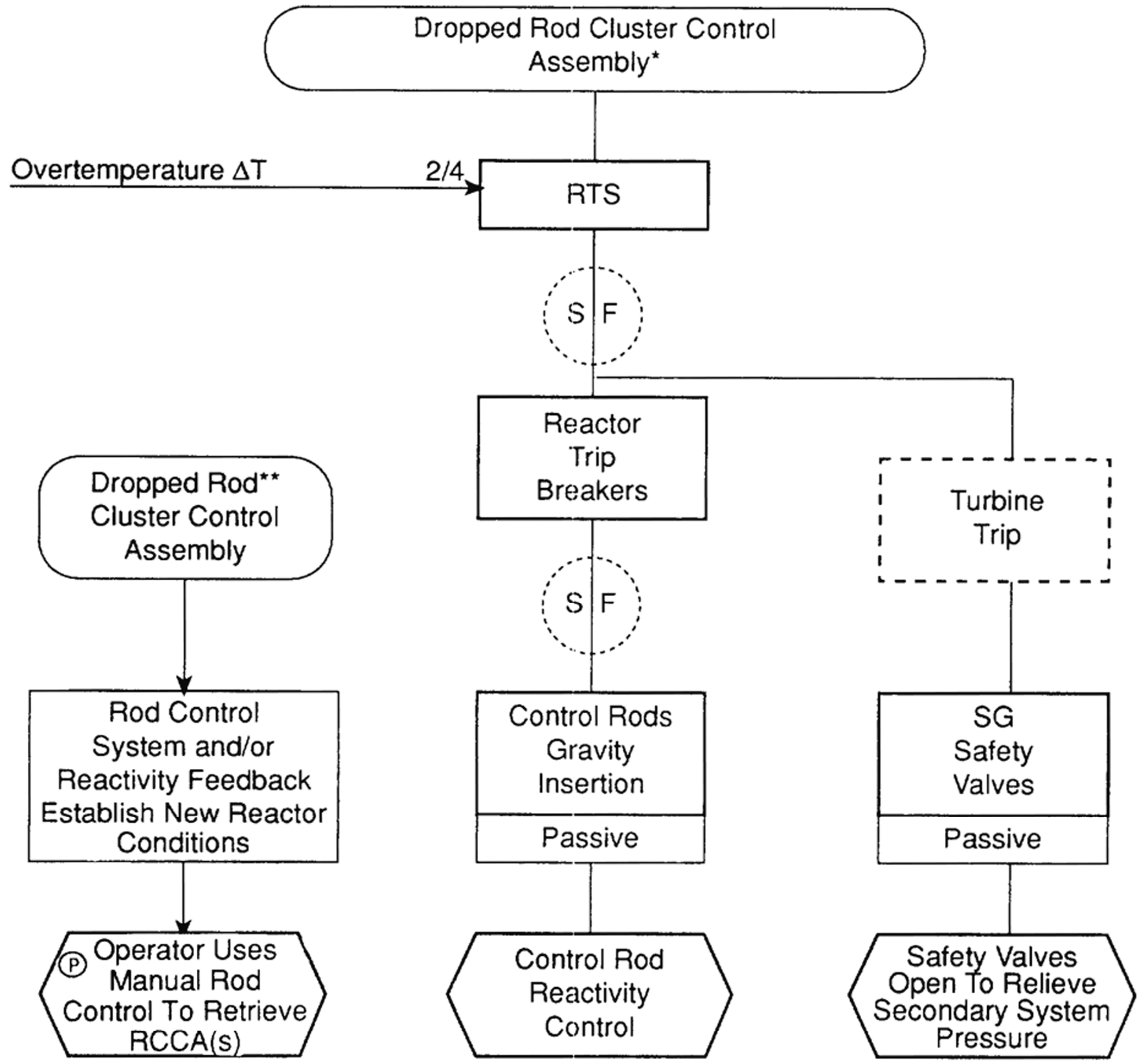
UNIT 1 AND UNIT 2

UNCONTROLLED ROD CLUSTER CONTROL
ASSEMBLY BANK WITHDRAWAL

FIGURE 15.0.1–9



* Trip sequence may occur if rod worth is low and automatic rod withdrawal is rapid. Trip is not modeled in the safety analysis.

** No direct reactor trip occurs.

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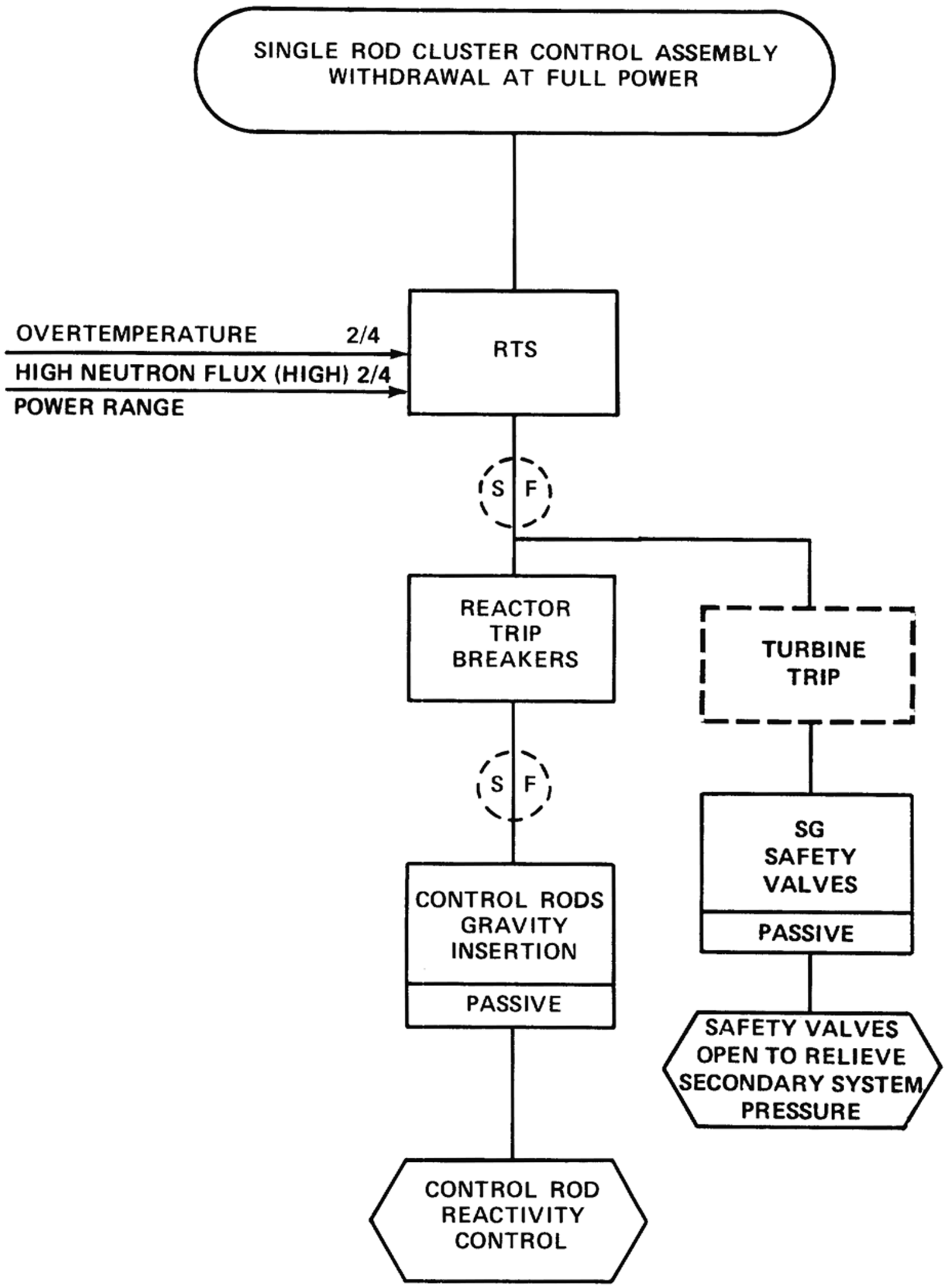

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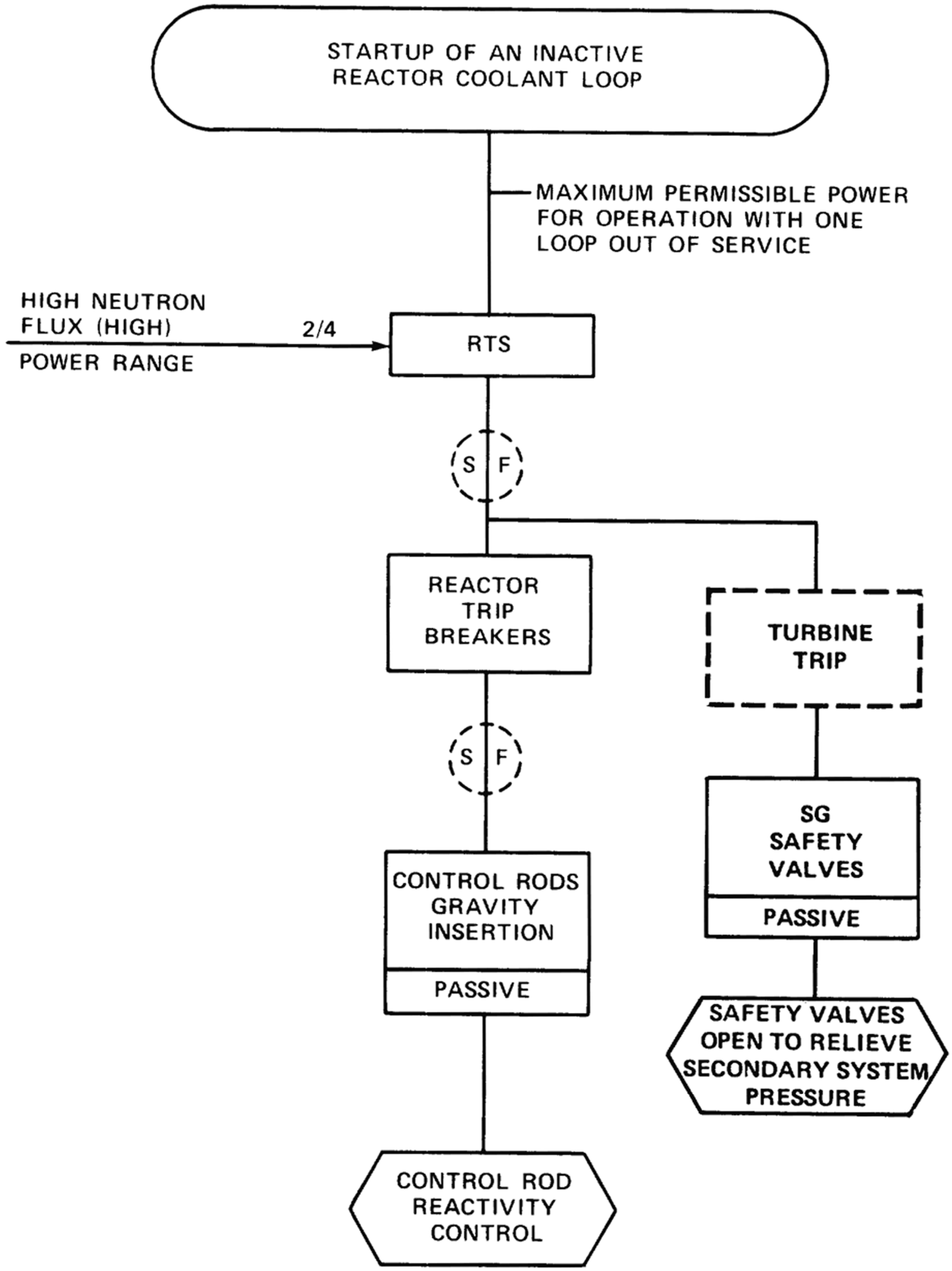
SOUTHERN COMPANY Energy to Serve Your World®	VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2	SINGLE ROD CLUSTER CONTROL ASSEMBLY WITHDRAWAL AT FULL POWER FIGURE 15.0.1–11



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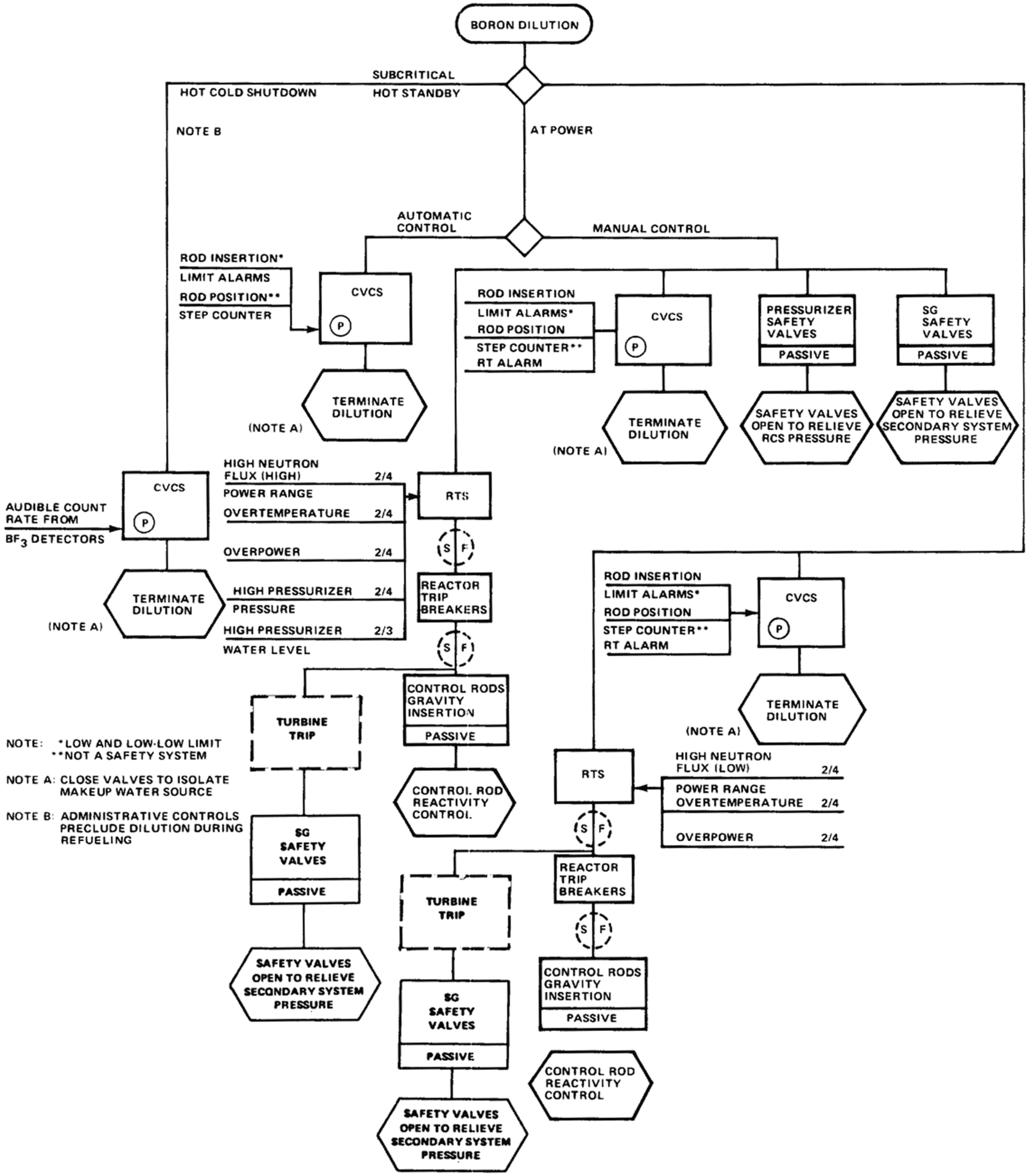
SOUTHERN COMPANY Energy to Serve Your World® VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2	STARTUP OF AN INACTIVE REACTOR COOLANT LOOP
	FIGURE 15.0.1–12



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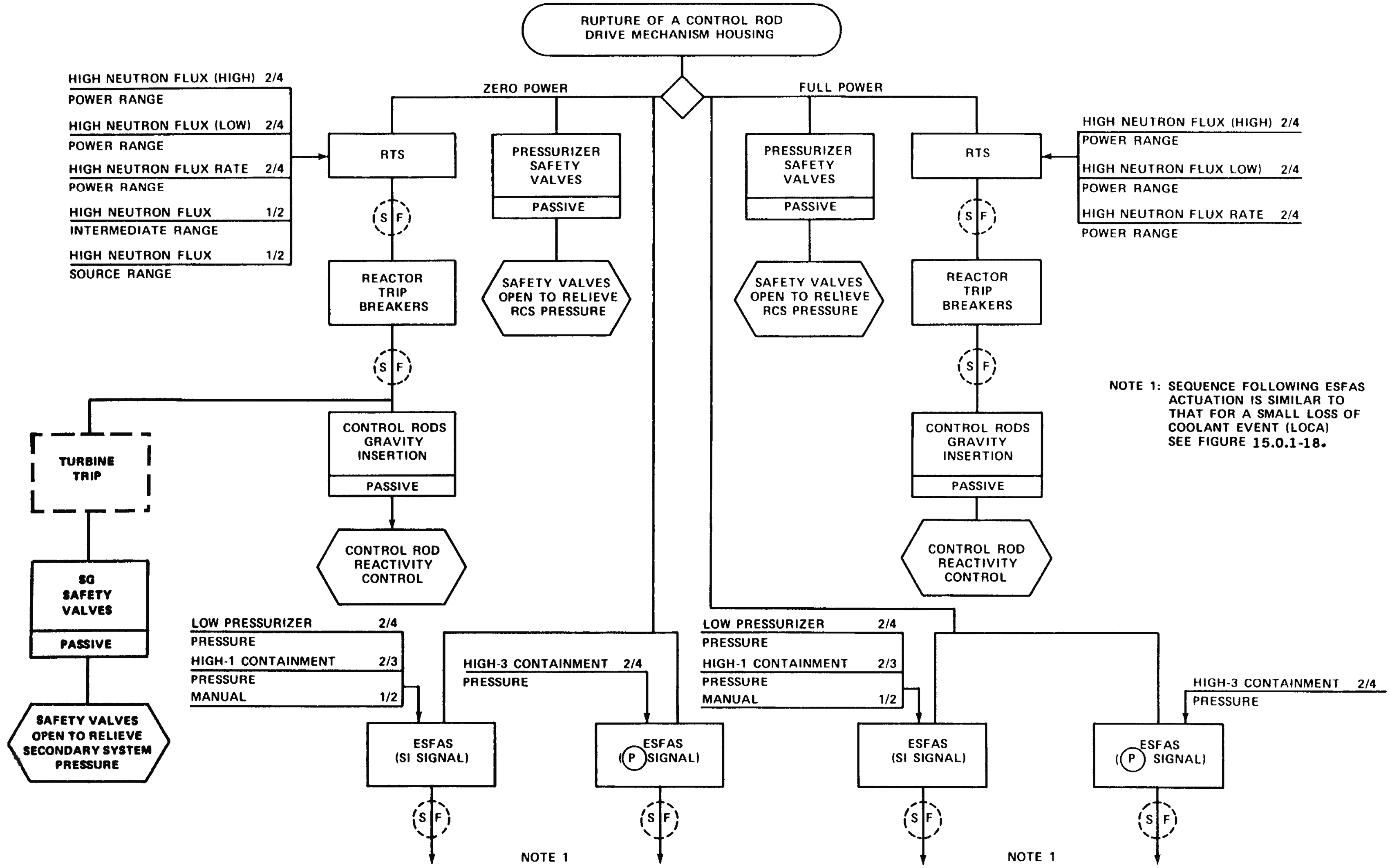
UNIT 1 AND UNIT 2

BORON DILUTION

FIGURE 15.0.1–13



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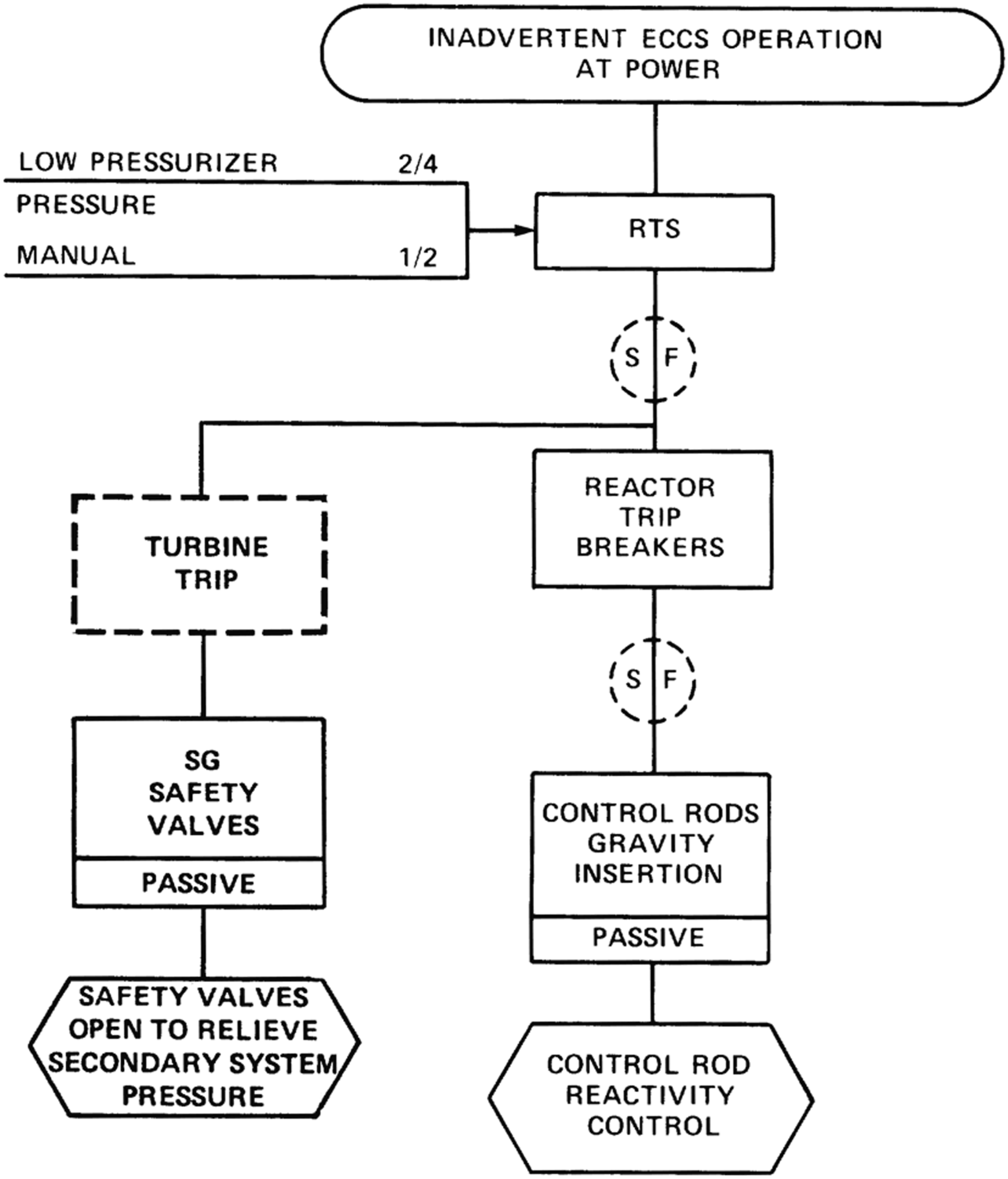
UNIT 1 AND UNIT 2

RUPTURE ON CONTROL ROD DRIVE MECHANISM HOUSING

FIGURE 15.0.1–14



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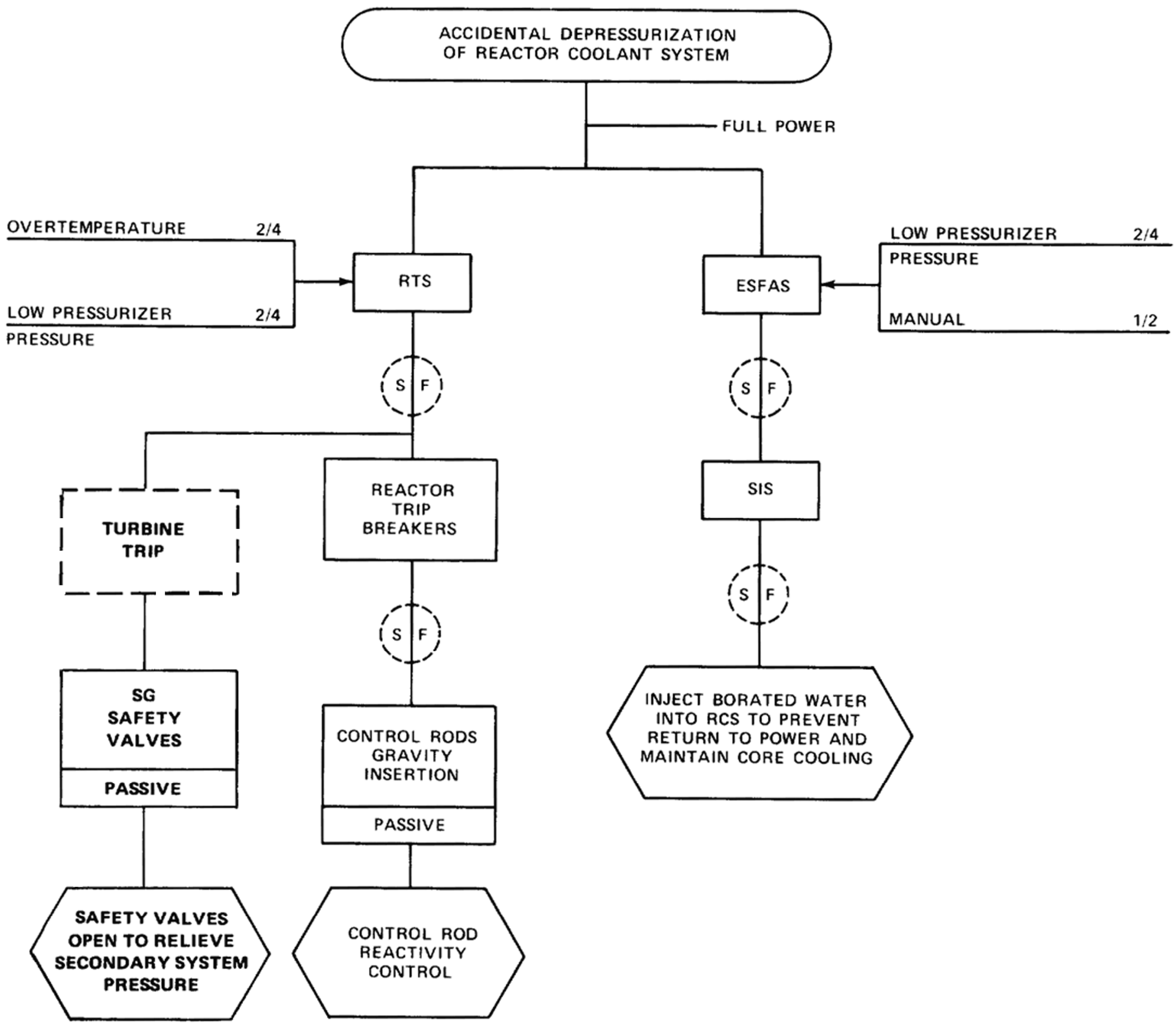
UNIT 1 AND UNIT 2

INADVERTENT ECCS OPERATION AT POWER

FIGURE 15.0.1–15



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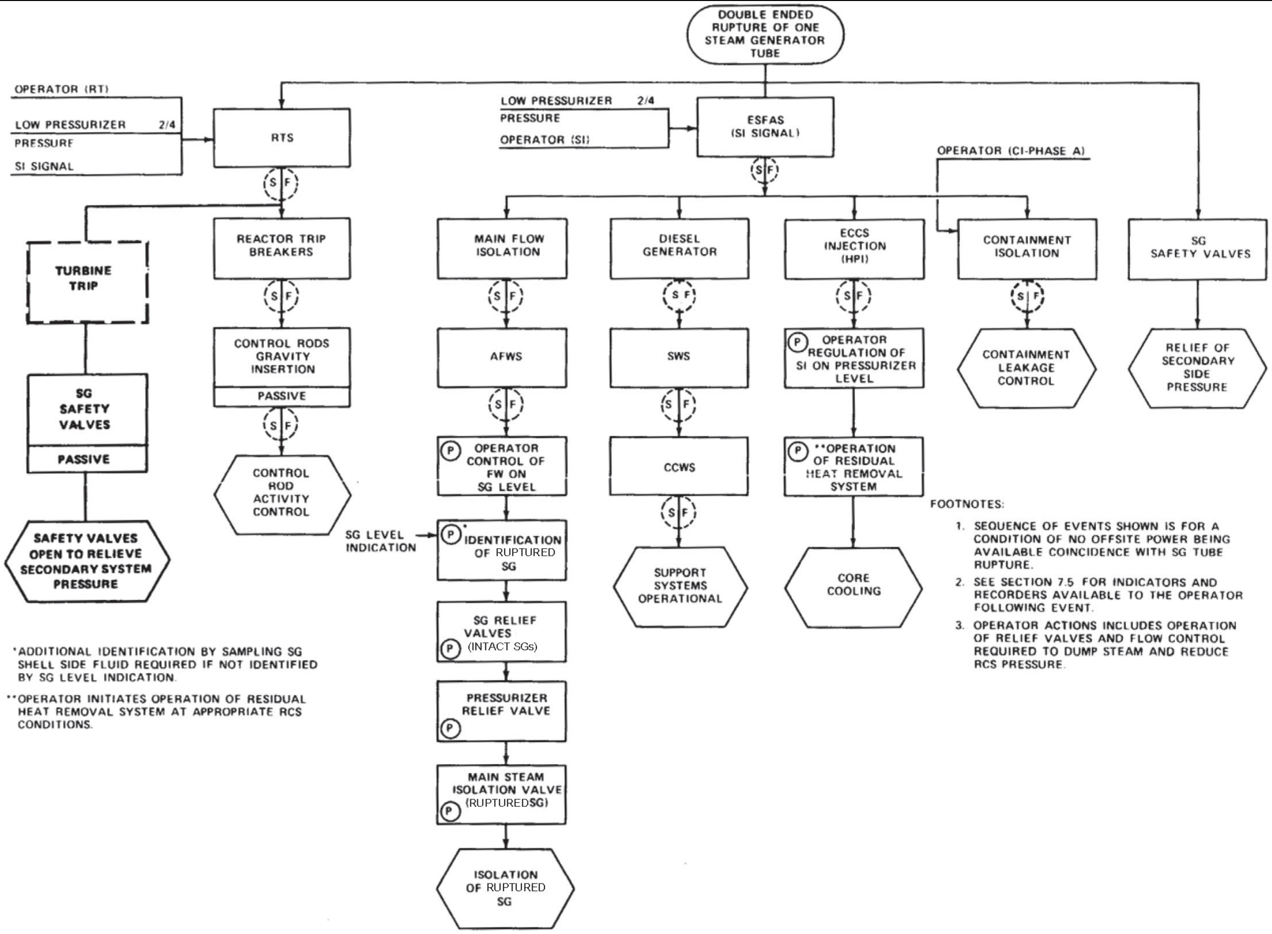
SOUTHERN COMPANY *Energy to Serve Your World®*	**VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2**	ACCIDENTAL DEPRESSUERIZATION OF REACTOR COOLANT SYSTEM
		FIGURE 15.0.1–16



*ADDITIONAL IDENTIFICATION BY SAMPLING SG SHELL SIDE FLUID REQUIRED IF NOT IDENTIFIED BY SG LEVEL INDICATION.

**OPERATOR INITIATES OPERATION OF RESIDUAL HEAT REMOVAL SYSTEM AT APPROPRIATE RCS CONDITIONS.

FOOTNOTES:

1. SEQUENCE OF EVENTS SHOWN IS FOR A CONDITION OF NO OFFSITE POWER BEING AVAILABLE COINCIDENCE WITH SG TUBE RUPTURE.
2. SEE SECTION 7.5 FOR INDICATORS AND RECORDERS AVAILABLE TO THE OPERATOR FOLLOWING EVENT.
3. OPERATOR ACTIONS INCLUDES OPERATION OF RELIEF VALVES AND FLOW CONTROL REQUIRED TO DUMP STEAM AND REDUCE RCS PRESSURE.

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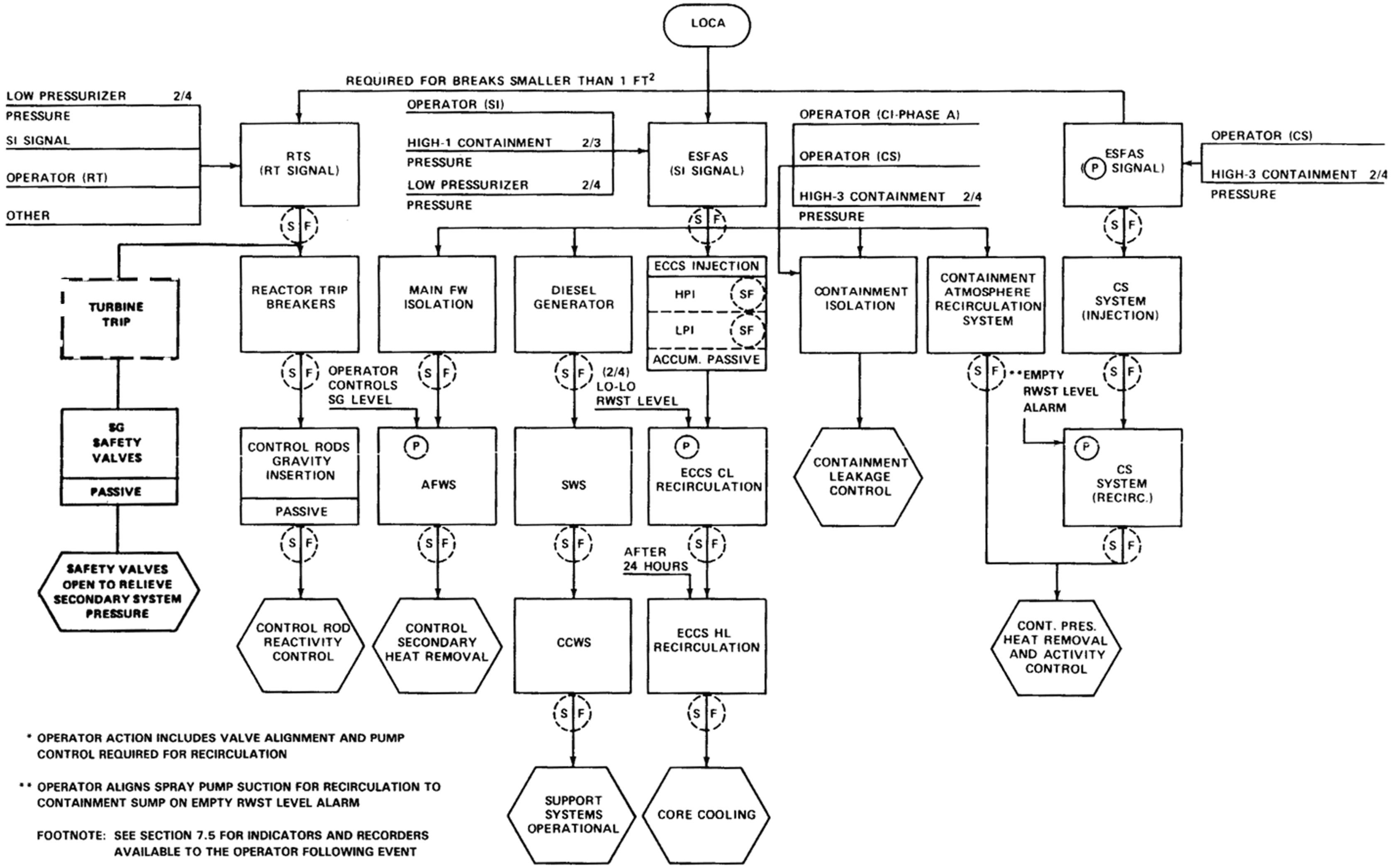
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

STEAM GENERATOR TUBE RUPTURE

FIGURE 15.0.1–17



* OPERATOR ACTION INCLUDES VALVE ALIGNMENT AND PUMP CONTROL REQUIRED FOR RECIRCULATION

** OPERATOR ALIGNS SPRAY PUMP SUCTION FOR RECIRCULATION TO CONTAINMENT SUMP ON EMPTY RWST LEVEL ALARM

FOOTNOTE: SEE SECTION 7.5 FOR INDICATORS AND RECORDERS AVAILABLE TO THE OPERATOR FOLLOWING EVENT

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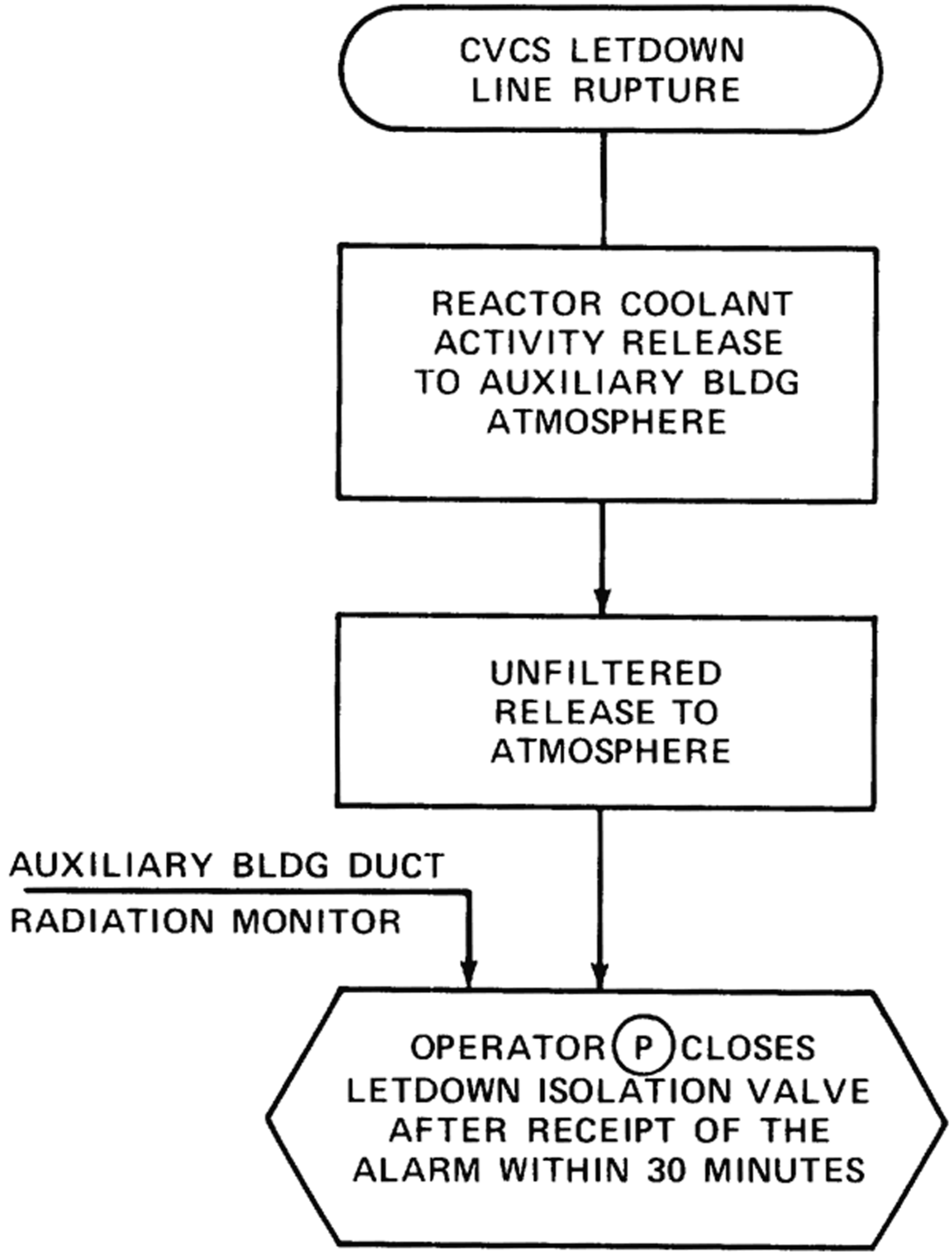
UNIT 1 AND UNIT 2

LOSS–OF–COOLANT ACCIDENT

FIGURE 15.0.1–18



CVCS LETDOWN LINE RUPTURE
REACTOR COOLANT ACTIVITY RELEASE TO AUXILIARY BLDG ATMOSPHERE
UNFILTERED RELEASE TO ATMOSPHERE
AUXILIARY BLDG DUCT RADIATION MONITOR
OPERATOR P CLOSES LETDOWN ISOLATION VALVE AFTER RECEIPT OF THE ALARM WITHIN 30 MINUTES

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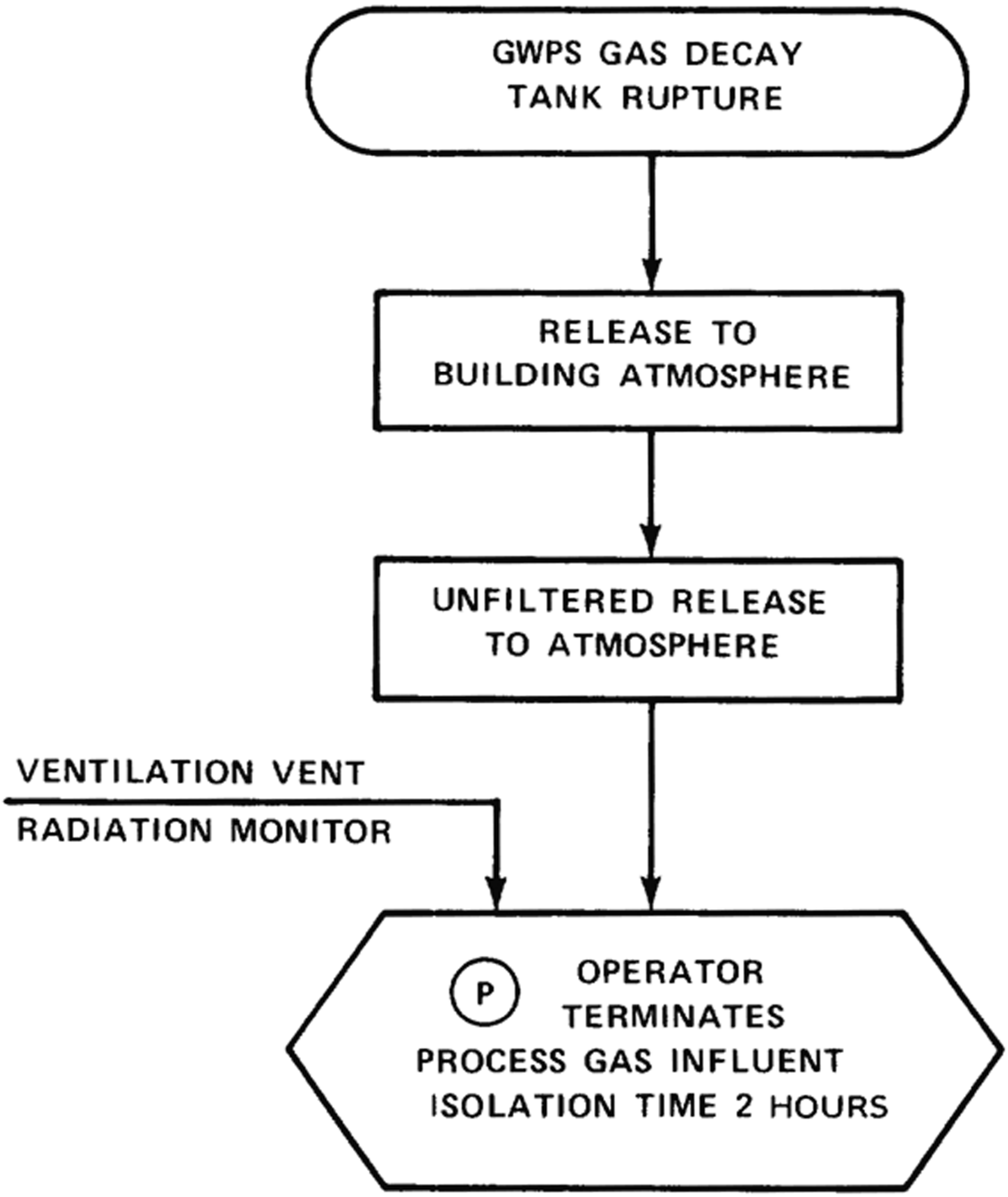

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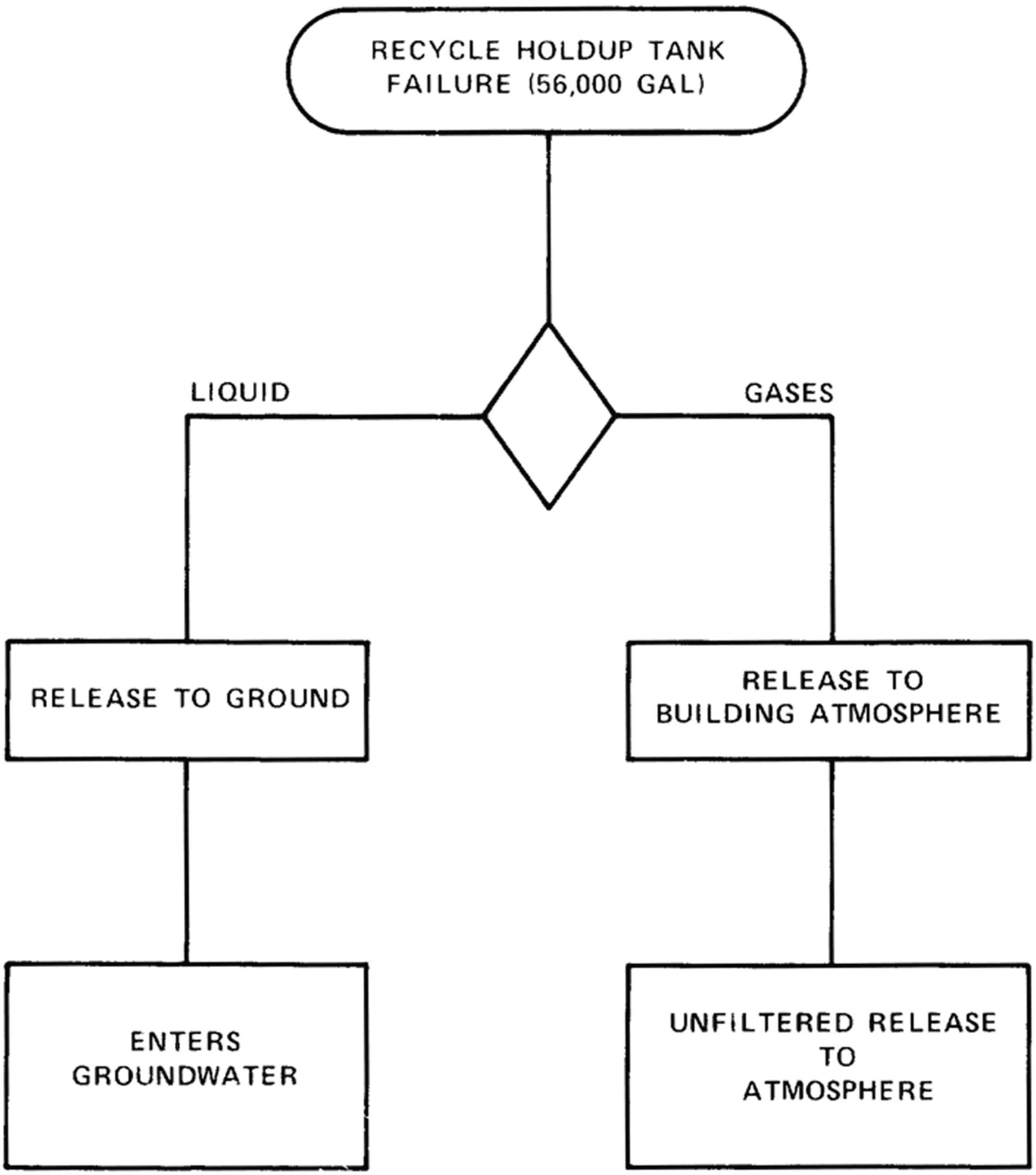
UNIT 1 AND UNIT 2

GWPS GAS DECAY TANK RUPTURE

FIGURE 15.0.1–20



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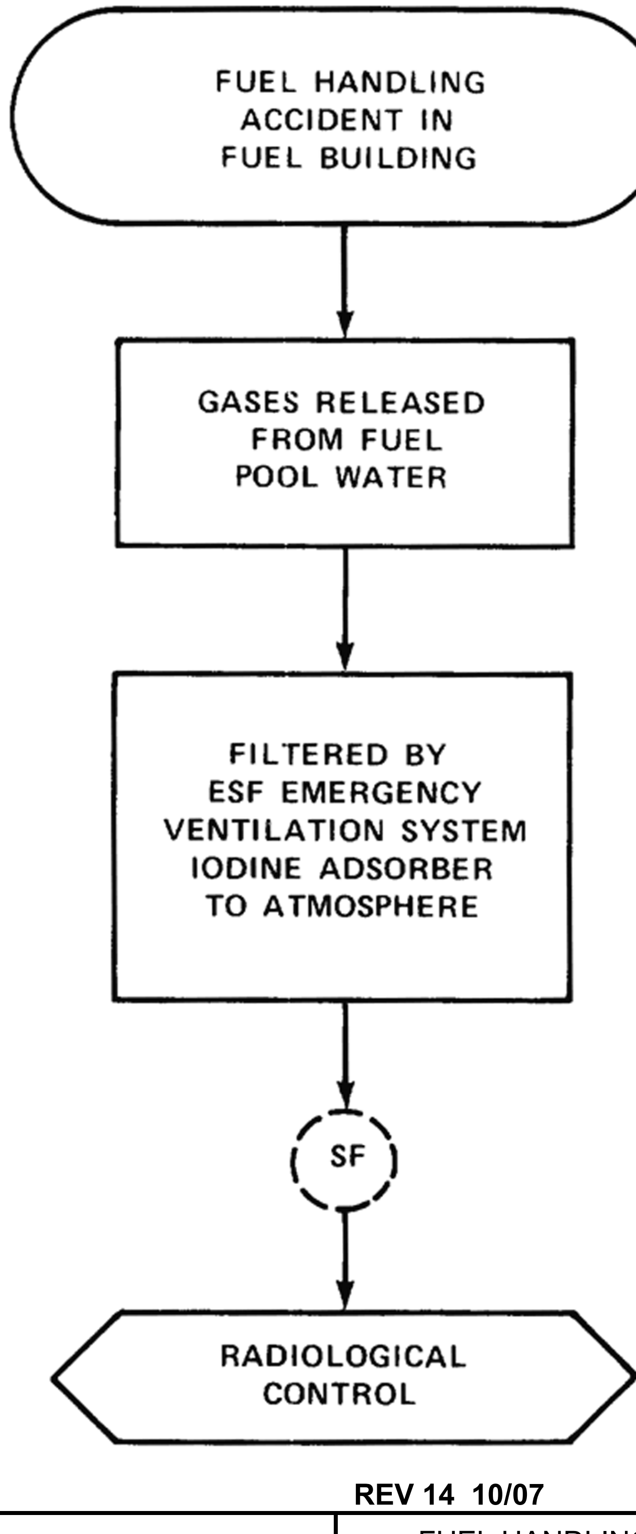
UNIT 1 AND UNIT 2

RECYCLE HOLDUP TANK FAILURE

FIGURE 15.0.1–21



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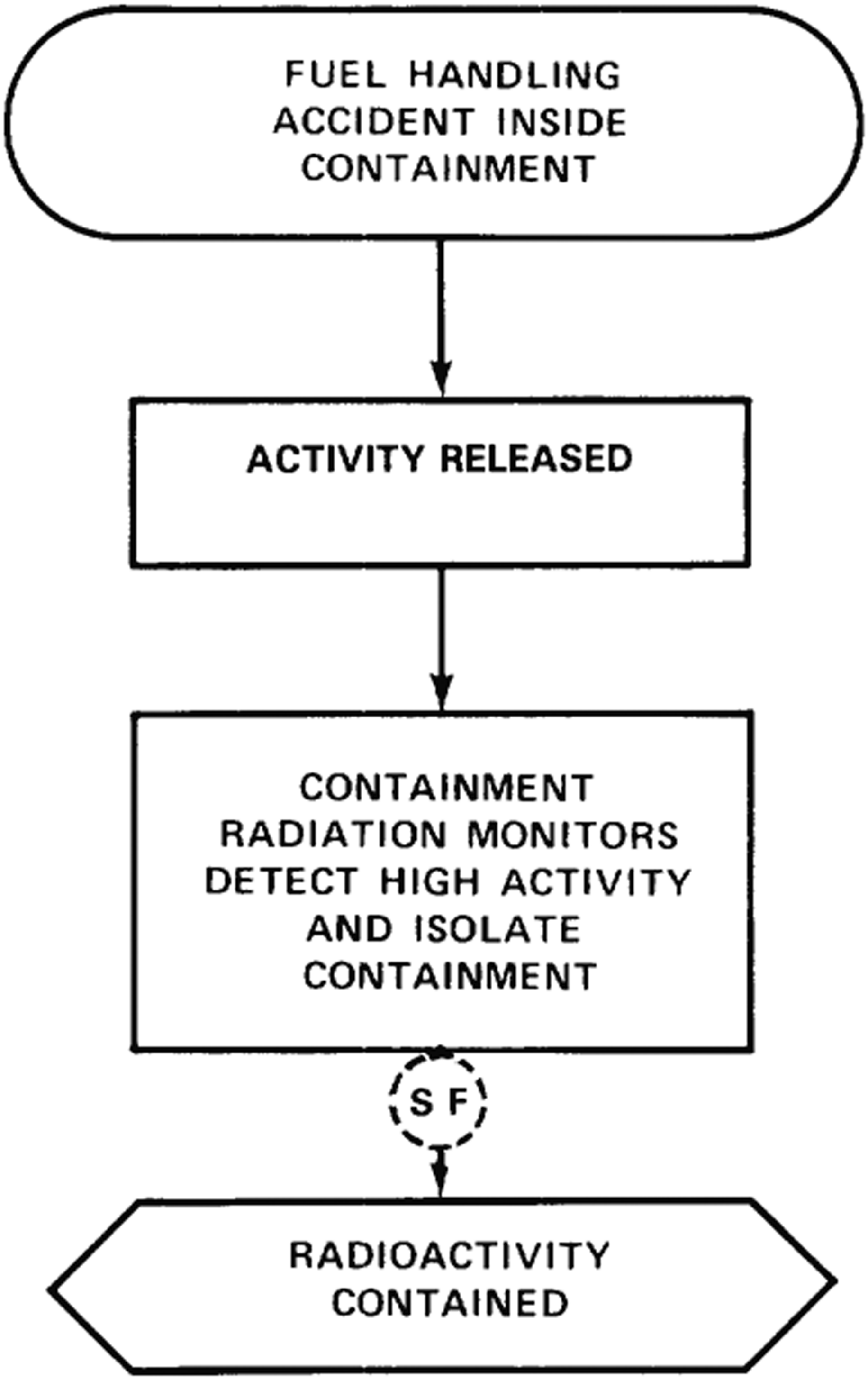
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

FUEL HANDLING ACCIDENT IN FUEL BUILDING

FIGURE 15.0.1–22



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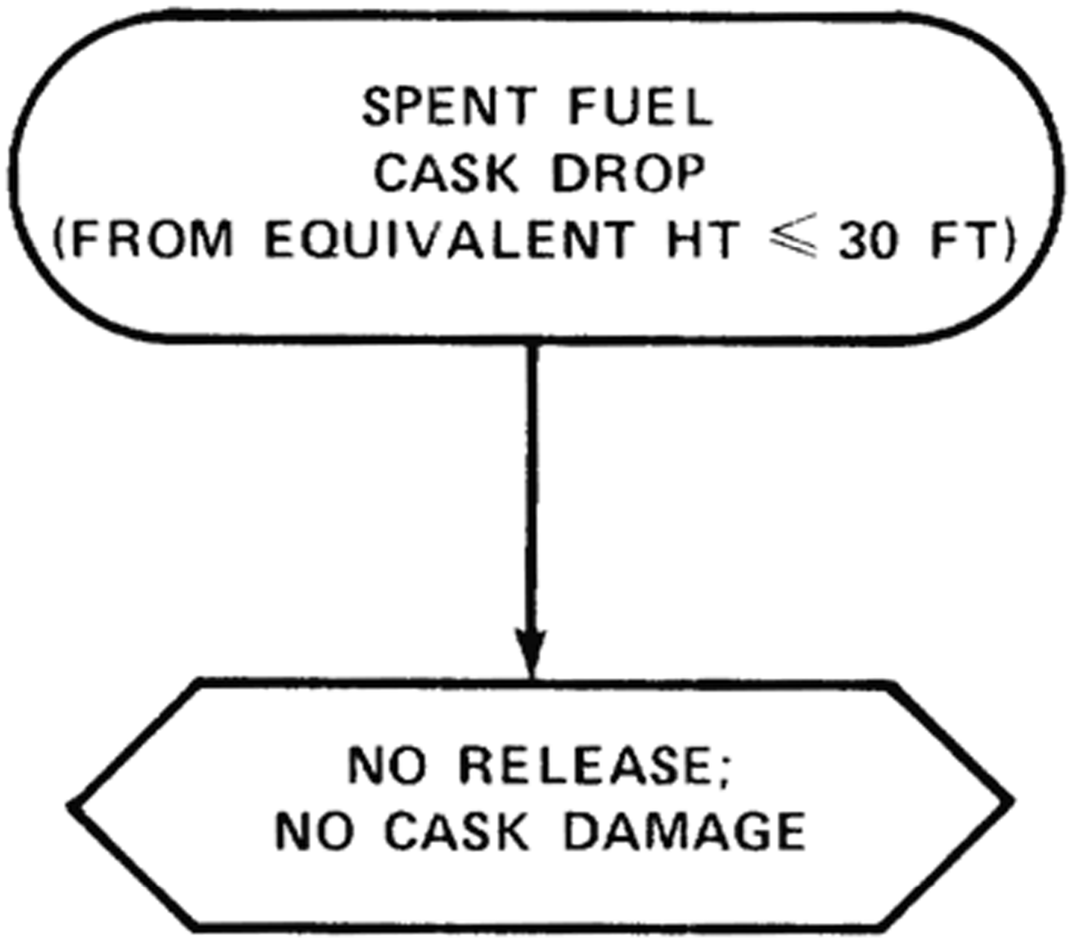
SOUTHERN COMPANY *Energy to Serve Your World®*	**VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2**	FUEL HANDLING ACCIDENT INSIDE CONTAINMENT
		FIGURE 15.0.1–23



SPENT FUEL CASK DROP ACCIDENT

FIGURE 15.0.1–24

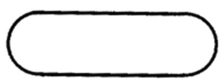
ABBREVIATIONS USED:

- AFWS – AUXILIARY FEEDWATER SYSTEM
- CVCS – CHEMICAL AND VOLUME CONTROL SYSTEM
- ESFAS – ENGINEERED SAFETY FEATURES ACTUATION SYSTEM
- FW – FEEDWATER
- RTS – REACTOR TRIP SYSTEM
- SIS – SAFETY INJECTION SYSTEM
- SI – SAFETY INJECTION
- RT – REACTOR TRIP
- CS – CONTAINMENT SPRAY
- ECCS – EMERGENCY CORE COOLING SYSTEM
- HL – HOT LEG
- CL – COLD LEG
- CCWS – COMPONENT COOLING WATER SYSTEM
- RCS – REACTOR COOLANT SYSTEM
- SWS – SERVICE WATER SYSTEM
- HPI – HIGH PRESSURE INJECTION
- LPI – LOW PRESSURE INJECTION
- CI – CONTAINMENT ISOLATION
- SG – STEAM GENERATOR
- **GWPS – GASEOUS WASTE PROCESSING SYSTEM**

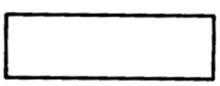
NOTES:

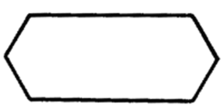
1. FOR TRIP INITIATION AND SAFETY SYSTEM ACTUATION, MULTIPLE SIGNALS ARE SHOWN BUT ONLY A SINGLE SIGNAL IS REQUIRED. THE OTHER SIGNALS ARE BACKUPS.
2. NO TIMING SEQUENCE IS IMPLIED BY POSITION OF VARIOUS BRANCHES. REFER TO EVENT TIMING SEQUENCES PRESENTED IN TABULAR FORM IN PERTINENT ACCIDENT ANALYSIS SECTION OF CHAPTER 15 OF THE FSAR.
3. **PASSIVE MEANS NO ACTION SIGNAL REQUIRED.**

DIAGRAM SYMBOLS:


– EVENT TITLE

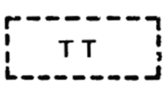

– BRANCH POINT FOR DIFFERENT PLANT CONDITIONS


– SAFETY SYSTEM


– SAFETY ACTION

– SYSTEM REQUIRED TO MEET SINGLE-FAILURE CRITERIA

– MANUAL ACTION REQUIRED DURING SYSTEM OPERATION



– **TURBINE TRIP**

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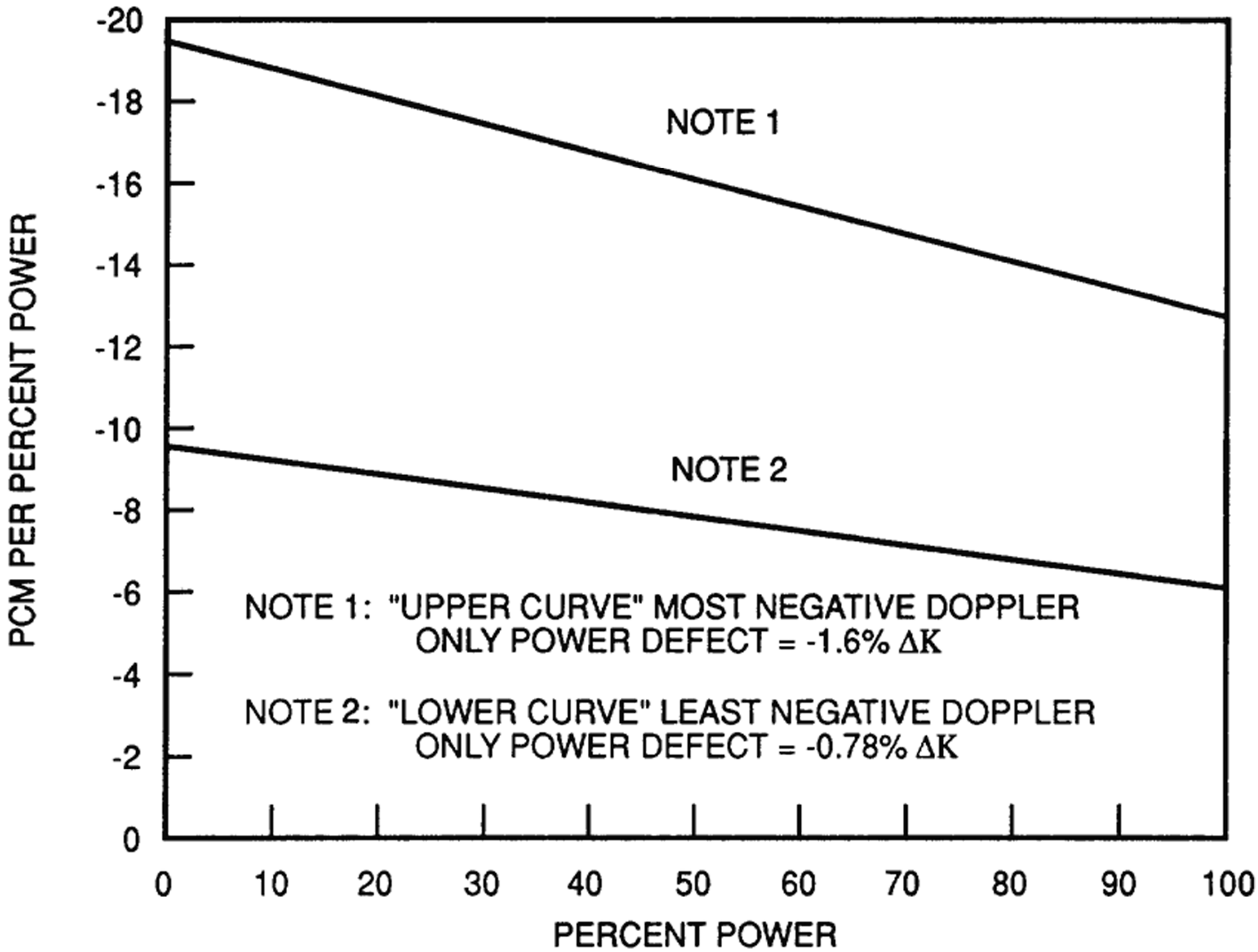
UNIT 1 AND UNIT 2

ABBREVIATIONS AND SYMBOLS USED IN SEQUENCE DIAGRAMS

FIGURE 15.0.1–25



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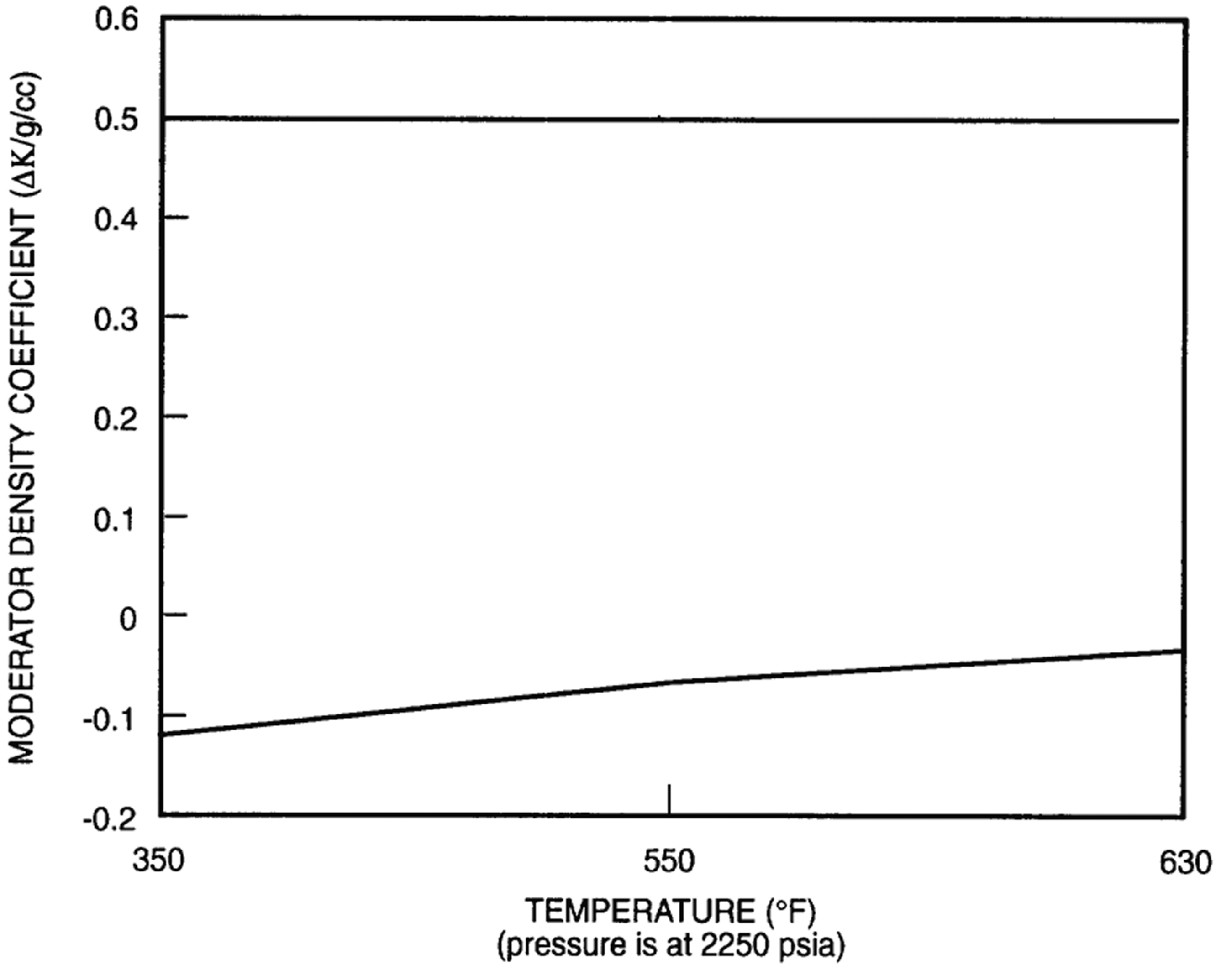
SOUTHERN COMPANY *Energy to Serve Your World®*	**VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2**	DOPPLER POWER COEFFICIENT USED IN ACCIDENT ANALYSES FIGURE 15.0.4–1



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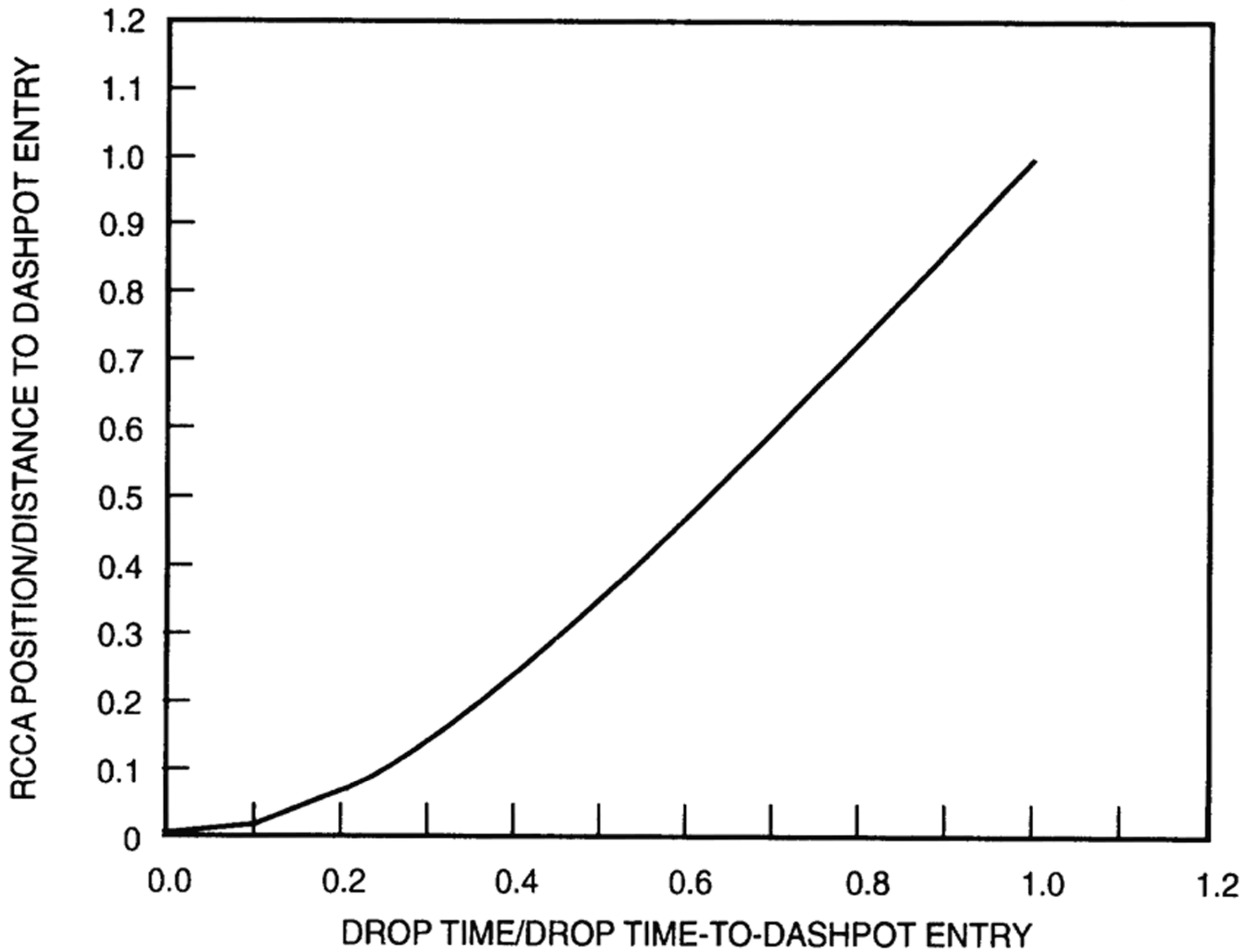
UNIT 1 AND UNIT 2

MODERATOR DENSITY COEFFICIENT USED IN ACCIDENT ANALYSES

FIGURE 15.0.4–2



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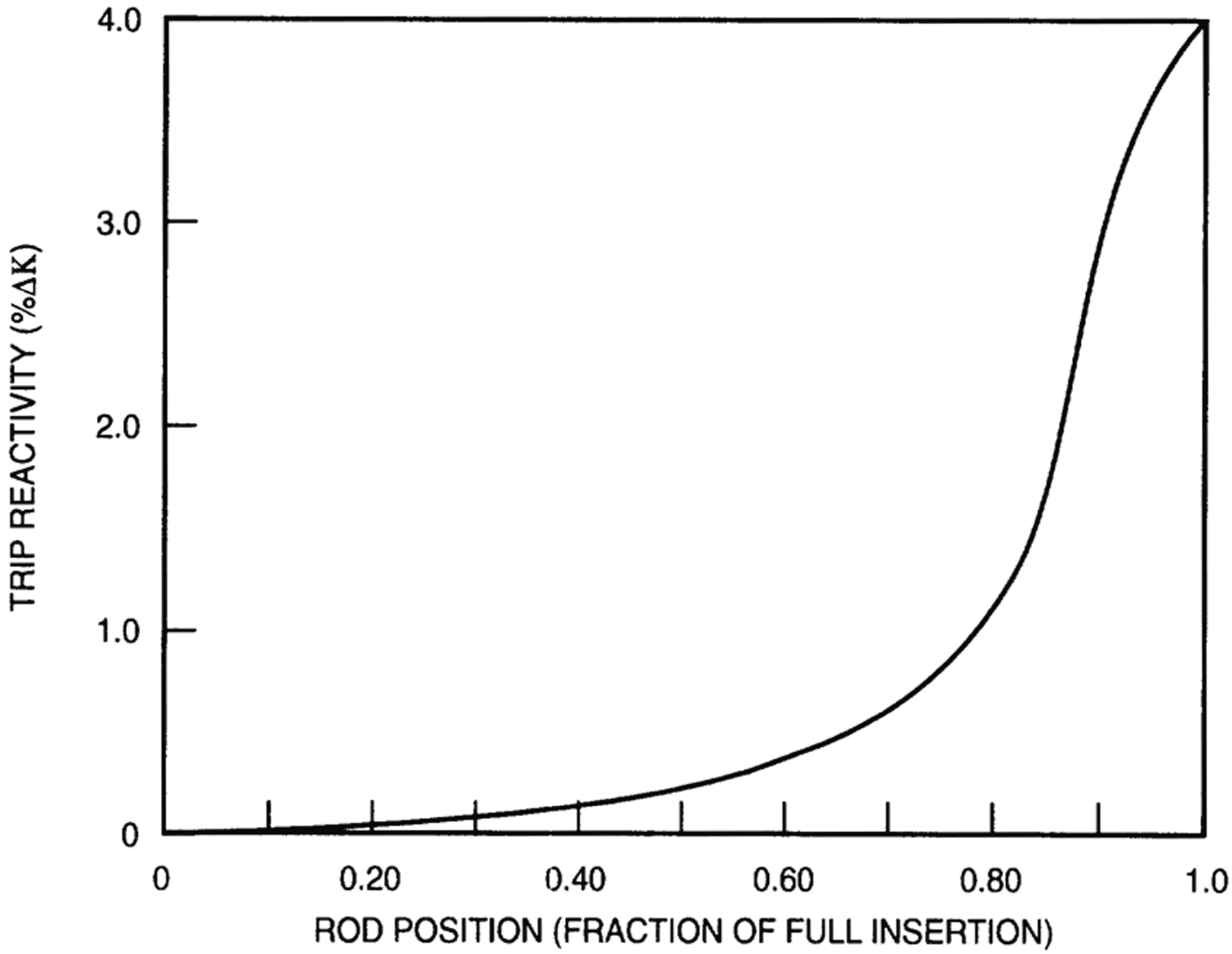
VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

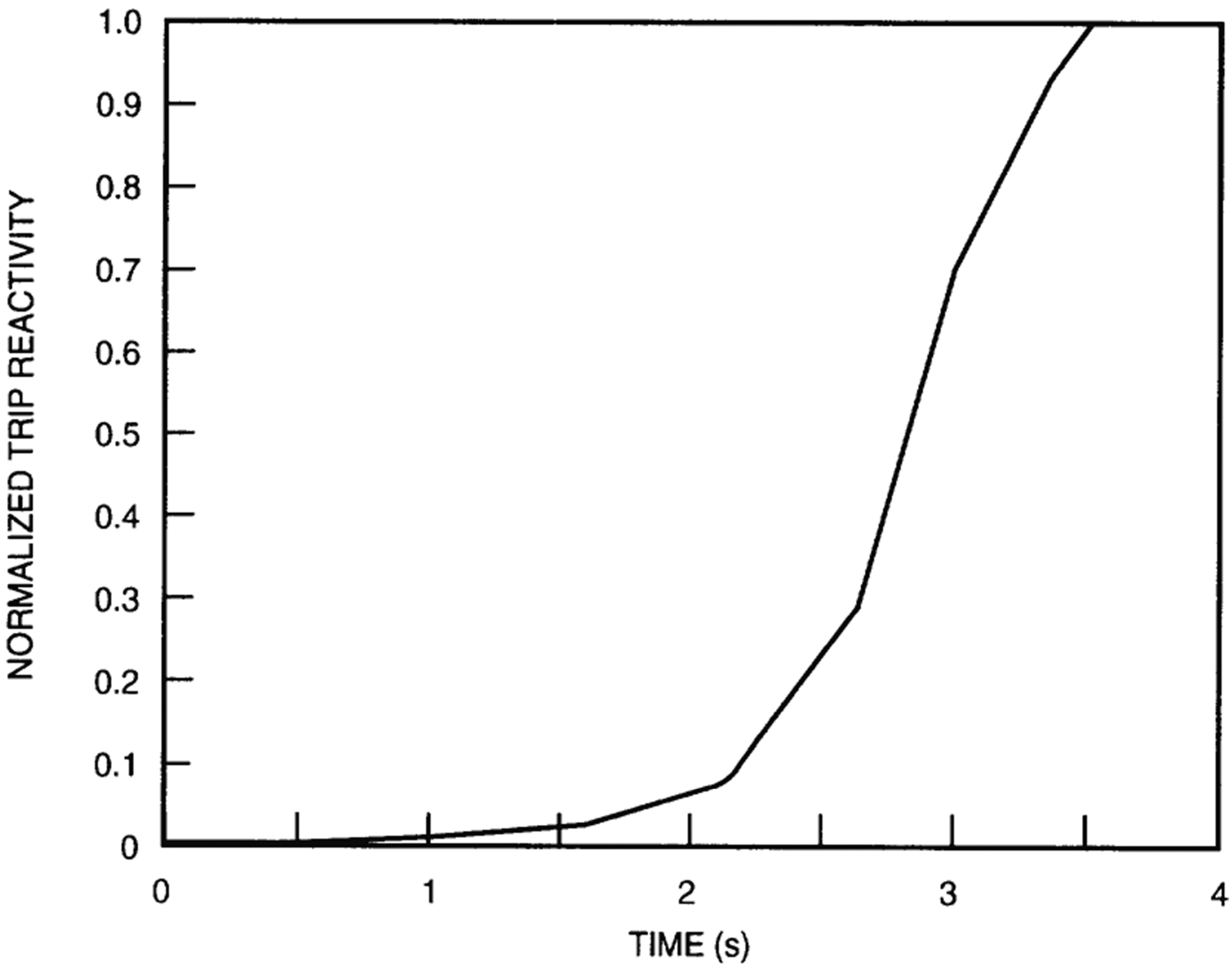
ROD POSITION VERSUS TIME
ON REACTOR TRIP

FIGURE 15.0.5–1



REV 14 10/07

SOUTHERN COMPANY *Energy to Serve Your World®*	VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2	RCCA REACTIVITY WORTH VERSUS FRACTION FULL INSERTION
		FIGURE 15.0.5–2



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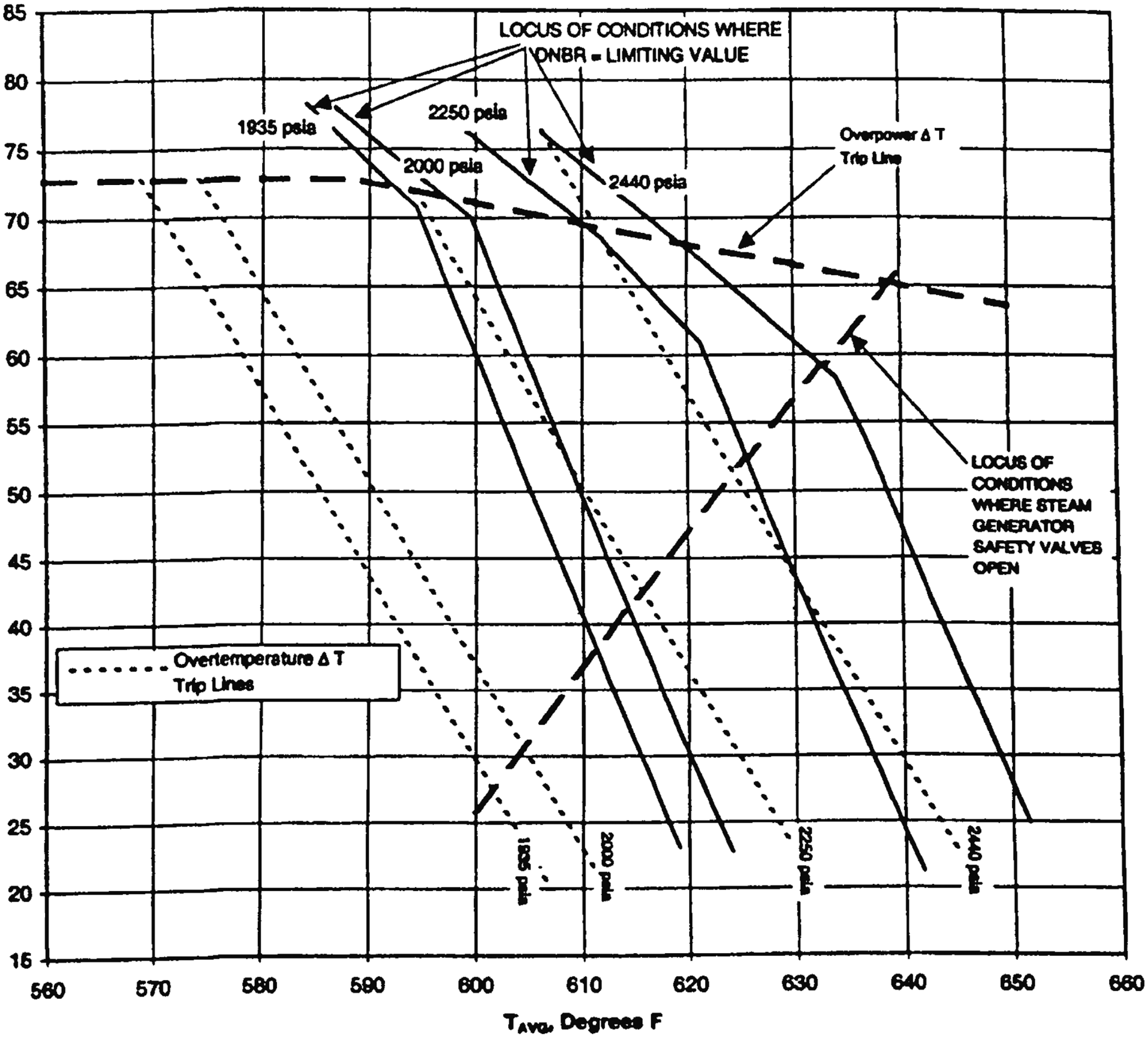
NORMALIZED RCCA BANK WORTH
VERSUS TIME AFTER TRIP

FIGURE 15.0.5–3



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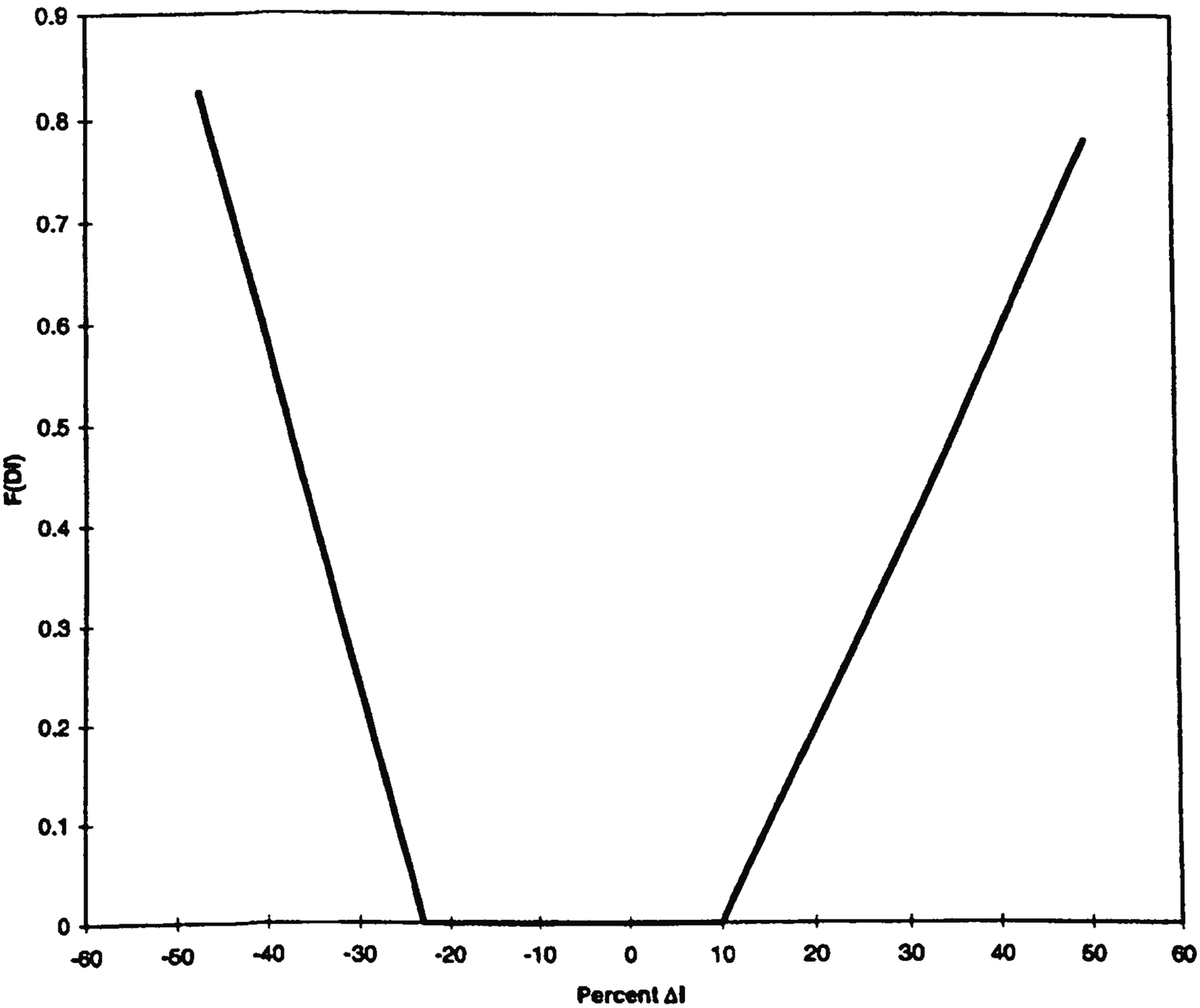
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

OVERTEMPERATURE AND OVERPOWER
ΔT PROTECTION

FIGURE 15.0.6–1



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SOUTHERN COMPANY Energy to Serve Your World®	VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2	F(ΔI) PENALTY FIGURE 15.0.6–2

15.1 INCREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM

Several events have been postulated which could result in an increase in heat removal from the reactor coolant system (RCS) by the secondary system. Detailed analyses are presented for several such events which have been identified as limiting cases.

Discussions of the following RCS cooldown events are presented in this section:

- Feedwater system malfunctions causing a reduction in feedwater temperature.
- Feedwater system malfunctions causing an increase in feedwater flow.
- Excessive increase in secondary steam flow.
- Inadvertent opening of a steam generator relief or safety valve.
- Steam system piping failure.

All of the accidents in this section have been analyzed. It has been determined that the most severe radiological consequences will result from the main steam line break accident discussed in subsection 15.1.5. Therefore, the radiological consequences are only reported for that limiting case.

15.1.1 FEEDWATER SYSTEM MALFUNCTIONS THAT RESULT IN A DECREASE IN FEEDWATER TEMPERATURE

15.1.1.1 Identification of Causes and Accident Description

Reductions in feedwater temperature will result in an increase in core power by initially decreasing reactor coolant temperature. The thermal capacity of the secondary plant and of the RCS attenuates such transients. The high neutron flux trip, OTΔT trip, and OPΔT trip prevent any power increase which could lead to a DNBR less than the limit value.

A low-pressure feedwater train or a high-pressure heater out of service may cause a reduction in feedwater temperature. If a spurious heater drain pump trips, there is a sudden reduction in feedwater inlet temperature to the steam generators. At power, this increased subcooling will create a greater load demand on the RCS.

With the plant at no-load conditions, the addition of cold feedwater will cause a decrease in RCS temperature and a reactivity insertion due to the effects of the negative moderator temperature coefficient of reactivity; however, the rate of energy change is reduced as load and feedwater flow decrease, so the transient is less severe than the full power case.

The net effect on the RCS due to a reduction in feedwater temperature is similar to the effect of increasing secondary steam flow; i.e., the reactor will reach a new equilibrium condition at a power level corresponding to the new steam generator ΔT.

This is an American Nuclear Society (ANS) Condition II incident.

The protection available to mitigate the consequences of a decrease in feedwater temperature is the same as that for an excessive steam flow increase, as discussed in subsection 15.0.8 and listed in table 15.0.8-1.

15.1.1.2 Analysis of Effects and Consequences

15.1.1.2.1 Method of Analysis

This transient is analyzed by computing conditions at the feedwater pump inlet following the removal of a low-pressure feedwater train or a high-pressure heater from service. These feedwater conditions are then used to recalculate a heat balance for the RCS loop containing a reduced number of feedwater heaters in service. This heat balance gives the new feedwater conditions at the steam generator inlet.

The following assumptions are made:

A. Plant initial power level corresponds to design NSSS thermal output at the uprated condition

B. One string of feedwater heaters is isolated

This accident analysis employs the RTDP with the initial conditions shown in tables 15.0.3-2 and 15.0.3-3.

No single active failure in any plant systems or equipment will adversely affect the consequences of the accident.

15.1.1.2.2 Results

Isolation of a string of low-pressure feedwater heaters causes a reduction in feedwater temperature, which increases the thermal load on the primary system. The calculated reduction in feedwater temperature is less than 30°F. This reduction in feedwater temperature results in an increase in heat load on the primary system of less than 10 percent of full power. Thus, increased thermal load due to a spurious heater drain pump trip would result in a transient very similar (but of a reduced magnitude) to that presented in subsection 15.1.3 for an excessive increase in secondary steam flow transient. The consequences of a 10-percent step load increase are evaluated in subsection 15.1.3; therefore, there is no presentation of the results of this analysis.

15.1.1.3 Conclusions

The decrease in feedwater temperature transient is less severe than the increase in secondary steam flow event (subsection 15.1.3). Based on results presented in subsection 15.1.3, the applicable acceptance criteria for the decrease in feedwater temperature event have been met.

15.1.2 FEEDWATER SYSTEM MALFUNCTIONS THAT RESULT IN AN INCREASE IN FEEDWATER FLOW

15.1.2.1 Identification of Causes and Accident Description

Addition of excessive feedwater will cause an increase in core power by decreasing reactor coolant temperature. The thermal capacity of the secondary plant and the RCS attenuates such transients. Although not relied upon for mitigation of this transient, the high neutron flux trip, OPΔT trip, and OTΔT trip prevent any power increase which could lead to DNBR less than the minimum allowable value in the event that the steam generator high-high water level protection does not actuate. The results of the analysis of the addition of excessive feedwater show that the minimum DNBR for this transient is reached prior to reactor trip. The event is self-limiting in that the cooldown due to the excess feedwater results in a limited power increase. Once the power has increased to its maximum value, the DNBR will stabilize. The high-high water level signal (P-14) terminates the excessive feedwater flow. It also initiates a turbine trip that results in a reactor trip, and trips the SG feedwater pumps. There are backup trips that are generated but not credited in the safety analysis.

Should the turbine trip not initiate a reactor trip, the subsequent reduction in steam generator water level will initiate a reactor trip on steam generator low-low level.

The full opening of a feedwater regulating valve due to a feedwater control system malfunction or an operator error may cause excessive feedwater flow. At power conditions, this excess flow causes a greater load demand on the RCS due to increased subcooling in the steam generator. With the plant at no-load conditions, the addition of cold feedwater may cause a decrease in RCS temperature and thus a reactivity insertion due to the effects of the negative moderator temperature coefficient of reactivity.

The steam generator high-high water level ESF (P-14) signal does the following:

- Closes the feedwater valves.
- Closes the feedwater pump discharge valves.
- Trips the turbine.
- Trips the main feedwater pumps.
- Prevents continuous addition of excessive feedwater.

When power is greater than P-9, the turbine trip signal initiates a reactor trip.

This is an ANS Condition II incident.

15.1.2.2 Analysis of Effects and Consequences

15.1.2.2.1 Method of Analysis

The analysis of the excessive heat removal due to a feedwater system malfunction transient uses the detailed digital computer code LOFTRAN (reference 1). This code simulates the neutron kinetics, RCS, pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generator, steam generator safety valves, and feedwater control system. The code computes pertinent plant variables including temperatures, pressures, and power level.

The purpose of the system analysis is to demonstrate acceptable plant behavior in the event of an excessive feedwater addition due to a control system malfunction or operator error which allows a feedwater addition due to a control system malfunction or operator error which allows a feedwater regulating valve to open fully. Cases analyzed assuming a conservatively large negative moderator temperature coefficient follow:

- Accidental full opening of one main feedwater regulating valve and one bypass feedwater regulating valve or two main feedwater regulating valves and two bypass feedwater regulating valves with the reactor just critical at zero power conditions (Mode 2) with the reactor in both automatic and manual rod control.

- Accidental full opening of one main feedwater regulating valve and one bypass feedwater regulating valve or two main feedwater regulating valves and two bypass feedwater regulating valves with the reactor at full power assuming automatic and manual rod control.

The calculation of the reactivity insertion rate following a feedwater system malfunction uses the following assumptions:

A. For the feedwater regulating valve accident at full power, one feedwater regulating valve malfunctions, which results in a step increase in flow to one steam generator to 157 percent of nominal feedwater flow to one steam generator. Additionally, a feedwater regulating valve malfunction, which results in a step increase in flow to two steam generators to 177 percent of nominal feedwater flow to each steam generator has been addressed.

B. For the feedwater regulating valve accident analyzed at no-load conditions (Mode 2), a main feedwater regulating valve and a bypass feedwater regulating valve malfunction occurs which results in a step increase in flow to one steam generator from 0 to 225 percent of the nominal full load value for one steam generator. Additionally, two main feedwater regulating valves and two bypass feedwater regulating valves malfunction, which results in a step increase in flow to two steam generators from 0 to 225 percent of the nominal full load value for each steam generator.

C. For the no-load condition, feedwater temperature is at a conservatively low value of 32°F.

D. No credit is taken for the heat capacity of the RCS and steam generator thick metal in attenuating the resulting plant cooldown.

E. The feedwater flow resulting from a fully open regulating valve terminates by a steam generator high-high level ESF (P-14) signal which closes all feedwater regulating and isolation valves, and trips the main feedwater pumps. Subsequently, the steam generator level reduces to the low-low level setpoint. The steam generator low-low level signal initiates a reactor trip. (See section 7.2.)

F. The analysis assumes the RCS flow equivalent to the operation of four RCPs.

G. The analysis considers cases with and without automatic rod control. The automatic rod control cases assume that the rods can automatically insert and withdraw.

H. No credit is taken for reactor trip on turbine trip > P-9.

The analysis uses RTDP methodology in the determination of initial reactor power, pressure, and RCS temperature. This analysis is at full power. (See tables 15.0.3-2 and 15.0.3-3.)

Normal reactor control systems are not required to function. For the purpose of this analysis, reactor trip is assumed to occur on steam generator water level low-low following feedwater isolation. Reactor trip may also occur on overpower or as a result of turbine trip and feedwater isolation. However, these functions are not credited in the analysis. No single active failure will prevent operation of the reactor protection system. A discussion of anticipated transients without trip considerations is presented in section 15.8.

15.1.2.2.2 **Results**

A bounding zero power analysis is performed by modeling only Mode 2 conditions. This approach is valid, since the transient data are not compared with rod withdrawal from subcritical statepoints. The analysis is performed with the LOFTRAN code. The transient data are evaluated to determine that the departure from nucleate boiling design basis is met. It was also confirmed that the hot zero power feedwater malfunction statepoints are bounded by the hot zero power steam line break statepoints. Therefore, the hot zero power feedwater malfunction event is bounded by the zero power case of a steam system piping failure event analyzed in subsection 15.1.5. The limiting transient data of the case of increased feedwater flow to two steam generators was determined to be the overall limiting transient data for both of the cases: (that is, the case for increased feedwater flow to one steam generator and the case for increased feedwater flow to two steam generators).

The results of the full power analysis, presented in table 15.1.2-1 and figures 15.1.2-1, 15.1.2-2, and 15.1.2-3 are based on operation with an analog feedwater control system. These results are similar to the results of the analysis modeling operation with the digital feedwater control system that is now in operaton. The specific results for operation with digital feedwater control system are not presented since the detailed results and conclusions presented for operation with the analog feedwater control system are considered valid for operation with the digital feedwater control system.

The full power case, which assumes maximum reactivity feedback coefficients, results in the greatest power increase. Assuming manual or automatic rod control (insertion and withdrawal) results in a similar power increase. Assuming automatic rod control results in a more severe transient with respect to DNBR.

When the steam generator water level in the faulted loop reaches the P-14 high-high level setpoint, the feedwater regulating valves and feedwater pump discharge valves are automatically closed and the main feedwater pumps trip. This prevents continuous addition of feedwater. The steam generator high-high water level condition will also result in a turbine trip which, in turn, causes a reactor trip. However, for the purpose of this analysis, the reactor trip on turbine trip is not credited. Following feedwater isolation, the water level in the intact steam generators reduces to the low-low level setpoint, and this results in a reactor trip. Minimum DNBR for this transient is reached prior to the reactor trip.

Transient results show the increase in nuclear power associated with the increased thermal load on the reactor. (See figures 15.1.2-1 through 15.1.2-3.) The DNBR does not drop below the limit value. Following the reactor trip, the plant approaches a stabilized condition; standard plant shutdown procedures then apply to further cool down the plant.

Since the power level rises during the excessive feedwater flow incident, the fuel temperature will also rise until after reactor trip occurs. The core heat flux lags behind the neutron flux response due to the fuel rod thermal time constant; hence, the peak heat flux does not exceed 118 percent of its nominal value (i.e., the assumed high neutron flux trip setpoint). Thus, the peak fuel temperature will remain well below the fuel melting temperature.

The transient results show that DNB does not occur at any time during the excessive feedwater flow incident; thus, there is no reduction in the ability of the primary coolant to remove heat from the fuel rods. The fuel cladding temperature, therefore, does not rise significantly above its initial value during the transient.

The transient results also show that the peak pressures of the reactor coolant system and main steam system do not challenge the pressure limits of these systems anytime during the event.

The analysis considers cases with and without automatic rod control. The automatic rod control cases assume that the rods can automatically insert and withdraw. This bounds operation at Vogtle Unit 1 and Unit 2 with the automatic rod withdrawal feature physically disabled.

The calculated sequence of events for the increase in feedwater flow for the full power case is shown in table 15.1.2-1.

15.1.2.3 Conclusions

The results of the analysis show that the DNBRs encountered for an excessive feedwater addition at full power conditions are above the limit value. A bounding hot zero power feedwater malfunction analysis is performed modeling excessive feedwater addition only in Mode 2. The results show that the hot zero power feedwater malfunction analysis is bounded by the hot zero power steam line break analysis (subsection 15.1.5).

15.1.2.4 Reference

1. Burnett, T. W. T., et al., "LOFTRAN Code Description," WCAP-7907-P-A (proprietary), WCAP-7907-A (nonproprietary), April 1984.

15.1.3 EXCESSIVE INCREASE IN SECONDARY STEAM FLOW

15.1.3.1 Identification of Causes and Accident Description

A rapid increase in steam flow that causes a power mismatch between the reactor core power and the steam generator load demand defines an excessive load increase incident. The reactor control system design accommodates a 10-percent step load increase and a 5-percent per minute ramp load increase in the range of 15 to 100 percent of full power. Any loading rate more than these values may cause a reactor trip actuated by the reactor protection system. Subsections 15.1.4 and 15.1.5 discuss steam flow increases greater than 10 percent.

This accident could result from either an administrative violation such as excessive loading by the operator or an equipment malfunction in the steam dump control or turbine speed control.

Reactor coolant condition signals control turbine bypass to the condenser during power operation; i.e., high reactor coolant temperature indicates a need for turbine bypass by using steam dumps. A single controller malfunction does not cause turbine bypass; an interlock blocks the opening of the steam dump valves unless a large turbine load decrease or turbine trip occurs.

The following reactor protection system signals protect against an excessive load increase accident:

- OPΔT
- OTΔT
- Power range high neutron flux.
- Low pressurizer pressure.

This is an ANS Condition II incident.

15.1.3.2 Analysis of Effects and Consequences

15.1.3.2.1 Method of Analysis

The analysis of this accident uses the LOFTRAN code (reference 1). This code simulates the neutron kinetics, RCS, pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generator, steam generator safety valves, and feedwater system. The code computes pertinent plant variables including temperatures, pressures, and power level.

The analysis includes four cases that demonstrate the plant behavior following a 10-percent step load increase from rated load. These cases are as follows:

A. Manual rod control with minimum reactivity feedback.

B. Manual rod control with maximum reactivity feedback.

C. Automatic rod control with minimum reactivity feedback.

D. Automatic rod control with maximum reactivity feedback.

The analysis considers cases with and without automatic rod control. For the minimum moderator feedback cases, the analysis assumes that the core has a zero moderator temperature coefficient of reactivity and the least negative Doppler-only power coefficient curve. This results in the least inherent transient response capability. The zero moderator temperature coefficient of reactivity bounds a positive moderator temperature coefficient for this cooldown event. For the maximum moderator feedback cases, the moderator temperature coefficient of reactivity has its highest absolute value and the most negative Doppler-only power coefficient curve. This results in the largest amount of reactivity feedback due to changes in coolant temperature. For the cases with automatic rod control, no credit was taken for ΔT trips on overtemperature or overpower in order to demonstrate the inherent transient capability of the plant. Under actual operating conditions, such a trip may occur after which the plant would quickly stabilize.

The analysis assumes a 10-percent step increase in steam demand, and the analysis of all the cases does not take credit for pressurizer heaters. The analysis of this accident uses RTDP as described in reference 2. The analysis assumes nominal values for the initial reactor power, pressure, and RCS temperature. The limit DNBR includes uncertainties in initial conditions. Tables 15.0.3-2 and 15.0.3-3 show the plant characteristics and initial conditions.

The analysis does not require normal reactor control systems and engineered safety systems to function. The analysis assumes the reactor protection system to be operable; however, due to the error allowances assumed in the setpoints, a reactor trip does not occur. No single active failure will prevent the reactor protection system from performing its intended function.

15.1.3.2.2 Results

Figures 15.1.3-1 through 15.1.3-4 illustrate the transient with the reactor in the manual control mode. As expected, for the minimum moderator feedback case there is a slight power increase, and the average core temperature shows a large decrease. This results in a DNBR which increases above its initial value. For the maximum moderator feedback manual rod controlled case, there is a large increase in reactor power due to the moderator feedback. A reduction in DNBR occurs, but DNBR remains above the limit value.

The analysis considers cases with and without automatic rod control. The automatic rod control cases assume that the rods can automatically insert and withdraw. This bounds operation at Vogtle Unit 1 and Unit 2 with the automatic rod withdrawal feature physically disabled.

Figures 15.1.3-5 through 15.1.3-8 illustrate the transient assuming the reactor is in the automatic rod control mode (insertion and withdrawal). Both the minimum and maximum moderator feedback cases show that core power increases, thereby reducing the rate of decrease in coolant average temperature and pressurizer pressure. For both of these cases, the minimum DNBR remains above the limit value.

For all cases, the plant rapidly reaches a stabilized condition at the higher power level where a reduction in power can occur by following normal plant operating procedures. Note that due to the measurement errors assumed in the setpoints, it is possible that reactor trip could occur for the automatic control cases. The plant would then reach a stabilized condition following the trip.

The excessive load increase incident is an overpower transient for which the fuel temperatures will rise. Reactor trip does not occur for the cases analyzed, and the plant reaches a new equilibrium condition at a higher power level corresponding to the increase in steam flow.

Since DNB does not occur at any time during the excessive load increase transients, there is no reduction in the ability of the primary coolant to remove heat from the fuel rod. Thus, the fuel cladding temperature does not rise significantly above its initial value during the transient.

The calculated sequence of events for the excessive load increase incident is shown in table 15.1.2-1.

15.1.3.3 Conclusions

The analysis presented above shows that for a 10-percent step load increase, the DNBR remains above the limit value. The plant reaches a stabilized condition following the load increase.

15.1.3.4 References

1. Burnett, T. W. T, et al., "LOFTRAN Code Description," WCAP-7907-P-A (Proprietary), WCAP-7907-A (Nonproprietary), April 1984.
2. Friedland, A. J., and Ray, S., "Revised Thermal Design Procedure," WCAP-11397-P-A, April 1989.

15.1.4 INADVERTENT OPENING OF A STEAM GENERATOR RELIEF OR SAFETY VALVE

The analyses presented in this section are bounded by the main steam line piping failure analyzed in subsection 15.1.5. This section is retained for historical purposes.

15.1.4.1 Identification of Causes and Accident Description

The most severe core conditions resulting from an accidental depressurization of the main steam system are associated with an inadvertent opening of a single steam dump, relief, or safety valve. The analyses performed assuming a rupture of a main steam line are given in subsection 15.1.5.

A piping failure in the vicinity of the steam generator relief valves could be assumed to cause the valves to stick open. The combination of the pipe failure, an assumed single failure, and the stuck open valve(s) may result in inadequate auxiliary feedwater flow. The VEGP main steam atmospheric relief valves and the associated pressure transmitters have been procured as Class 1E devices which are environmentally qualified for the effects of high energy line breaks. Therefore, this scenario does not present a safety problem for the VEGP design.

The steam release, as a consequence of this accident, results in an initial increase in steam flow which decreases during the accident as the steam pressure falls. The energy removal from the reactor coolant system (RCS) causes a reduction of coolant temperature and pressure. In the presence of a negative moderator temperature coefficient, the cooldown results in an insertion of positive reactivity.

The analysis is performed to demonstrate that the following criterion is satisfied:

- Assuming the most reactive stuck rod cluster control assembly (RCCA), with offsite power available, and assuming a single failure in the engineered safety features (ESF) system, there will be no consequential damage to the fuel or RCS after reactor trip for a steam release equivalent to the spurious opening, with failure to close, of the largest of any single steam dump, relief, or safety valve. This criterion is met by showing the departure from nucleate boiling (DNB) design basis is not exceeded.

Accidental depressurization of the secondary system is classified as an American Nuclear Society Condition II event.

The following systems provide the necessary protection against an accidental depressurization of the main steam system.

A. Safety injection (SI) system actuation from any of the following:
   1. Two out of four low pressurizer pressure signals.
   2. Two out of three low steam line pressure signals in any one loop.

B. The overpower reactor trips (neutron flux and ΔT) and the reactor trip occurring in conjunction with receipt of the SI signal.

C. Redundant isolation of the main feedwater lines.

Sustained high feedwater flow would cause additional cooldown. Therefore, in addition to the normal control action which will close the main feedwater valves following reactor trip, an SI signal will rapidly close all feedwater regulating valves and back up feedwater isolation valves, trip the main feedwater pumps, and close the feedwater pump discharge valves.

D. Trip of the fast-acting main steam line isolation valves (designed to close in less than or equal to 7 s for the events analyzed) on:

1. SI system actuation derived from two out of three low steam line pressure signals in any one loop (above permissive P-11).

2. Two out of three high negative steam pressure rates in any loop (below permissive P-11).

Plant systems and equipment which are available to mitigate the effects of the accident are also discussed in subsection 15.0.8 and listed in table 15.0.8-1.

15.1.4.2 Analysis of Effects and Consequences

15.1.4.2.1 Method of Analysis

The following analyses of a secondary system steam release are performed for this section.

A. A full plant digital computer simulation using the LOFTRAN code[(1)] to determine RCS temperature and pressure during cooldown, and the effect of SI.

B. Analyses to determine that there is no damage to the fuel or RCS.

The following conditions are assumed to exist at the time of a secondary steam system release:

A. End-of-life shutdown margin at no-load, equilibrium xenon conditions, and with the most reactive RCCA stuck in its fully withdrawn position. Operation of RCCA banks during core burnup is restricted by the insertion limits so that addition of positive reactivity induced by a secondary system steam release accident will not lead to a more adverse condition than the case analyzed.

B. A negative moderator coefficient corresponding to the end-of-life rodded core with the most reactive RCCA in the fully withdrawn position. The variation of the coefficient with temperature and pressure is included. The k_{eff} versus temperature at 1150 psi corresponding to the negative moderator temperature coefficient used is shown in figure 15.1.4-1.

C. Minimum capability for injection of boric acid solution corresponding to the most restrictive single failure in the SI system. This corresponds to the flow delivered by one centrifugal charging pump delivering its full contents to the cold leg header. Reactor coolant seal injection flow is not included in the total core delivery. No credit has been taken for the low concentration boric that acid must be swept from the SI lines downstream of the refueling water storage tank isolation valves prior to the delivery of boric acid (2400 ppm) to the reactor coolant loops.

D. The case studied is a steam flow of 268 lb/s at 1200 psia with offsite power available. This is the maximum capacity of any single steam dump, relief, or safety valve. Initial hot shutdown conditions at time zero are assumed since this represents the most conservative initial condition.

Should the reactor be just critical or operating at power at the time of a steam release, the reactor will be tripped by the normal overpower protection when power level reaches a trip point. Following a trip at power, the RCS contains more stored energy than at no-load; the average coolant temperature is higher than at no-load; and there is appreciable energy stored in the fuel. Thus, the additional stored energy is removed via the cooldown caused by the steam release before the no-load conditions of RCS temperature and shutdown margin assumed in the analyses are reached. After the additional stored energy has been removed, the cooldown and reactivity insertions proceed in the same manner as in the analysis which assumes no-load condition at time zero. However, since the initial steam generator water inventory is greatest at no-load, the magnitude and duration of the RCS cooldown are less for steam line release occurring at power.

E. In computing the steam flow, the Moody Curve[(2)] for f(L/D) = 0 is used.

F. Perfect moisture separation in the steam generator is assumed.

G. Offsite power is assumed, since this would maximize the cooldown.

H. Maximum cold auxiliary feedwater flow is assumed.

I. Four reactor coolant pumps are operating.

15.1.4.2.2 Results

The results presented are a conservative indication of the events which would occur assuming a secondary system steam release since it is postulated that all of the conditions described above occur simultaneously.

Figures 15.1.4-2 through 15.1.4-5 show the transient results for a steam line break of 0.11 ft^2.

The assumed steam release is that which would occur from the opening of a steam generator safety valve. Safety injection is initiated automatically by low pressurizer pressure. Operation of one centrifugal charging pump is assumed. Boron solution at 2400 ppm enters the RCS providing sufficient negative reactivity to prevent core damage. The transient is quite conservative with respect to cooldown, since no credit is taken for the energy stored in the system metal other than that of the fuel elements and steam generator tubes. Since the transient occurs over a period of several minutes, the neglected stored energy is likely to have a significant effect in slowing the cooldown. The calculated time sequence of events for this accident is listed in table 15.1.2-1.

15.1.4.3 Conclusions

The analysis shows that the criterion stated earlier in this section is satisfied. For an inadvertant opening of a steam generator relief or safety valve, the DNB design basis is met.

15.1.4.4 References

1. Burnett, T. W. T., et al., "LOFTRAN Code Description," WCAP-7907A, April 1, 1984.
2. Moody, F. S., "Transactions of the ASME," Journal Of Heat Transfer, Figure 3, page 134, February 1965.

15.1.5 STEAM SYSTEM PIPING FAILURE

The analyses of the rupture of a main steam line at power presented in this section are bounded by the zero power case and are retained for historical purposes.

15.1.5.1 Identification of Causes and Accident Description

The steam release arising from a rupture of a main steam line will result in an initial increase in steam flow that decreases during the accident as the steam pressure falls. The energy removal from the reactor coolant system (RCS) causes a reduction of coolant temperature and pressure. In the presence of a negative moderator temperature coefficient, the cooldown results in an insertion of positive reactivity. If the most reactive rod cluster control assembly (RCCA) is assumed stuck in its fully withdrawn position after reactor trip, there is an increased possibility that the core will become critical and return to power. A return to power following a steam line rupture is a potential problem mainly because of the high-power peaking factors which exist, assuming the most reactive RCCA to be stuck in its fully withdrawn position. The core is ultimately shut down by the boric acid solution delivered by the safety injection (SI) system and accumulators.

A failure of secondary system piping inside the containment may cause pressurizer power-operated relief valves (PORV) to open. The resultant secondary break coincident with PORV opening may have more severe consequences than those accidents previously analyzed. The VEGP pressurizer PORV and associated pressure transmitters meet Class 1E requirements and are qualified to the postulated accident environments inside the containment. Therefore, this scenario does not present a safety problem for the VEGP design.

The analysis of a main steam line rupture is performed to demonstrate that the following criterion is satisfied:

- Assuming the most reactive stuck RCCA with or without offsite power and assuming a single failure in the engineered safety features (ESF), there is no consequential damage to the primary system and the core must remain in place and intact. Radiation doses do not exceed the guidelines of 10 CFR 100.

Although departure from nucleate boiling (DNB) and possible clad perforation following a steam pipe rupture are not necessarily unacceptable, the following analysis, in fact, shows that the DNB design basis is not exceeded for any rupture, assuming the most reactive assembly stuck in its fully withdrawn position. The departure from nucleate boiling ratio (DNBR) design basis is discussed in section 4.4.

A major steam line rupture is classified as an American Nuclear Society Condition IV event.

Effects of minor secondary system pipe breaks are bounded by the analysis presented in this section. Minor secondary system pipe breaks are classified as Condition III events, as described in paragraph 15.0.1.3.

The rupture of a main steam line is the most limiting cooldown transient and is analyzed at zero power with no decay heat. Decay heat would retard the cooldown, thereby reducing the likelihood that the reactor will return to power. A detailed analysis of this transient with the most limiting break size, a double-ended rupture, is presented here. The assumptions used in this analysis are discussed in reference 1. Reference 1 also contains a discussion of the spectrum of break sizes and power levels analyzed.

An analysis of the steam line break event initiated from full power is also performed.

The following functions provide the protection for a steam line rupture: (See paragraph 7.2.1.1.2.)

A. SI system actuation from any of the following:

   1. Two out of four low pressurizer pressure signals.

   2. Two out of three high-1 containment pressure signals.

   3. Two out of three low steam line pressure signals in any loop.

B. The overpower reactor trips (neutron flux and ΔT) and the reactor trip occurring in conjunction with receipt of the SI signal.

C. Redundant isolation of the main feedwater lines.

   Sustained high feedwater flow would cause additional cooldown. Therefore, in addition to the normal control action which will close the main feedwater valves, an SI signal will rapidly close all feedwater regulating valves and back up feedwater isolation valves, trip the main feedwater pumps, and close the feedwater pump discharge valves.

D. Trip of the fast-acting main steam line isolation valves (MSIVs) (designed to close in less than or equal to 7 s for the events analyzed) on:

   1. Two out of three high-2 containment pressure.

   2. SI system actuation derived from two out of the three low steam line pressure signals in any one loop (above permissive-11).

   3. Two out of three high negative steam pressure rates in any one loop (below permissive-11).

Two fast-acting MSIVs are provided in each steam line; these valves fully close within 7 s (for the events analyzed) of actuation following a large break in the steam line. An additional delay of 3 s is included for sensor and protection system delays. For breaks downstream of the MSIVs, closure of at least one valve in each line will completely terminate the blowdown. For any break in any location, no more than one steam generator would experience an uncontrolled blowdown even if one of the MSIVs fails to close. A description of steam line isolation is included in chapter 10.

Flow restrictors are installed in the steam generator outlet nozzle, an integral part of the steam generator. The effective throat area of the nozzles is 1.4 ft^2, which is considerably less than the main steam pipe area; thus, the nozzles also serve to limit the maximum steam flow for a break at any location.

Table 15.1.5-1 lists the equipment required in the recovery from a high-energy line rupture. Not all equipment is required for any one particular break since the requirements will vary depending upon postulated break size and location. Design criteria and methods of protection of safety-related equipment from the dynamic effects of postulated piping ruptures are provided in section 3.6.

15.1.5.2 Analysis of Effects and Consequences

15.1.5.2.1 Method of Analysis

The analysis of the steam pipe rupture has been performed to determine:

A. The core heat flux and RCS temperature and pressure resulting from the cooldown following the steam line break. The LOFTRAN code[2] has been used.

B. The thermal and hydraulic behavior of the core following a steam line break. A detailed thermal and hydraulic digital computer code, VIPRE-01 has been used to determine if DNB occurs for the core conditions computed in item A above.

The following conditions were assumed to exist at the time of a main steam line break accident initiated from zero power:

A. End-of-life shutdown margin at no-load, equilibrium xenon conditions, and the most reactive RCCA stuck in its fully withdrawn position. Operation of the control rod banks during core burnup is restricted in such a way that addition of positive reactivity in a steam line break accident will not lead to a more adverse condition than the case analyzed.

B. A negative moderator coefficient corresponding to the end-of-life rodded core with the most reactive RCCA in the fully withdrawn position. The variation of the coefficient with temperature and pressure has been included. The k_{eff} versus temperature at 1150 psia corresponding to the negative moderator temperature coefficient used is shown in figure 15.1.5-0.

   The core properties associated with the sector nearest the affected steam generator and those associated with the remaining sectors were conservatively combined to obtain average core properties for reactivity feedback calculations. Further, it was conservatively assumed that the core power distribution was uniform. These two conditions cause underprediction of the reactivity feedback in the high-power region near the stuck rod. The stuck rod is assumed in the region of the core of lowest temperature.

   To verify the conservatism of this method, the reactivity as well as the power distribution was checked for the limiting conditions of the cases analyzed. This core analysis considered the Doppler reactivity from the high-fuel temperature near the stuck RCCA, moderator feedback from the high-water enthalpy near the stuck RCCA, power redistribution, and nonuniform core inlet temperature effects. For cases in which steam generation occurs in the high-flux regions of the core, the effect of void formation was also included. It was confirmed that the reactivity feedback model employed in the kinetics analysis was consistent with the core analysis and the overall analysis is conservative.

C. Minimum capability for injection of boric acid (2400 ppm) solution corresponding to the most restrictive single failure in the SI system. Minimum boron concentration of injection fluid of 1900 ppm is assumed for the accumulators. The emergency core cooling system (ECCS) consists of three systems:

   - The passive accumulators.

   - The residual heat removal (RHR) system.

- The SI system.

The SI system and 3 of the 4 passive accumulators are modeled for the steam line break accident analysis.

The actual modeling of the SI system in LOFTRAN is described in reference 2. The flow corresponds to that delivered by one charging pump delivering its full flow to the cold leg header. No credit has been taken for the 2400 ppm borated water, that must be swept from the lines downstream of the refueling water storage tank isolation valves prior to the delivery of boric acid to the reactor coolant loops.

For the cases where offsite power is assumed, the sequence of events in the SI system is as follows. After the generation of the SI signal (appropriate delays for instrumentation, logic, and signal transport included), the appropriate valves begin to operate and the high-head SI pump starts. In 27.8 s the valves are assumed to be in their final position, and the pump is assumed to be at full speed. The volume containing the assumed unborated water is swept before the 2400 ppm boric acid (from the refueling water storage tank) reaches the core. This delay, described above, is inherently included in the modeling.

In cases where offsite power is not available, an additional 15-s delay is assumed to start the diesels and to load the necessary SI equipment onto them. A total delay of 42 s is assumed.

D. Design value of the steam generator heat transfer coefficient.

E. Since the steam generators are provided with integral flow restrictors with a 1.4-ft^2 throat area, any rupture with a break area greater than 1.4 ft^2, regardless of location, would have the same effect on the nuclear steam supply system as the 1.4-ft^2 break. The following cases have been considered in determining the core power and RCS transients:

1. Complete severance of a pipe, with the plant initially at no-load conditions, full reactor coolant flow with offsite power available.

2. Case 1 with loss of offsite power. Loss of offsite power results in reactor coolant pump coastdown.

F. Power peaking factors corresponding to one stuck RCCA and nonuniform core inlet coolant temperatures are determined at end of core life. The coldest core inlet temperatures are assumed to occur in the sector with the stuck rod. The power peaking factors account for the effect of the local void in the region of the stuck control assembly during the return to power phase following the steam line break. This void in conjunction with the large negative moderator coefficient partially offsets the effect of the stuck assembly. The power peaking factors depend upon the core power, temperature, pressure, and flow.

Both cases above assume initial hot standby conditions at time zero since this represents the most pessimistic initial condition. If the reactor is just critical or operating at power at the time of a steam line break, the reactor will be tripped by the normal overpower protection system when power level reaches a trip setpoint. Following a trip at power, the RCS contains more stored energy than at no-load; the average coolant temperature is higher than at no-load, and there is appreciable energy stored in the fuel. Thus, the additional stored energy is removed via the cooldown caused by the steam line break before the no-load conditions of RCS temperature and shutdown margin assumed in the analyses are reached. After the additional stored energy has been removed, the cooldown and reactivity insertions proceed in the same manner as in the analysis which assumes no-load condition at time zero.

G. In computing the steam flow during a steam line break, the Moody Curve[3] for f(L/D) = 0 is used.

The following conditions were assumed to exist at the time of the steam line break event initiated from power.

A. The DNB analysis is performed using the revised thermal design procedure. Initial core power, and reactor coolant temperature and pressure are assumed to be at their nominal values consistent with steady-state full power operation. Uncertainties in initial conditions are included in the DNBR limit as described in WCAP-11397 (References 5). Note that only the random uncertainties are accounted for in the RTDP methodology and that a temperature bias of 1° and pressure bias of 13 psi are included in the initial conditions.

B. A conservatively large positive moderator density coefficient of 0.5 $\Delta k/g/cm^3$ (corresponding to a large negative moderator temperature coefficient) and a most-negative Doppler-only power coefficient form the basis for the end-of-life (EOL) maximum reactivity feedback assumption.

C. A range of break sizes was analyzed. The larger break sizes caused an early trip via a low steam line pressure signal. For the smaller break sizes, the reactor protection occurs via an OPΔT signal.

D. The RCCA trip insertion characteristic is based on the assumption that the highest-worth rod cluster control assembly is stuck in its fully withdrawn position.

E. Part-power cases were analyzed to determine the most limiting transient conditions.

F. The impact of a full power reactor coolant system T_{avg} window was considered for the steamline break at power analysis. A conservative calculation modeling the high end of the T_{avg} window was explicitly analyzed since this is limiting with respect to the event criteria.

15.1.5.2.2 Results

The calculated sequence of events for both cases analyzed from zero power is shown on table 15.1.2-1.

The time sequence of events for the limiting at-power case is shown in table 15.1.2-1.

The results presented are a conservative indication of the events which would occur assuming a steam line rupture, since it is postulated that all of the conditions described above occur simultaneously.

15.1.5.2.3 Core Power and Reactor Coolant System Transient

Figures 15.1.5-1 through 15.1.5-4 show the RCS transient and core heat flux following a main steam line rupture (complete severance of a pipe) at initial no-load condition (case 1).

Offsite power is assumed available so that full reactor coolant flow exists. The transient shown assumes an uncontrolled steam release from only one steam generator. Should the core be critical at near zero power when the rupture occurs, the initiation of SI by low steam line pressure will trip the reactor. Steam release from more than one steam generator will be prevented by automatic trip of the fast-acting MSIVs in the steam lines by high containment pressure signals or by low steam line pressure signals. Even with the failure of one valve, release is limited to no more than 10 s for the other steam generators while the one generator blows down. For the events analyzed, the MSIVs are designed to be fully closed in less than or equal to 7 s from receipt of a closure signal.

As shown in figures 15.1.5-4 and 15.1.5-4, the core attains criticality with the RCCAs inserted (with the design shutdown assuming one stuck RCCA) before boron solution at 2400 ppm (in the RWST) and 1900 ppm (in the accumulators) enters the RCS. A peak core power significantly lower than the nominal full-power value is attained.

The calculation assumes the boric acid is mixed with and diluted by the water flowing in the RCS prior to entering the reactor core. The concentration after mixing depends upon the relative flowrates in the RCS and in the SI system. The variation of mass flowrate in the RCS due to water density changes is included in the calculation as is the variation of flowrate in the SI system due to changes in the RCS pressure. The SI system flow calculation includes the line losses in the system as well as the pump head curve.

Figure 15.1.5-5 shows the transients for case 2, which corresponds to the case discussed above with additional loss of offsite power at the time the SI signal is generated. The SI system delay time includes 15 s to start the diesel in addition to 27 s to start the SI pump and open the valves. Criticality is achieved later, and the core power increase is slower than in the similar case with offsite power available. The ability of the emptying steam generator to extract heat from the RCS is reduced by the decreased flow in the RCS. The peak power remains well below the nominal full-power value.

Figures 15.1.5-9 through 15.1.5-11 illustrate the RCS transient for a steam line break occurring with the reactor at power. Following the break, the reactor coolant average temperature decreases, resulting in an increase in the core power due to the negative moderator temperature coefficient. The core power increases until a reactor trip occurs.

The limiting case for demonstrating DNB design basis is the largest break size that results in a trip on overpower ΔT. For larger break sizes, a reactor trip is generated within a few seconds of the break on the lead-lag compensated low steam pressure safety injection actuation signal.

It should be noted that following a steam line break, only one steam generator blows down completely. Thus, the remaining steam generators are still available for dissipation of decay heat after the initial transient is over. In the case of loss of offsite power, this heat is removed to the atmosphere via the steam line safety and relief valves.

15.1.5.2.4 Margin to Critical Heat Flux

A DNB analysis was performed for both of these cases. It was found that both cases had a minimum DNBR greater than the design limit.

15.1.5.3 Radiological Consequences

The evaluation of the radiological consequences of a postulated main steam line break (MSLB) outside containment assumes that the reactor has been operating with a small percent of defective fuel and leaking steam generator tubes for sufficient time to establish equilibrium concentrations of radionuclides in the reactor coolant and in the secondary coolant.

Following the rupture, auxiliary feedwater to the faulted loop is isolated and the steam generator is allowed to steam dry. Hence, radioiodines carried from the primary coolant to the generator via leaking tubes are assumed to be released directly to the environment. Iodines released from the generators in the intact loops via the steam line safety or power-operated relief valves are assumed to be mixed with the secondary coolant and partitioned between the generator liquid and steam before release to the environment.

15.1.5.3.1 Analytical Assumptions

The major assumptions and parameters used in the analysis are itemized in table 15.1.5-2.

15.1.5.3.1.1 Source Term Calculations. The concentration of nuclides in the primary and secondary system, prior to the accident are determined as follows:

A. The iodine concentrations in the reactor coolant will be based upon preaccident and accident initiated iodine spikes.[4]

   1. Accident Initiated Spike

      The reactor trip associated with the MSLB creates an iodine spike in the primary system which increases the iodine release rate from the fuel to the primary coolant to a value 500 times greater than the release rate corresponding to the maximum equilibrium primary system iodine concentration of 1 µCi/g of dose equivalent (DE) I-131.

   2. Preaccident Spike

      A reactor transient has occurred prior to the MSLB and has raised the primary coolant iodine concentration to 60 µCi/g of DE I-131.

B. The noble gas concentrations in the primary coolant are based on 1-percent defective fuel.

C. The secondary coolant activity is based on the DE of 0.1 µCi/g of I-131.

15.1.5.3.1.2 Mathematical Models Used in the Analysis. Mathematical models used in the analysis are described in the following sections:

A. The mathematical models used to analyze the activity released during the course of the accident are described in appendix 15A.

B. The atmospheric dispersion factors used in the analysis were calculated based on the onsite meteorological measurement programs described in subsection 2.3.3.

C. The thyroid inhalation dose and total body gamma immersion doses to a receptor at the exclusion area boundary and outer boundary of the low population zone were analyzed using the models described in appendix 15A.

15.1.5.3.1.3 Identification of Leakage Pathways and Resultant Leakage Activity. For evaluating the radiological consequences due to a postulated MSLB, the activity released from the affected steam generator (steam generator connected to the broken steam line) is released directly to the environment. The unaffected steam generators are assumed to continually discharge steam and entrained activity via the safety and relief valves up to the time initiation of the RHR system can be accomplished.

All activity is released to the environment with no consideration given to radioactive decay or cloud depletion by ground deposition during transport to the exclusion area boundary and low population zone. Hence, the resultant radiological consequences represent the most conservative estimate of the potential integrated dose due to the postulated MSLB.

15.1.5.3.2 Identification of Uncertainties and Conservative Elements in the Analysis

A. Reactor coolant activities are based on the technical specification limit of 1.0-µCi/g I-131 DE with extremely large iodine spike values, resulting in equivalent concentrations many times greater than the reactor coolant activities based on 0.12-percent defective fuel associated with normal operating conditions.

B. The noble gas activities are based on 1-percent defective fuel which cannot exist simultaneously with 1.0-µCi/g I-131. For iodines, 1-percent defects would be approximately three times the technical specification limit.

C. A 1-gal/min steam generator primary-to-secondary leakage is assumed, which is significantly greater than that anticipated during normal operation. Furthermore, it was conservatively assumed that 0.35-gal/min leakage goes to the affected steam generator.

D. The meteorological conditions which may be present at the site during the course of the accident are uncertain. However, it is highly unlikely that the assumed meteorological conditions would be present during the course of the accident for any extended period of time. Therefore, the radiological consequences evaluated, based on the meteorological conditions assumed, are conservative.

15.1.5.3.3 Conclusions

15.1.5.3.3.1 Filter Loadings. The only ESF filtration system considered in the analysis which limits the consequences of the MSLB is the control room filtration system.

Integrated activity on the control room filters has been evaluated for the more limiting loss-of-coolant accident (LOCA) analysis, as discussed in paragraph 15.6.5.4.6. Since the control room filters are capable of accommodating the potential design basis LOCA fission product iodine loadings, there will be sufficient capacity to accommodate any fission product loading due to a postulated MSLB.

15.1.5.3.3.2 Dose to Receptor at the Exclusion Area Boundary and Low Population Zone Outer Boundary. The potential radiological consequences resulting from the occurrence of a postulated MSLB have been conservatively analyzed using assumptions and models described. The total-body gamma dose due to immersion from direct radiation and the thyroid dose due to inhalation have been analyzed for the 0- to 2-h dose at the exclusion area boundary and for the duration of the accident (0 to 20 h) at the low population zone outer boundary. The results are listed in table 15.1.5-3. The resultant doses are well within the guideline values of 10 CFR 100.

15.1.5.4 References

1. Wood, D. C. and Hollingsworth, S. D., "Reactor Core Response to Excessive Secondary Steam Releases," WCAP-9226-P-A , Revision 1, February 1998.
2. Burnett, T. W. T., et al., "LOFTRAN Code Description," WCAP-7907-P-A, (Proprietary), WCAP-7907-A (Non-proprietary), April 1984.
3. Moody, F. S., "Transactions of the ASME," Journal of Heat Transfer, figure 3, page 134, February 1965.
4. "Radiological Consequences of Main Steam Line Failures Outside Containment of a PWR," USNRC Standard Review Plan 15.1.5, Appendix A, Revision 2, July 1981.
5. Friedland, A.J. and Ray, S., "Revised Thermal Design Procedure," WCAP-11397-P-A, (Proprietary), April 1989.

TABLE 15.1.2-1 (SHEET 1 OF 3)

TIME SEQUENCE OF EVENTS FOR INCIDENTS THAT RESULT IN AN INCREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM

Accident	Event	Time (s)
Feedwater system malfunctions that result in an increase in feedwater flow	One main feedwater control valve fails fully open	0.0
	High-high steam generator water level signal generated	84.9
	Feedwater isolation valves close automatically	91.9
	Minimum DNBR occurs	92.0
	Low-low steam generator water level signal generated	118.7
	Reactor trip occurs	120.7
Excessive increase in secondary steam flow		
1. Manual rod control (minimum moderator feedback)	10-percent step load increase	0.0
	Equilibrium conditions reached (approximate time only)	150
2. Manual rod control (maximum moderator feedback)	10-percent step load increase	0.0
	Equilibrium conditions	50
3. Automatic rod control (minimum moderator feedback)	10-percent step load increase	0.0
	Equilibrium conditions	125

TABLE 15.1.2-1 (SHEET 2 OF 3)

Accident	Event	Time (s)
4. Automatic rod control (maximum moderator feedwater)	10-percent step load increase	0.0
	Equilibrium conditions reached (approximate time only)	50
Inadvertent opening of a steam generator relief or safety valve	Inadvertent opening of one main steam safety or relief valve	0.0*
	Pressurizer empty	269.2*
	Safety injection (SI) actuation	298.8*
	Boron reaches core	359.7*
	*Historical	
Steam system piping failure		
1. Case 1: Reactor at hot zero power with offsite power available. All control rods inserted except most reactive RCCA. Shutdown margin = 1.3% $\Delta\rho$	Double-ended guillotine break occurs	0.0
	Low Steam Pressure SIS actuation setpoint reached	0.7
	Main Feedwater flow isolated 7 s after SIS actuation signal	7.7
	MSIVs closed 10 s after SIS actuation signal	10.7
	Reactor becomes critical	20.8
	High-head SI pump at rated speed 27.8 s after SI actuation signal	28.5
	Power reaches maximum level	101.8
	Reactor goes subcritical	161.2

TABLE 15.1.2-1 (SHEET 3 OF 3)

Accident	Event	Time (s)
2. Case 2: Reactor at hot zero power with offsite power unavailable. All control rods inserted except most reactive RCCA. Shutdown margin = 1.3% $\Delta\rho$	Double-ended guillotine break occurs	0.0
	Low Steam Pressure SIS actuation setpoint reached	0.7
	Main Feedwater flow isolated 7 s after SIS actuation signal	7.7
	MSIVs closed 10 s after actuation signal	10.7
	Reactor becomes critical	24.0
	High-head SI pump at rated speed 42 s after SI actuation signal	42.7
	Power reaches maximum level	253.0
	Reactor goes subcritical	281.5
3. Steam Line Break at Power	Steam line breaks	0.0*
	Overpower ΔT reactor trip setpoint reached	23.7*
	Rods begin to drop	31.7*
	Peak core heat flux occurs (minimum DNBR)	32.2*

*Historical

TABLE 15.1.5-1 (SHEET 1 OF 2)

EQUIPMENT REQUIRED FOLLOWING A RUPTURE OF A MAIN STEAM LINE

Short Term (Required for Mitigation of Accident)	Hot Standby	Required for Cooldown
Reactor trip and safeguard actuation channels, including sensors, circuitry, and processing equipment (The protection circuits used to trip the reactor on undervoltage, underfrequency, and turbine trip may be excluded.)	Auxiliary feedwater system, including pumps, water supply steam generator atmospheric relief valves, and system valves and piping (This system must be placed in service to supply water to operable steam generators no later than 10 min after the incident.)	Steam generator power-operated relief valves
		Control for defeating automatic SI actuation during a cooldown and depressurization (i.e., SI signal is reset)
SI system, including pumps, refueling water storage tank, and system valves and piping	Containment air coolers	RHR system, including pumps, heat exchanger, and system valves and piping necessary to cool and maintain the RCS in a cold shutdown condition
	Capability for obtaining reactor coolant system sample	
Containment spray system	Capability for boration to required hot standby concentration	Capability to depressurize the RCS to allow RHR system operation
Diesel generators and emergency power distribution equipment		
Nuclear service cooling water system		
Containment air coolers		
Auxiliary feedwater system, including pumps, water supplies, piping, and valves		
Main feedwater regulating valves (trip closed feature)		
Bypass feedwater regulating valves (trip closed feature)		
Primary and secondary safety valves		Capability to borate to cold shutdown concentration
Associated pump room coolers		

TABLE 15.1.5-1 (SHEET 2 OF 2)

Short Term (Required for Mitigation of Accident)	Hot Standby	Required for Cooldown
Circuits and/or equipment required to trip the main feedwater pumps		
Main feedwater isolation valves (trip closed feature)		
Main steam line isolation valves (trip closed feature)		
Steam generator Blowdown isolation valves (automatic closure feature)		
Batteries (Class 1E)		
Control room ventilation		
Control room equipment must not be damaged to an extent where any equipment will be spuriously actuated or any of the equipment contained elsewhere in this list cannot be operated.		
Emergency lighting		
Post-accident monitoring system[a]		

a. See section 7.5 for a discussion of the post-accident monitoring system.

TABLE 15.1.5-2 (SHEET 1 OF 3)

PARAMETERS USED IN EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A MAIN STEAM LINE BREAK

I. Source Data

A.	Core power level (MWt)	3636
B.	Total steam generator tube leakage (gal/min)	1
C.	Reactor coolant iodine activity	
	1. Accident initiated spike	Initial activity equal to the DE 1.0 μCi/g of I-131 with an assumed iodine spike that increases the rate of iodine release into the reactor coolant by a factor of 500. See table 15A-7.
	2. Preaccident spike	An assumed preaccident iodine spike which has resulted in the DE of 60 μCi/g of I-131 in the reactor coolant. See table 15A-6.
D.	Reactor coolant noble gas activity (both cases)	Based on 1-percent defective fuel. See table 11.1-2.
E.	Secondary system initial activity	DE of 0.1 μCi/g of I-131.
F.	Reactor coolant mass (g)	2.53×10^8
G.	Secondary coolant mass, 4 generators (g)	1.9×10^8
H.	Offsite power	Lost after trip
I.	Primary-to-secondary leakage duration (h)	20
J.	Species of iodine	100 percent elemental

TABLE 15.1.5-2 (SHEET 2 OF 3)

II.	Atmospheric Dispersion Factors	See table 15A-2.
III.	Activity Release Data for the Steam Generator in the Faulted Loop	
	A. Primary-to-secondary leakrate (gal/min)[(a)]	0.35
	B. Steam released (lb)	
	0 to 0.5 h	167,000
	0.5 to 8.0 h	945
	C. Iodine Partition Factor	1
IV.	Activity Release Data for the Steam Generators in the Intact Loops	
	A. Primary-to-secondary leakrate (gal/min)[(a)]	0.65
	B. Steam released (lb)	
	0 to 2 h	424,000
	2 to 8 h	960,000
	8 to 20 h	1,920,000
	C. Iodine partition factor	0.01
V.	Activity Released to the Environment	
	A. Accident initiated spike	

Isotope	0 to 2 h (Ci)	2 to 20 h (Ci)
I-131	12.1	8.41×10^2
I-132	25.3	7.11×10^2
I-133	25.9	1.60×10^2
I-134	9.3	1.50×10^2
I-135	16.0	7.52×10^2

TABLE 15.1.5-2 (SHEET 3 OF 3)

B. Preaccident spike

Isotope	0 to 2 h (Ci)	0 to 20 h (Ci)
I-131	10.8	67.2
I-132	9.1	6.8
I-133	20.1	93.6
I-134	1.7	0.2
I-135	9.4	23.4

C. Noble gases (both cases)

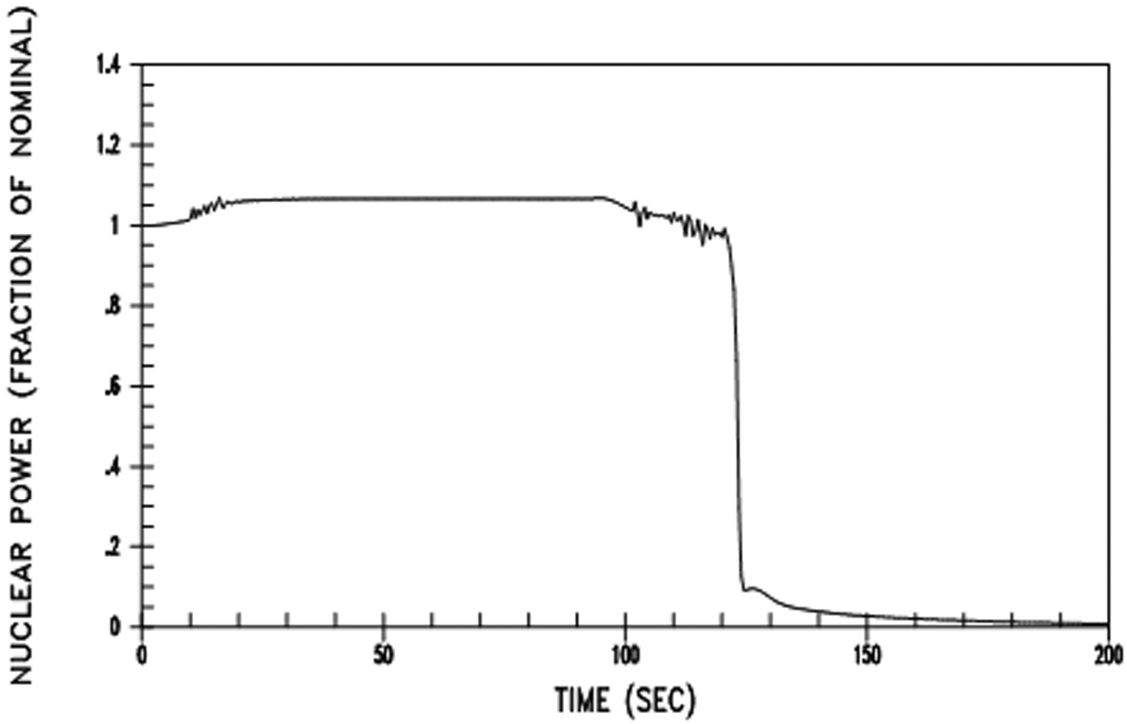
Isotope	0 to 2 h (Ci)	0 to 20 h (Ci)
Xe-131	0.6	8.3
Xe-133m	5.1	73.8
Xe-133	74.6	1.2×10^3
Xe-135m	3.9	5.5×10^2
Xe-135	4.0	1.0×10^3
Xe-138	0.03	1.2×10^{-4}
Kr-85m	0.5	1.9
Kr-85	2.5	32.9
Kr-87	0.2	0.2
Kr-88	0.8	1.9

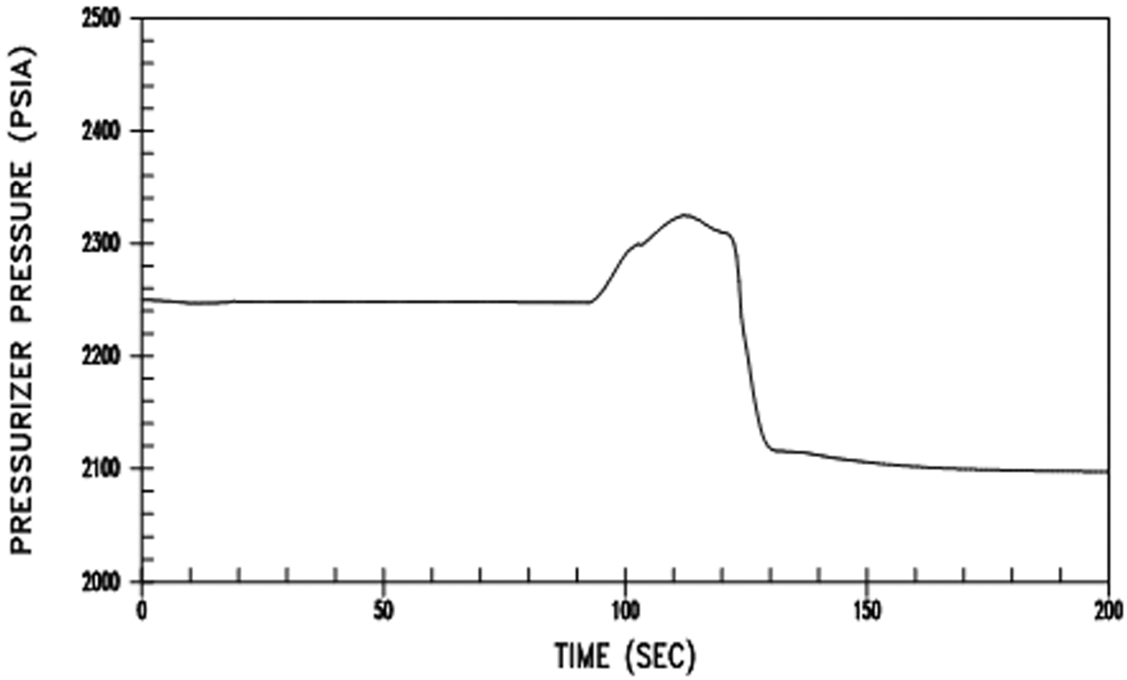
a. Based on water at 590°F, 2250 psia.

TABLE 15.1.5-3

RADIOLOGICAL CONSEQUENCES OF A
MAIN STEAM LINE BREAK

		Doses (rem)
Case 1	Accident Initiated Iodine Spike	
	Exclusion area boundary (0 to 2 h) Thyroid	1.1
	Low population zone outer boundary (8 h) Thyroid	2.2
Case 2	Preaccident Iodine Spike	
	Exclusion area boundary (0 to 2 h) Thyroid	1.0
	Low population zone outer boundary (8 h) Thyroid	1.5
Both Cases	Whole-Body Gamma	
	Exclusion area boundary (0 to 2 h)	< 0.1
	Low population zone outer boundary (8 h)	< 0.1





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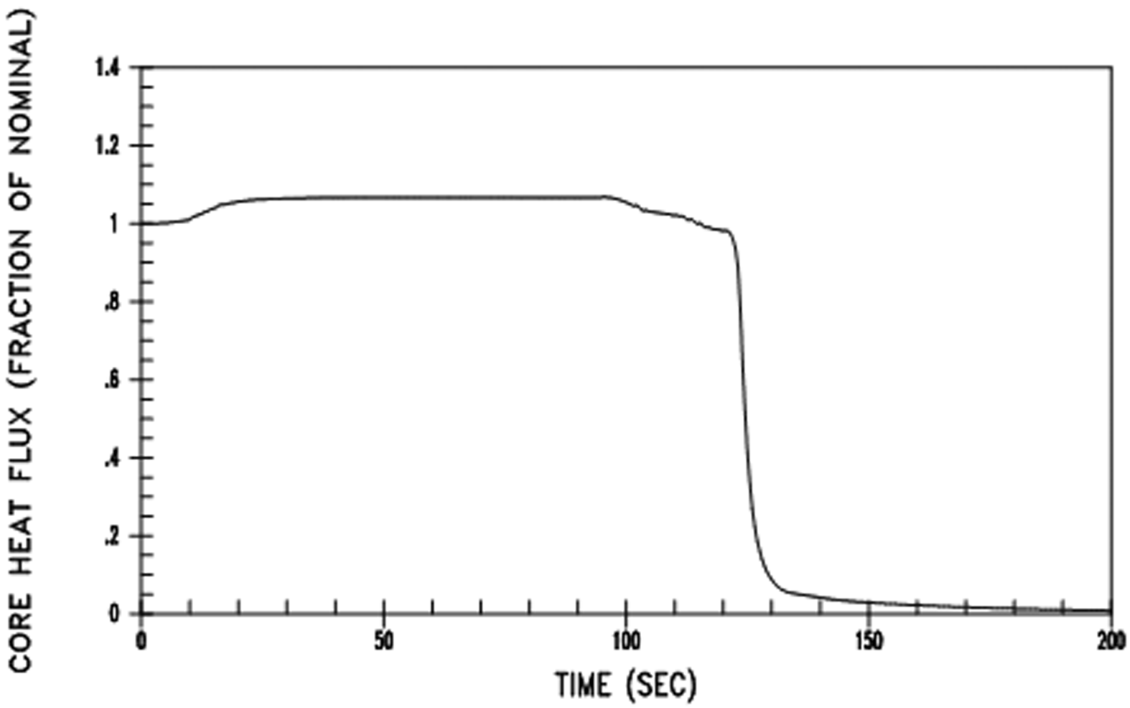
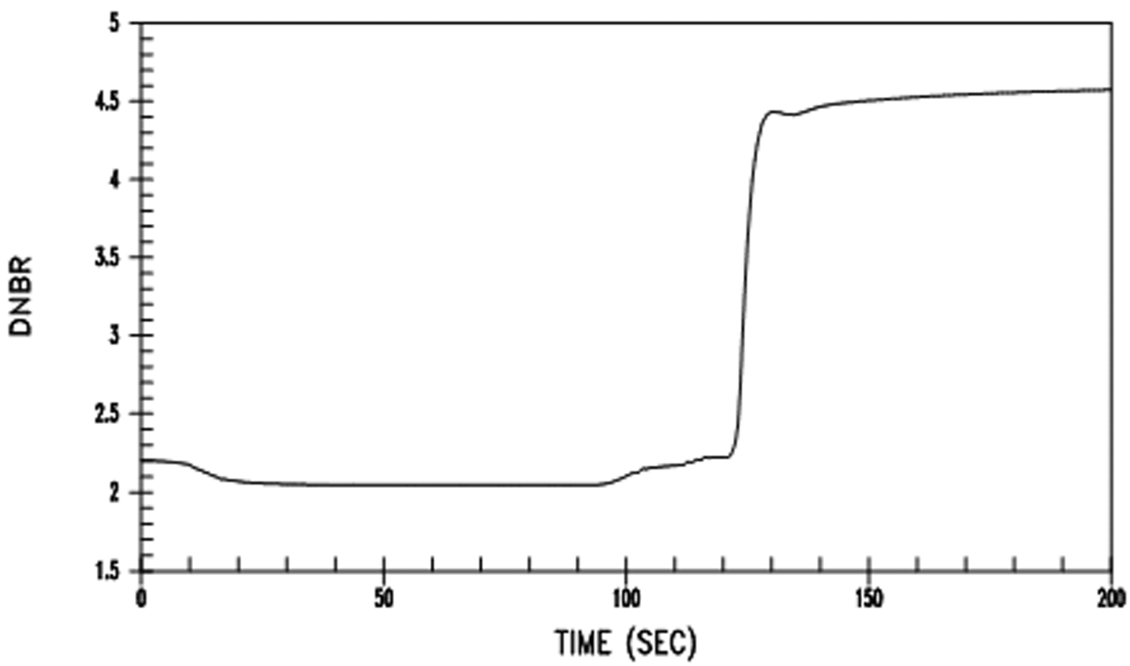
SOUTHERN COMPANY *Energy to Serve Your World®* **VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2**	NUCLEAR POWER AND PRESSURIZER PRESSURE TRANSIENTS FOR FEEDWATER CONTROL VALVE MALFUNCTION
	FIGURE 15.1.2–1





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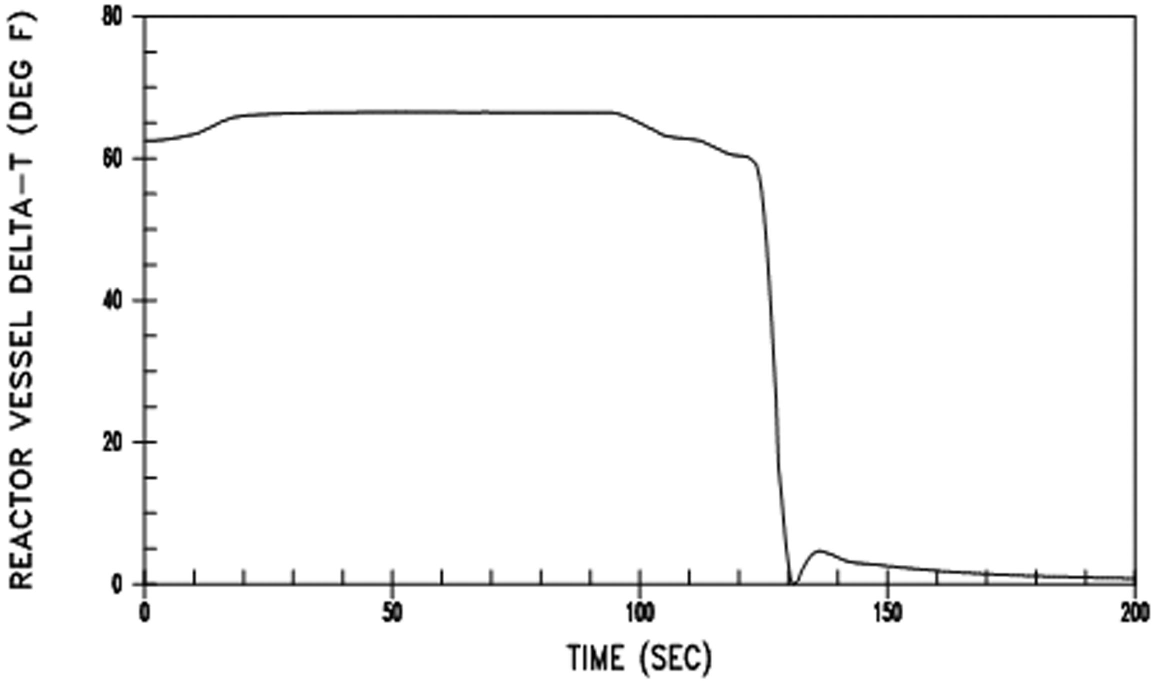
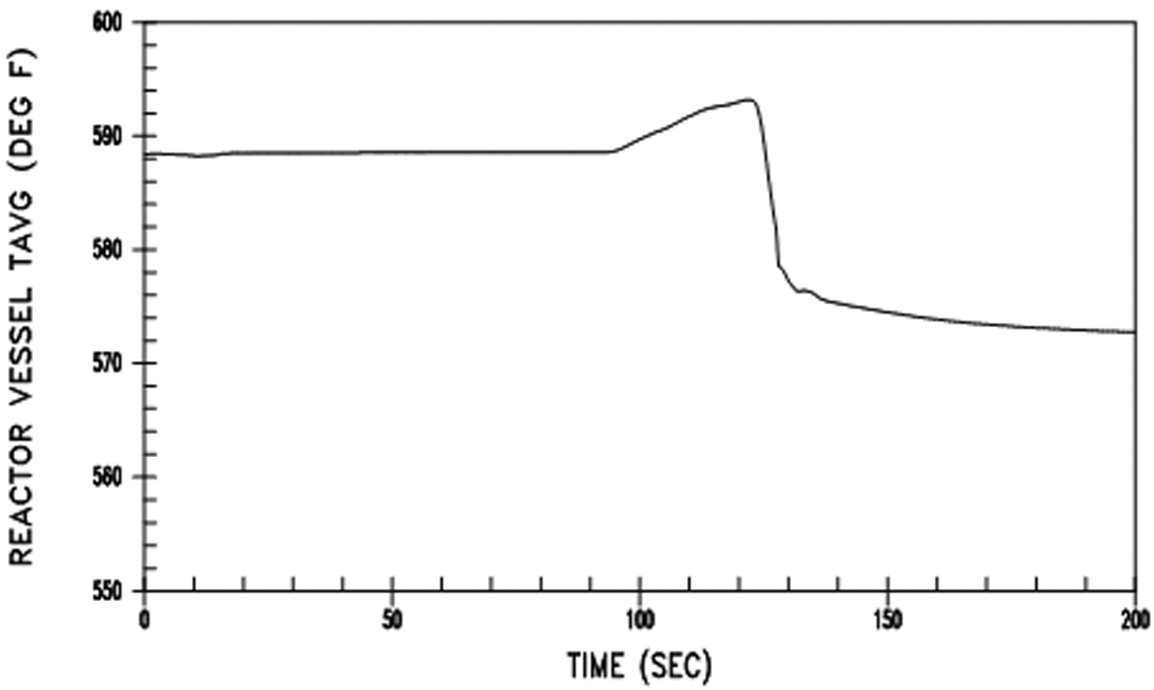
SOUTHERN COMPANY *Energy to Serve Your World®* VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2	CORE HEAT FLUX AND DNBR TRANSIENTS FOR FEEDWATER CONTROL VALVE MALFUNCTION
	FIGURE 15.1.2-2





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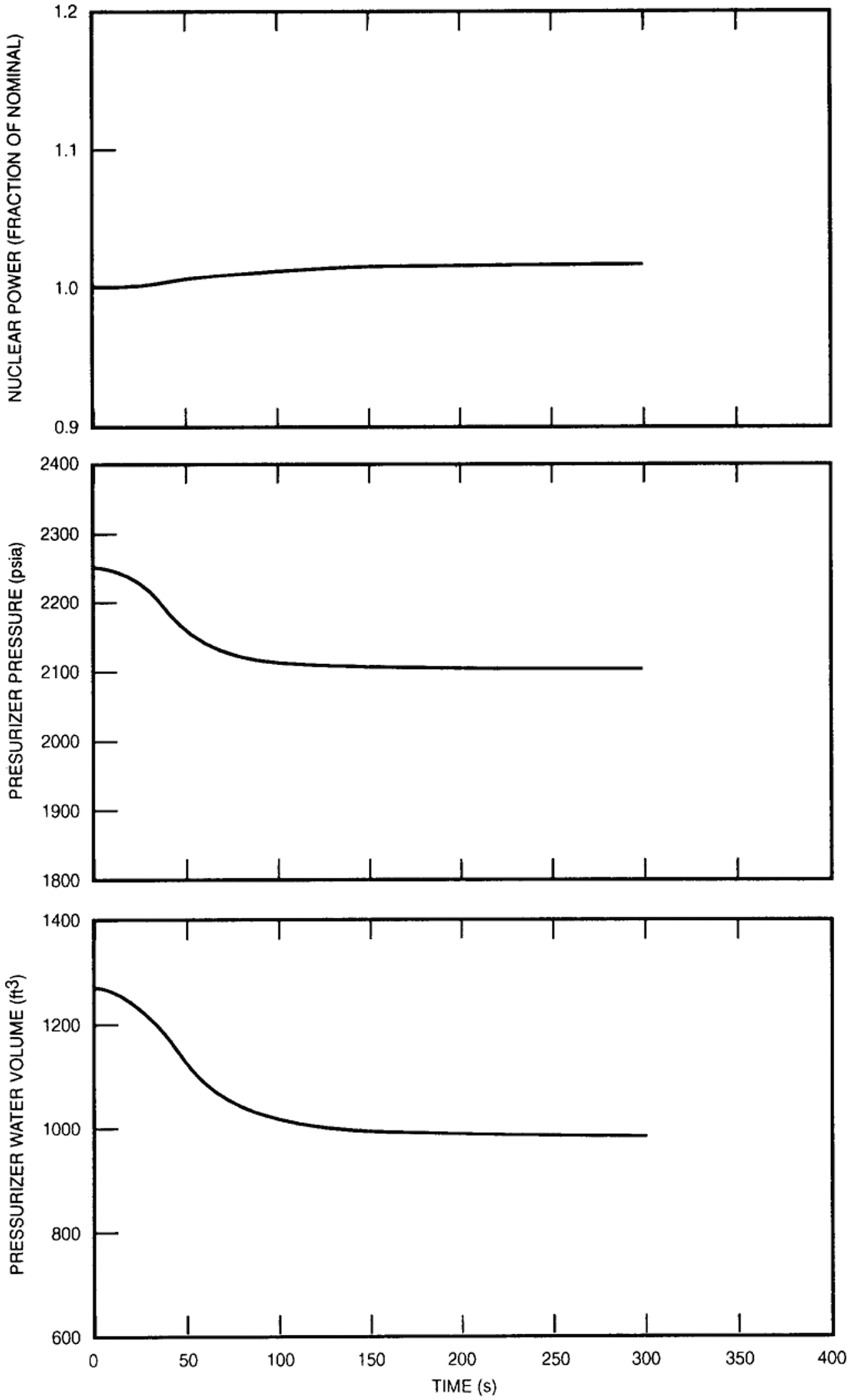
SOUTHERN COMPANY
Energy to Serve Your World®

VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

REACTOR VESSEL ΔT AND T_{AVG}
TRANSIENTS FOR FEEDWATER CONTROL VALVE MALFUNCTION

FIGURE 15.1.2-3



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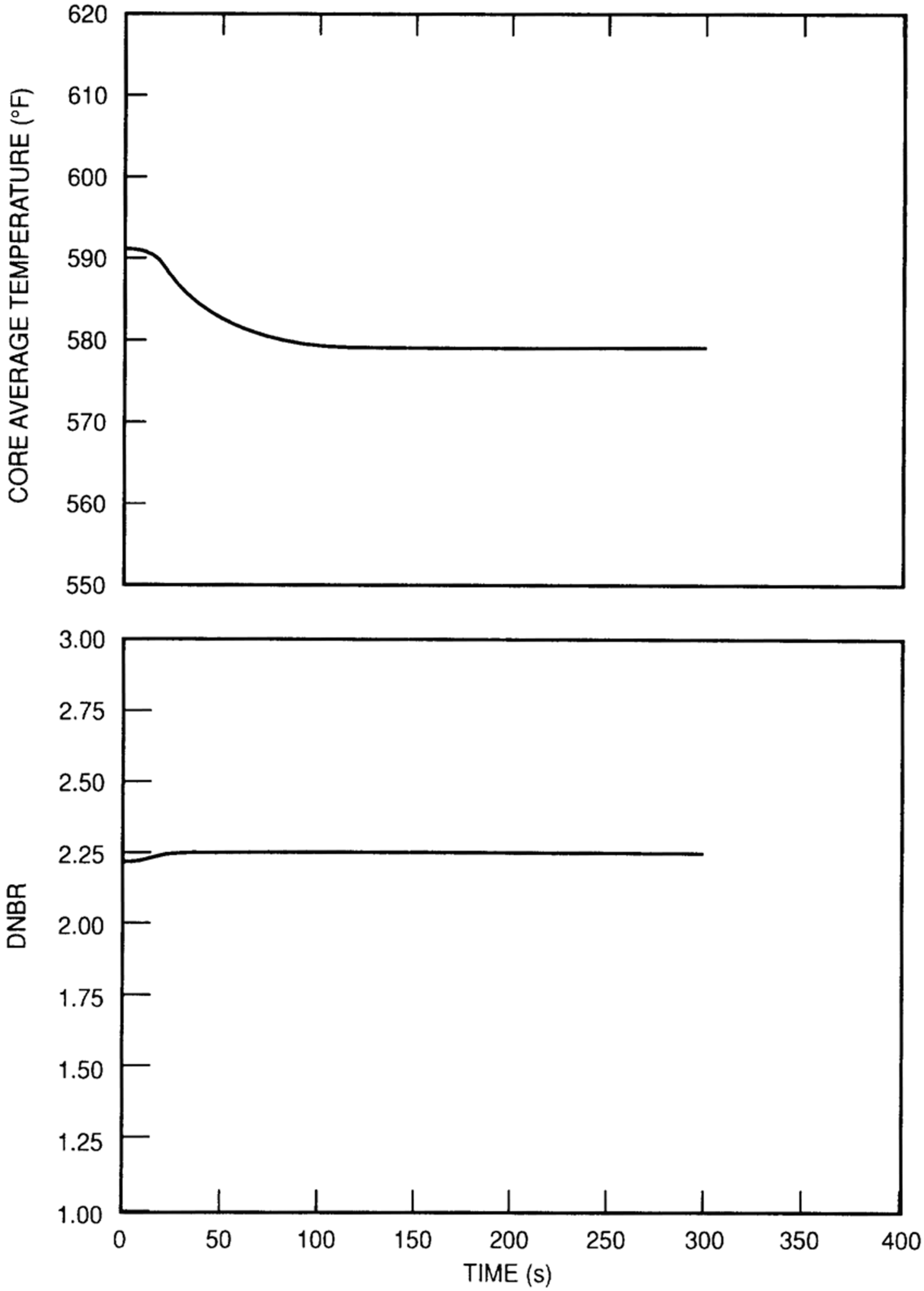
SOUTHERN COMPANY *Energy to Serve Your World®* **VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2**	TEN PERCENT STEP LOAD INCREASE, MINIMUM MODERATOR FEEDBACK, MANUAL ROD CONTROL
	FIGURE 15.1.3–1



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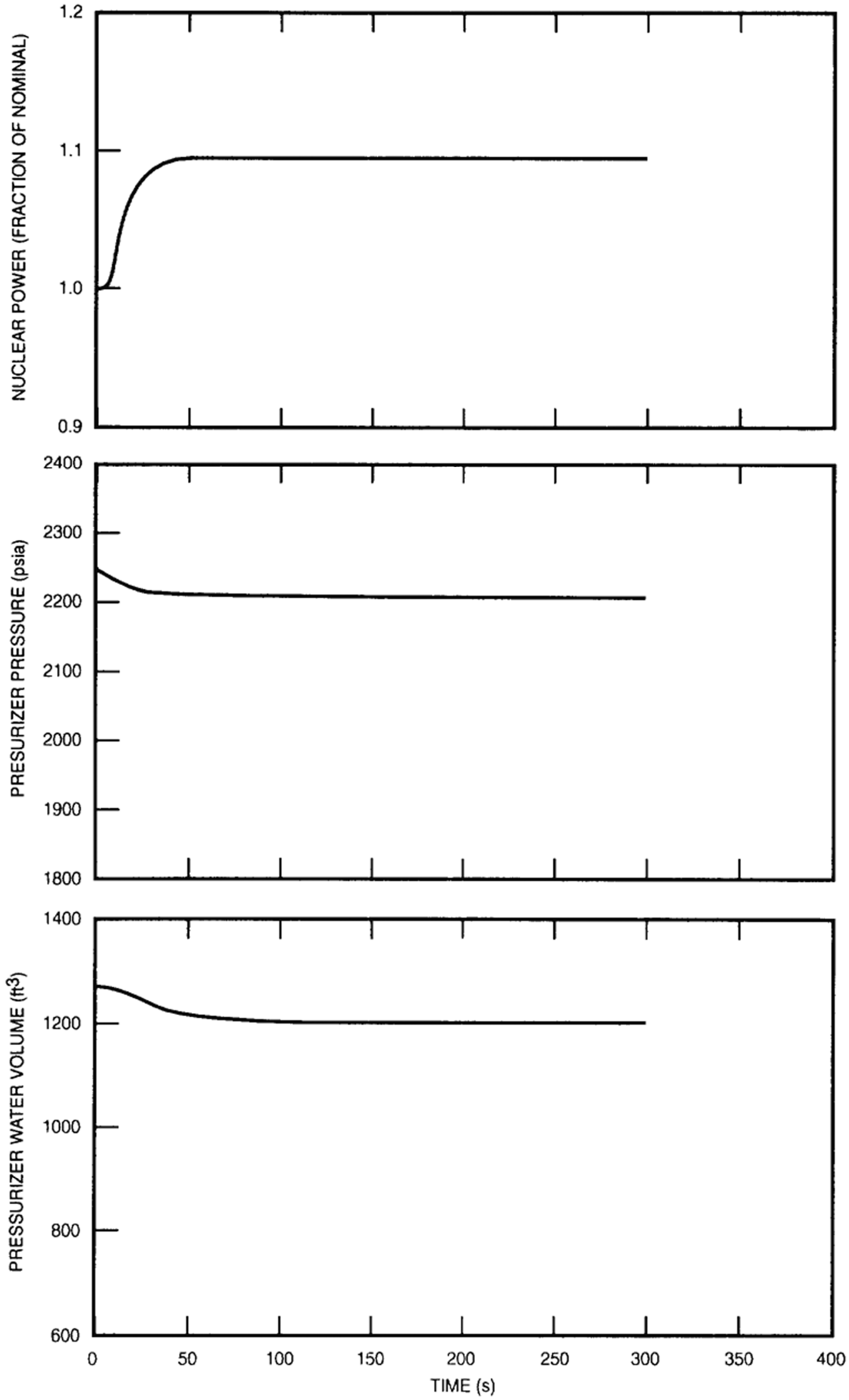
VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

TEN PERCENT STEP LOAD INCREASE, MINIMUM MODERATOR FEEDBACK, MANUAL ROD CONTROL

FIGURE 15.1.3–2



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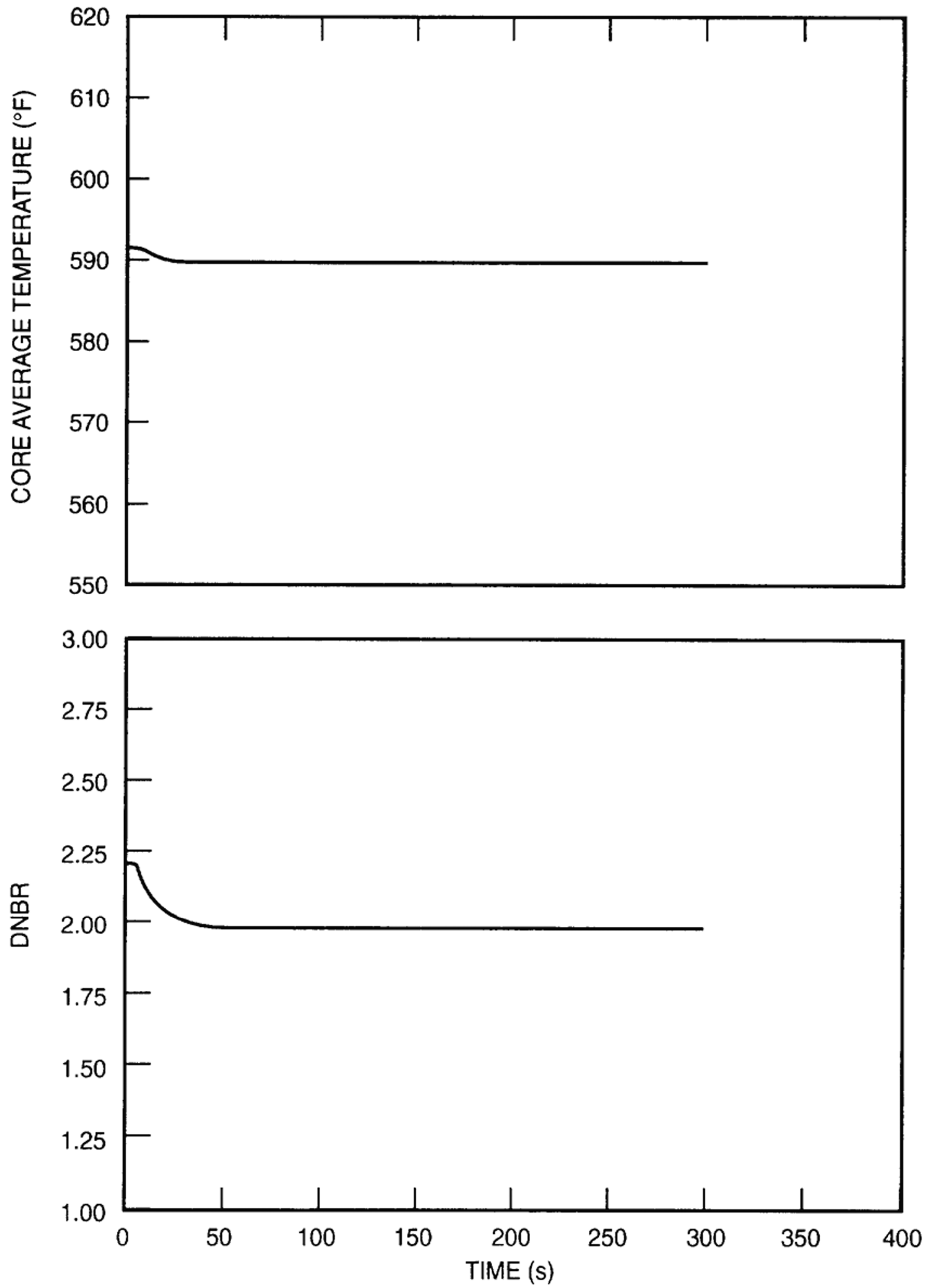
VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

TEN PERCENT STEP LOAD INCREASE, MAXIMUM MODERATOR FEEDBACK, MANUAL ROD CONTROL

FIGURE 15.1.3–3



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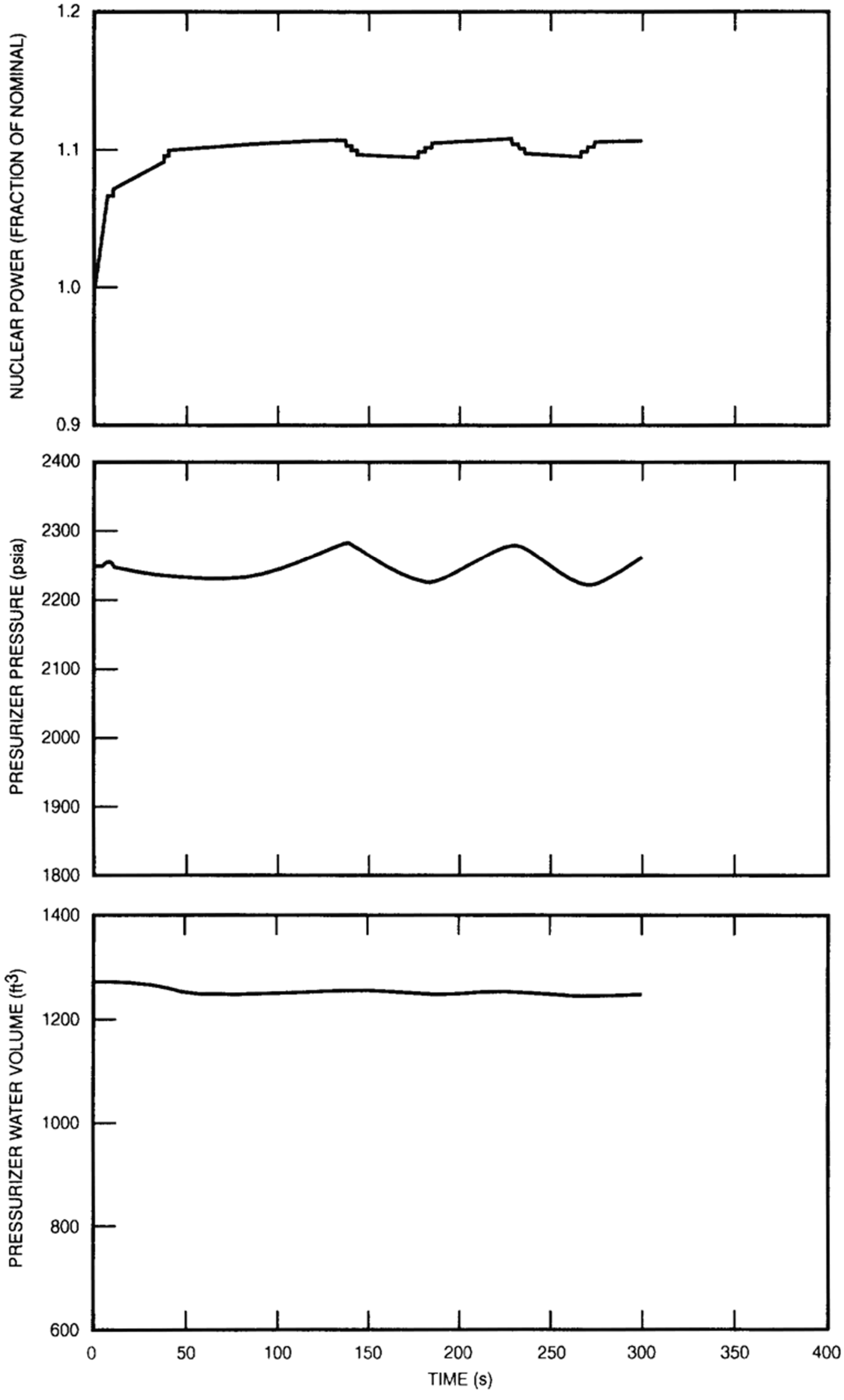
SOUTHERN COMPANY *Energy to Serve Your World®*	**VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2**	TEN PERCENT LOAD INCREASE, MAXIMUM MODERATOR FEEDBACK, MANUAL ROD CONTROL FIGURE 15.1.3–4



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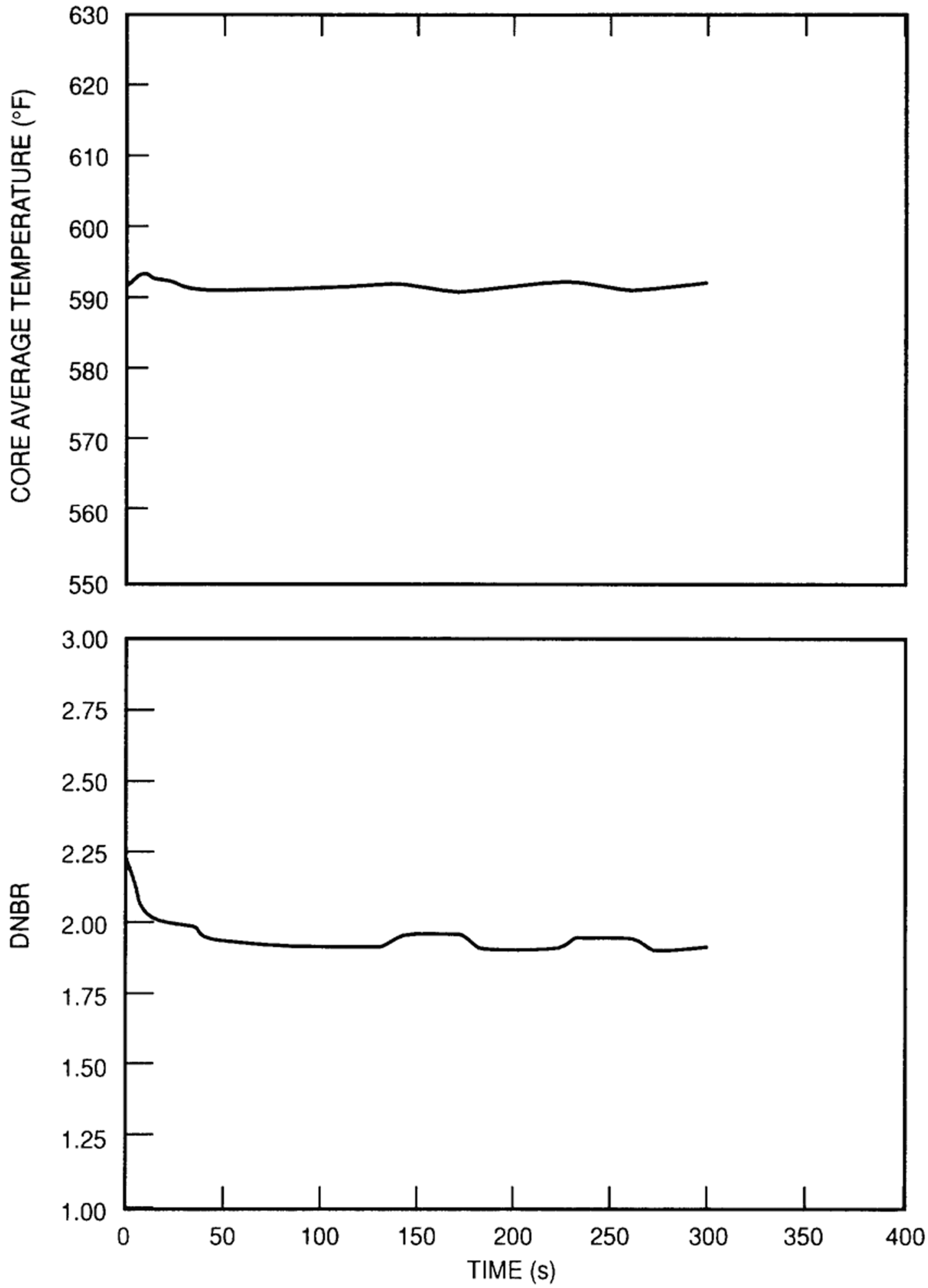
SOUTHERN COMPANY Energy to Serve Your World®	VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2	TEN PERCENT STEP LOAD INCREASE, MINIMUM MODERATOR FEEDBACK, AUTOMATIC REACTOR CONTROL
		FIGURE 15.1.3–5



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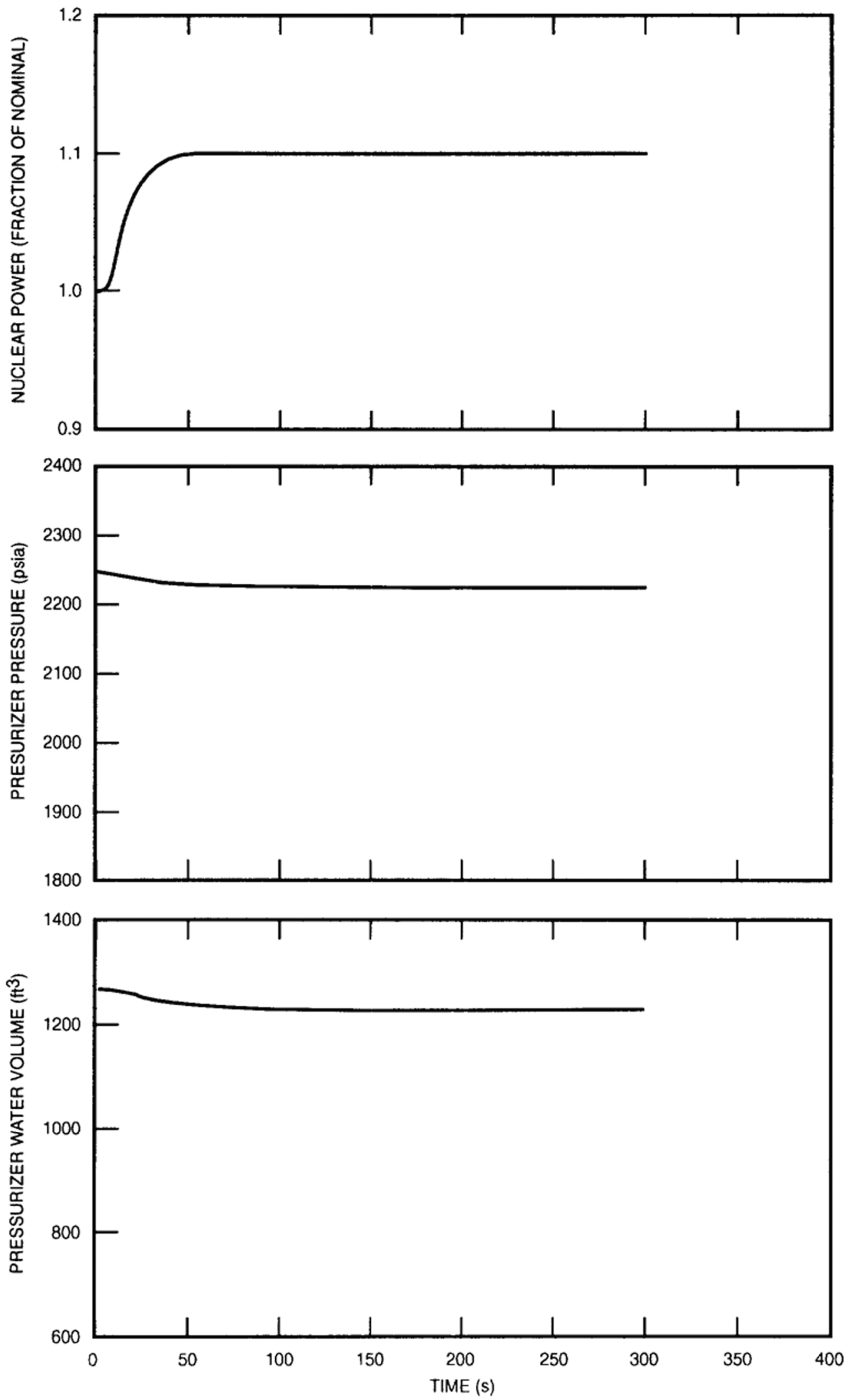
VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

TEN PERCENT STEP LOAD INCREASE, MINIMUM MODERATOR FEEDBACK, AUTOMATIC ROD CONTROL

FIGURE 15.1.3–6



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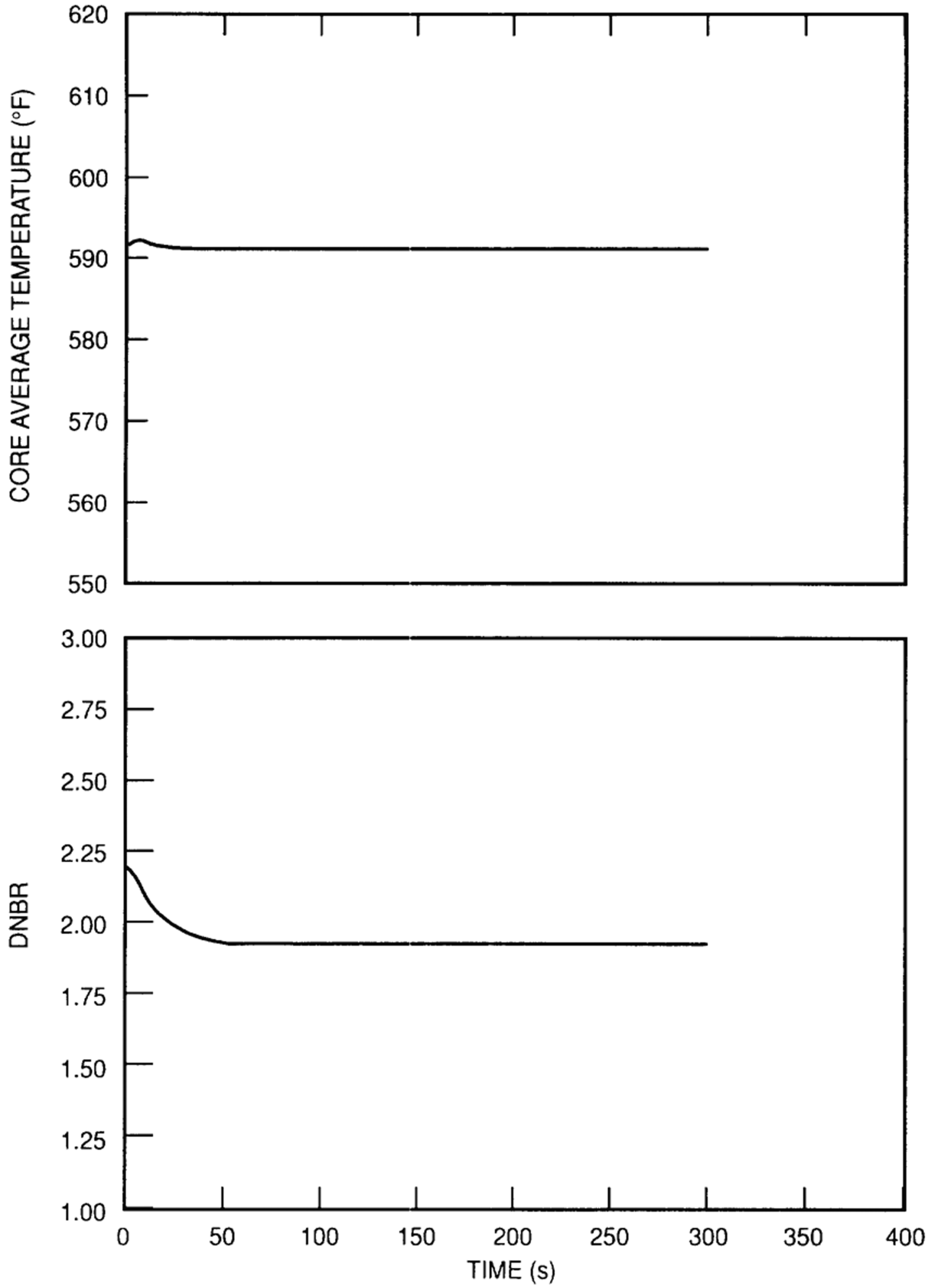
VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

TEN PERCENT STEP LOAD INCREASE, MAXIMUM MODERATOR FEEDBACK, AUTOMATIC ROD CONTROL

FIGURE 15.1.3–7



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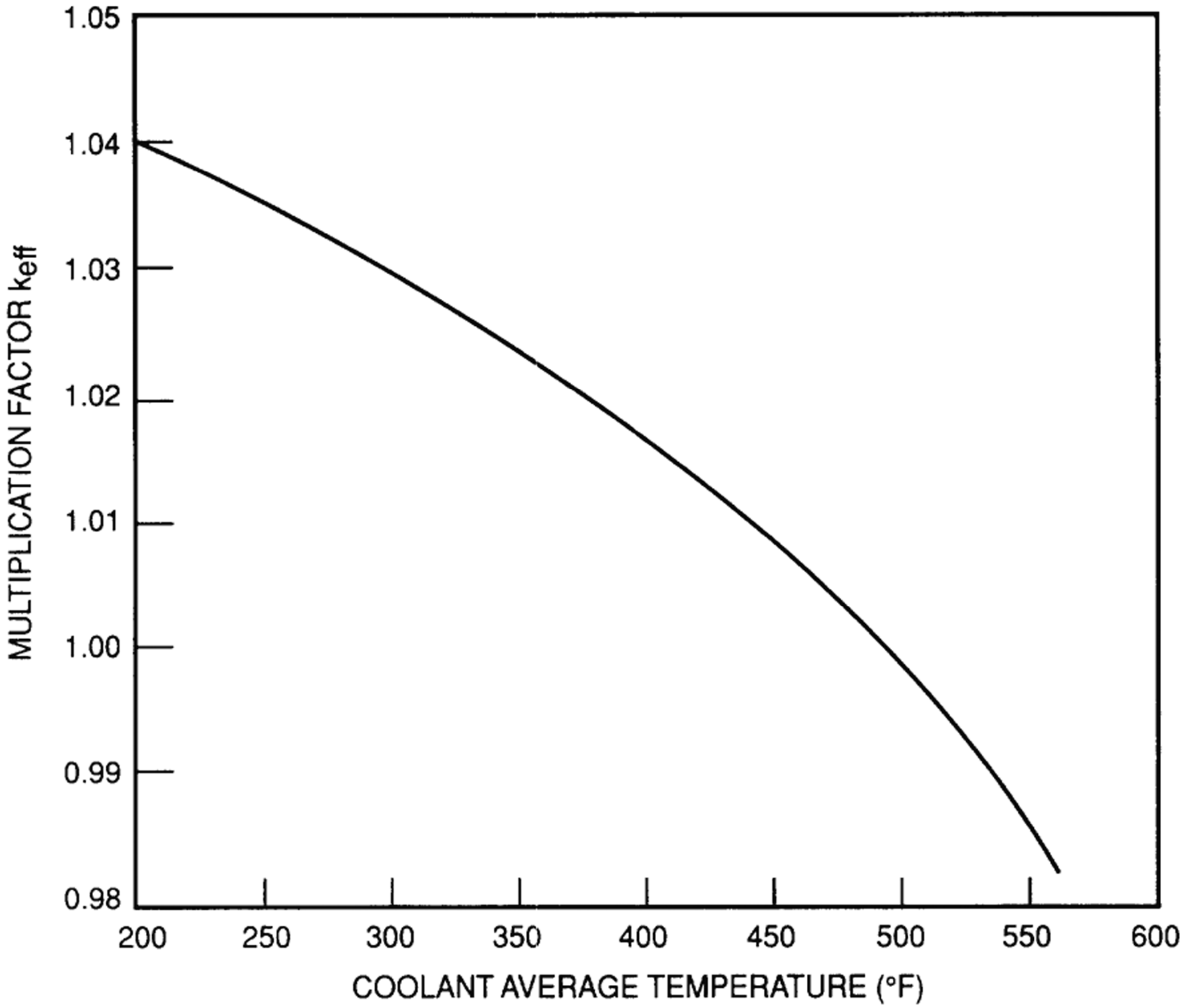
SOUTHERN COMPANY *Energy to Serve Your World®* **VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2**	TEN PERCENT STEP LOAD INCREASE, MAXIMUM MODERATOR FEEDBACK, AUTOMATIC ROD CONTROL
	FIGURE 15.1.3–8



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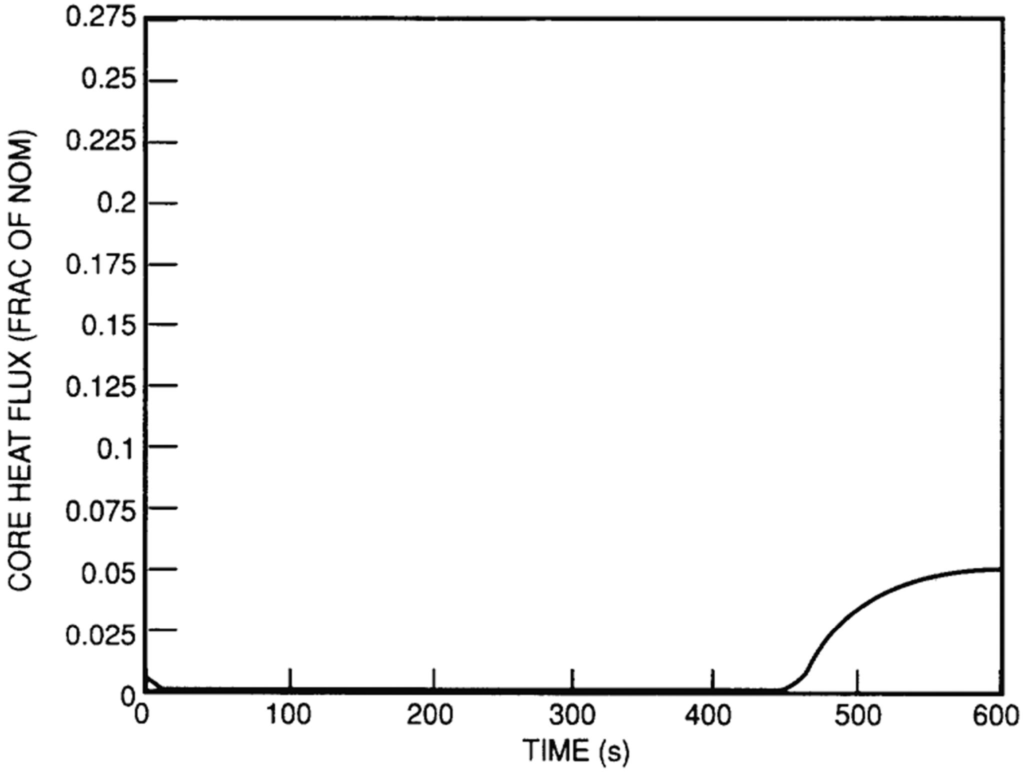
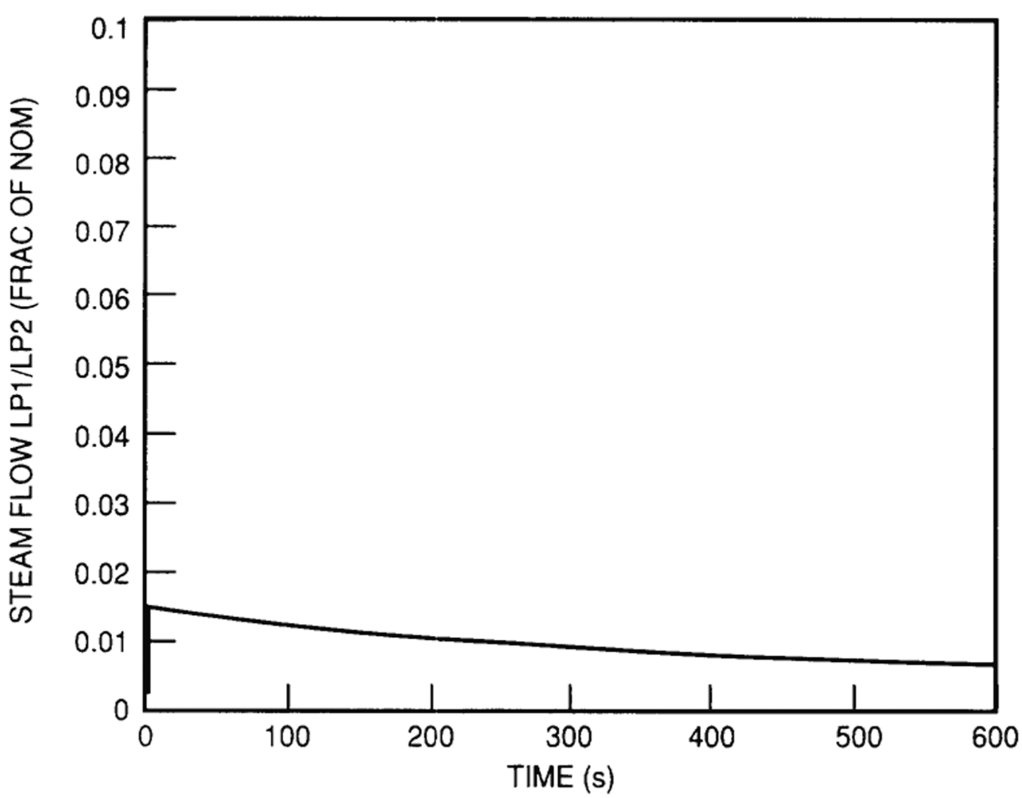
SOUTHERN COMPANY *Energy to Serve Your World®*	VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2	KEFF VS. COOLANT AVERAGE TEMPERATURE FIGURE 15.1.4–1





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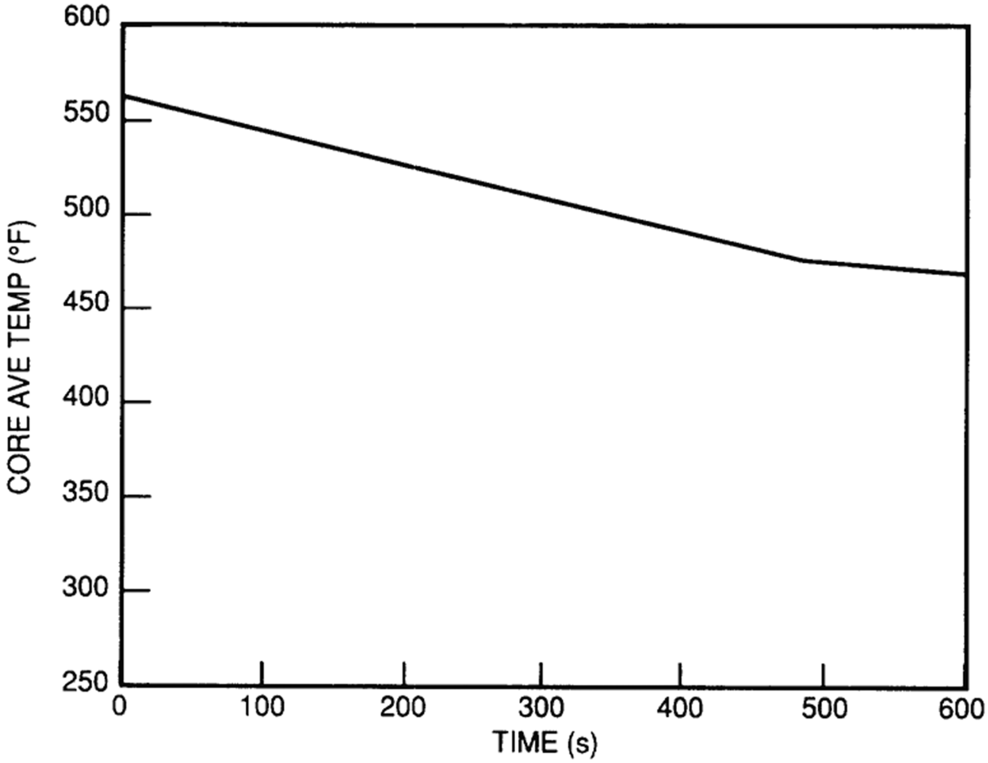
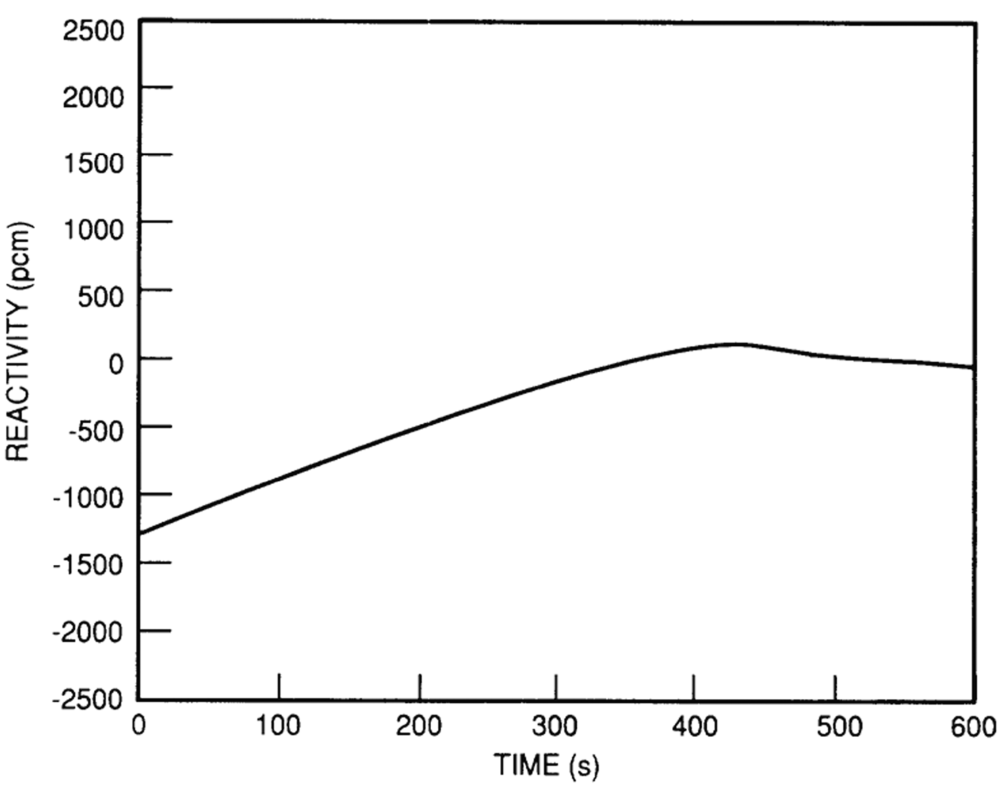
SOUTHERN COMPANY Energy to Serve Your World®	**VOGTLE** **ELECTRIC GENERATING PLANT** **UNIT 1 AND UNIT 2**	FAILURE OF A STEAM GENERATOR SAFETY OR DUMP VALVE (HEAT FLUX VS. TIME, STEAM FLOW VS. TIME) FIGURE 15.1.4–2





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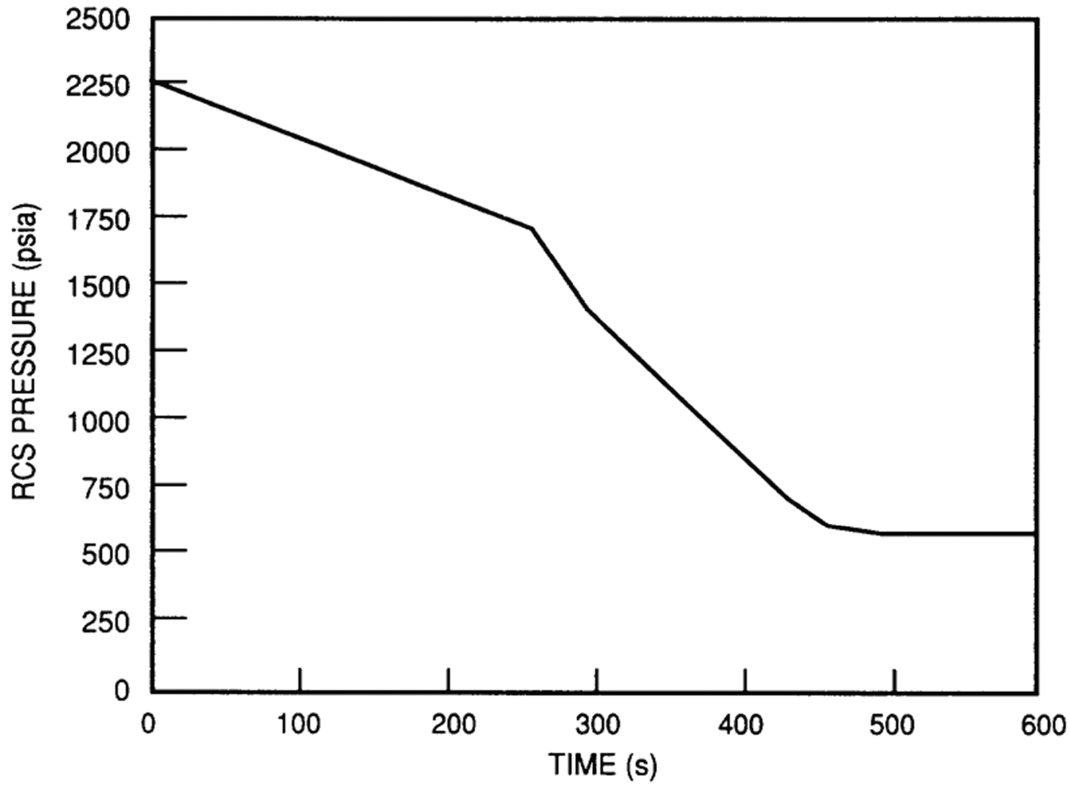
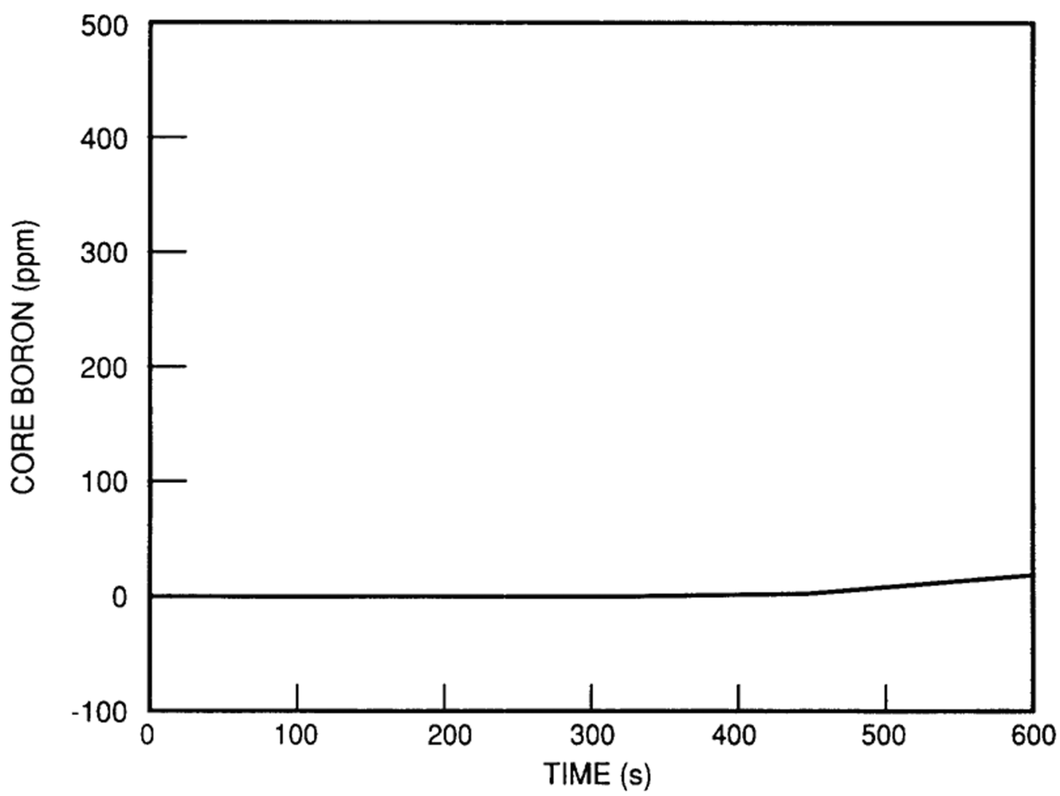
SOUTHERN COMPANY *Energy to Serve Your World®*	**VOGTLE** **ELECTRIC GENERATING PLANT** **UNIT 1 AND UNIT 2**	FAILURE OF A STEAM CENERATOR SAFETY OR DUMP VALVE (AVERAGE TEMPERATURE VS. TIME, REACTIVITY VS. TIME) FIGURE 15.1.4–3





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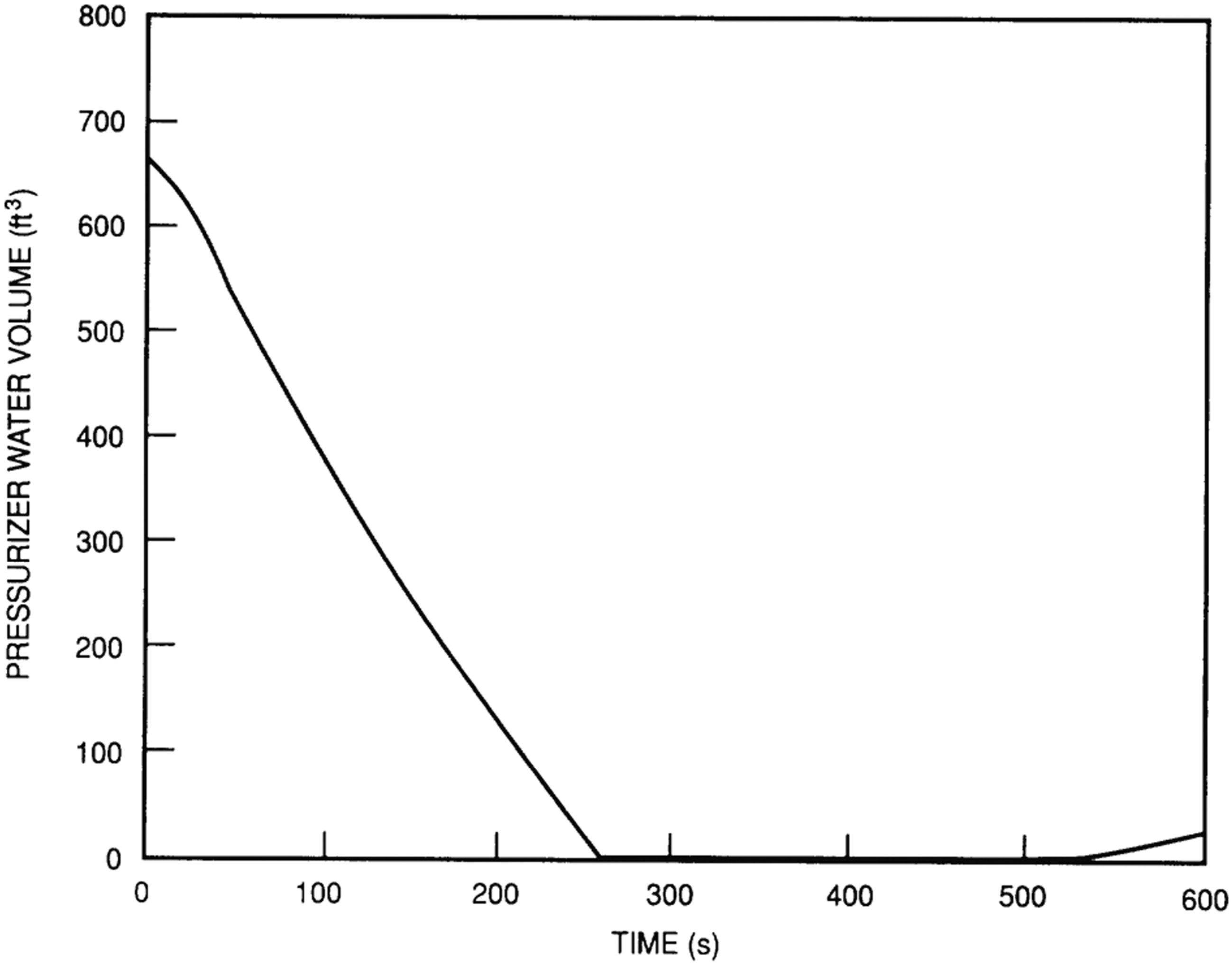
SOUTHERN COMPANY *Energy to Serve Your World®*	**VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2**	FAILURE OF A STEAM GENERATOR SAFETY OF DUMP VALVE (RCS PRESSURE VS. TIME, BORON CONCENTRATION VS. TIME)
		FIGURE 15.1.4–4



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SOUTHERN COMPANY
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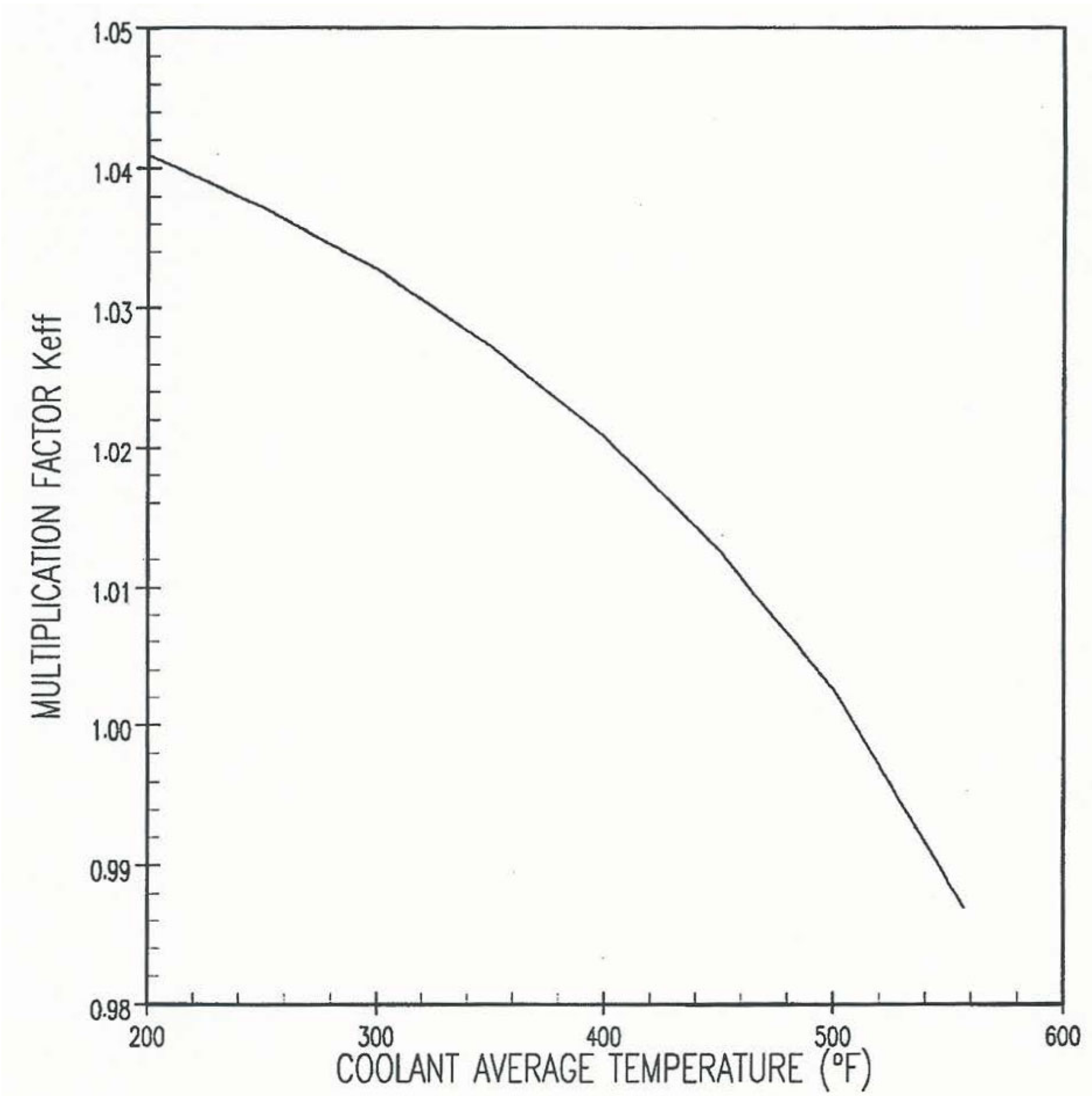
VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

FAILURE OF A STEAM GENERATOR
SAFETY OR DUMP VALVE
(PRESSURIZER WATER VOLUME VS. TIME)

FIGURE 15.1.4–5



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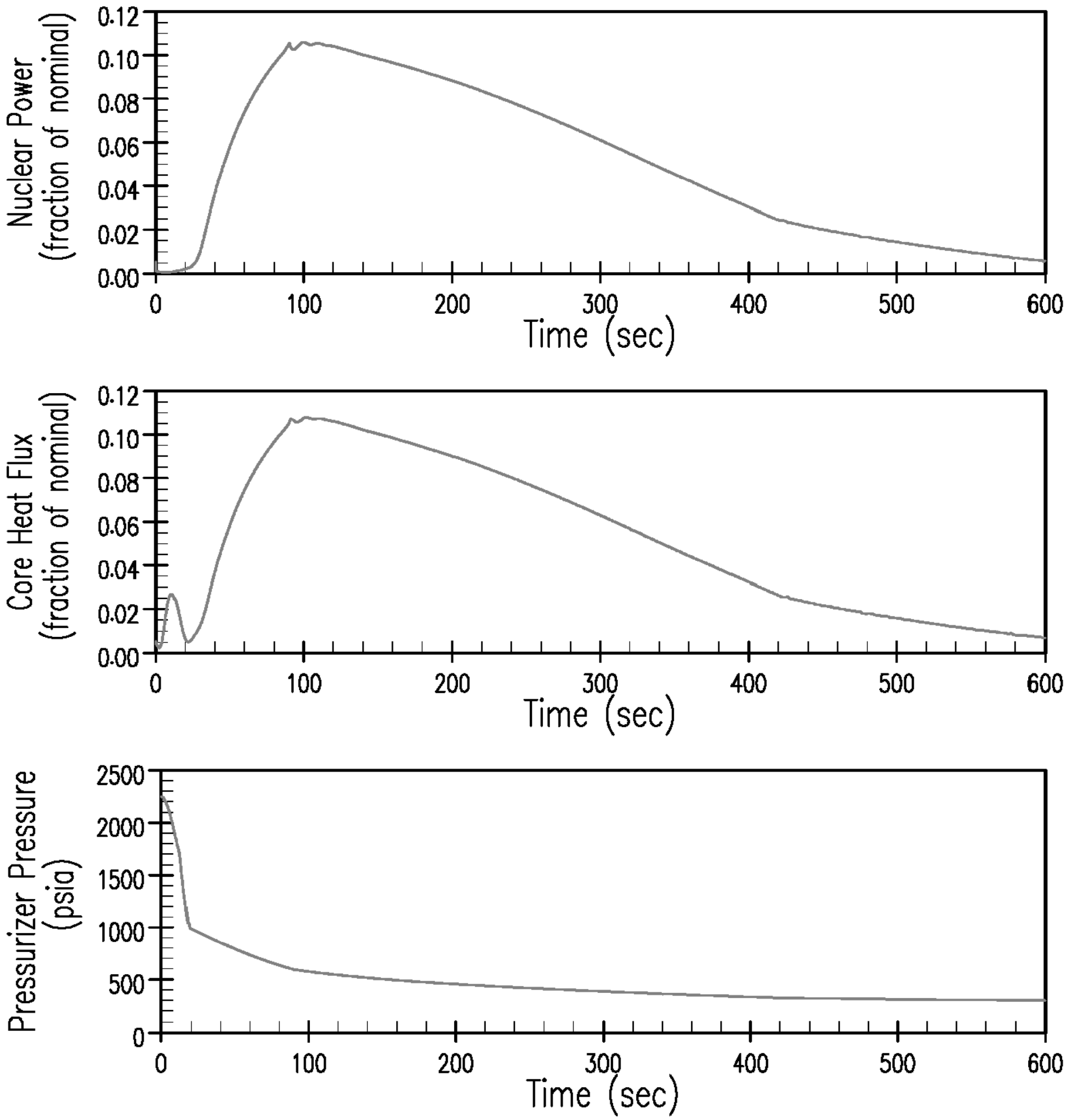
VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

1.4 ft^2 STEAM LINE RUPTURE AT HOT ZERO POWER Keff vs. COOLANT AVERAGE TEMPERATURE

FIGURE 15.1.5-0



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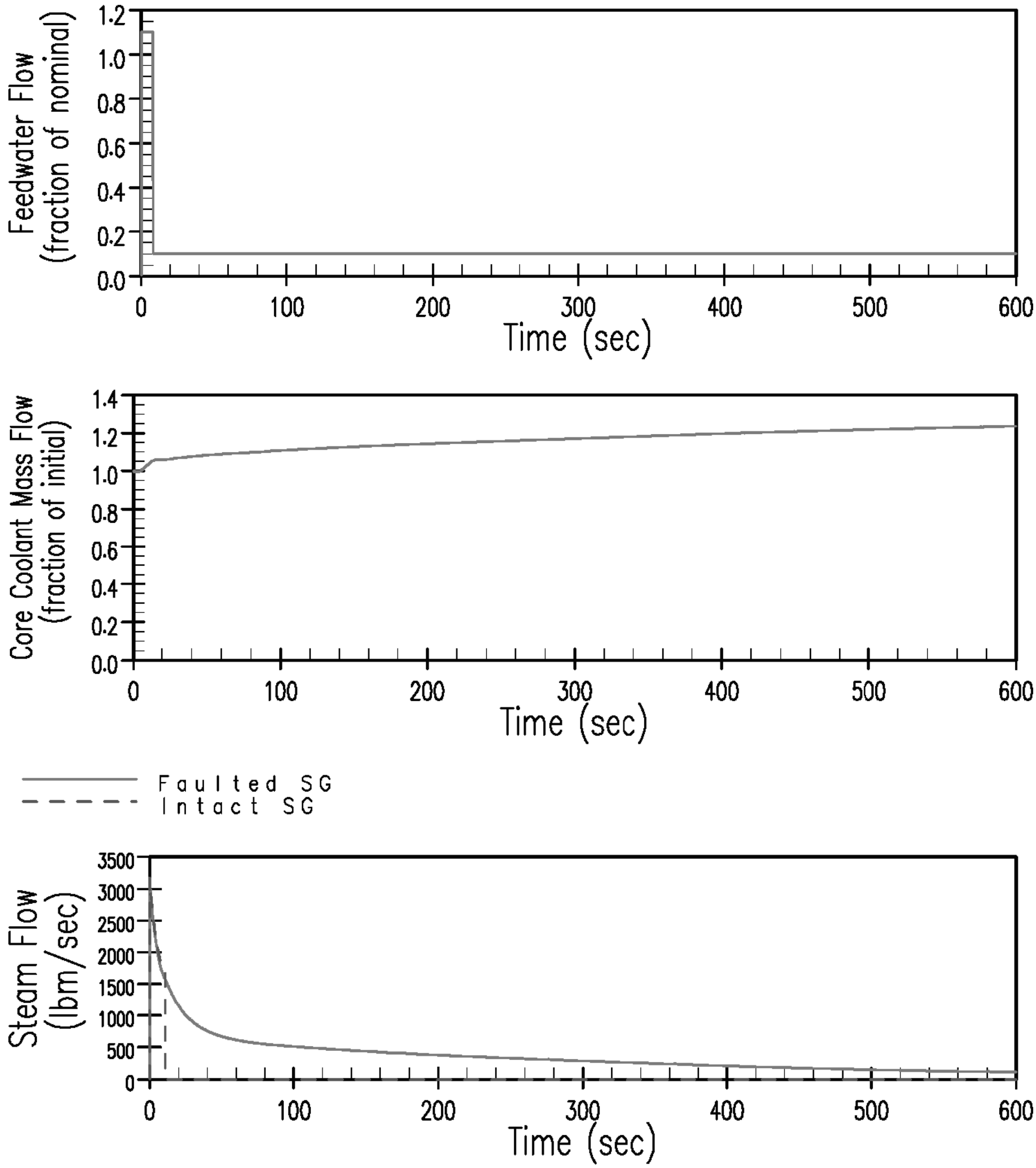
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

1.4 ft^2 STEAM LINE RUPTURE AT HOT ZERO WITH OFFSITE POWER AVAILABLE. NUCLEAR POWER, CORE HEAT FLUX AND PRESSURIZER PRESSURE TRANSIENTS

FIGURE 15.1.5-1



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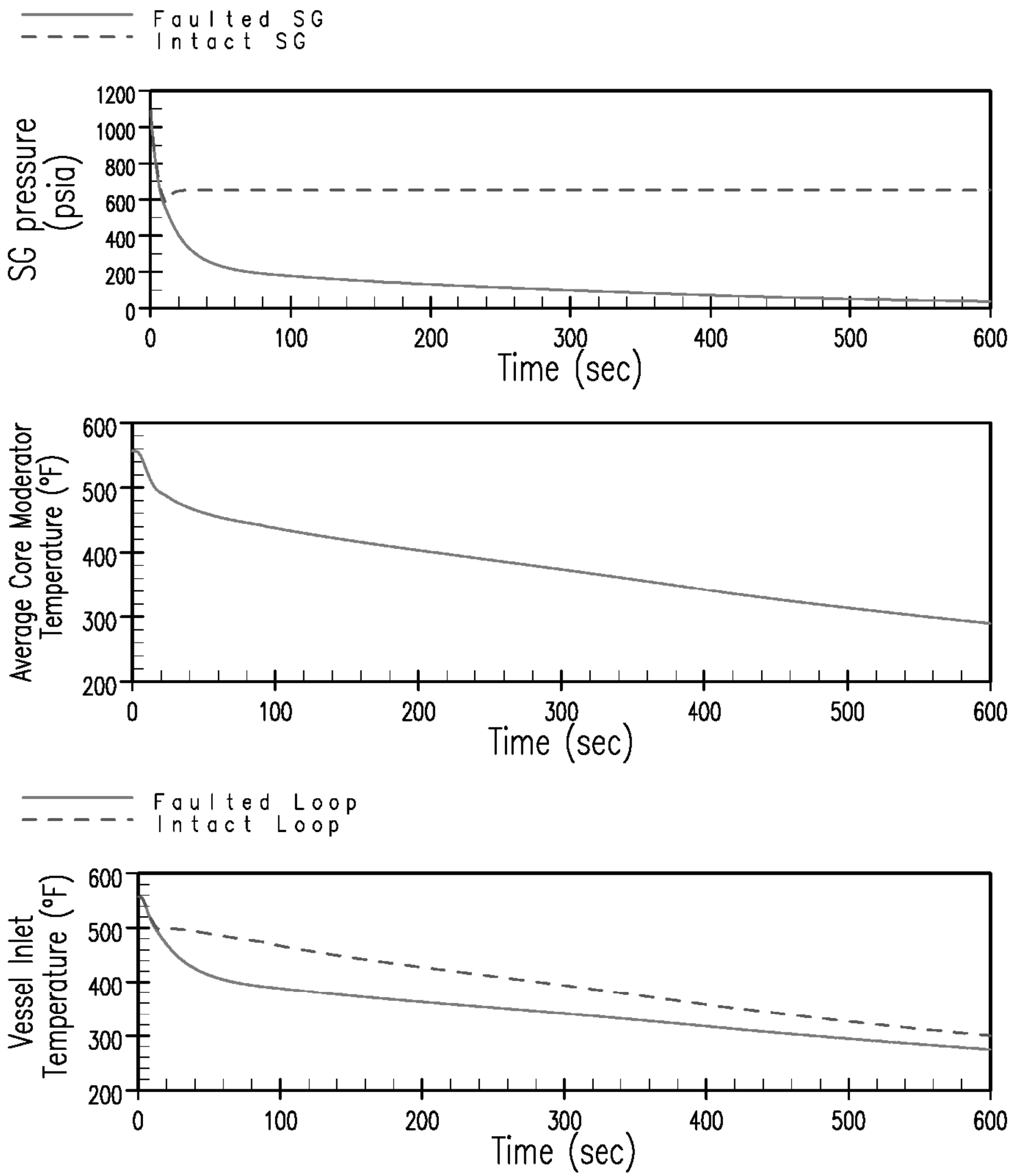
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

1.4 ft^2 STEAM LINE RUPTURE AT HOT ZERO POWER WITH OFFSITE POWER AVAILABLE. FEED FLOW, CORE FLOW AND STEAM FLOW TRANSIENTS

FIGURE 15.1.5-2



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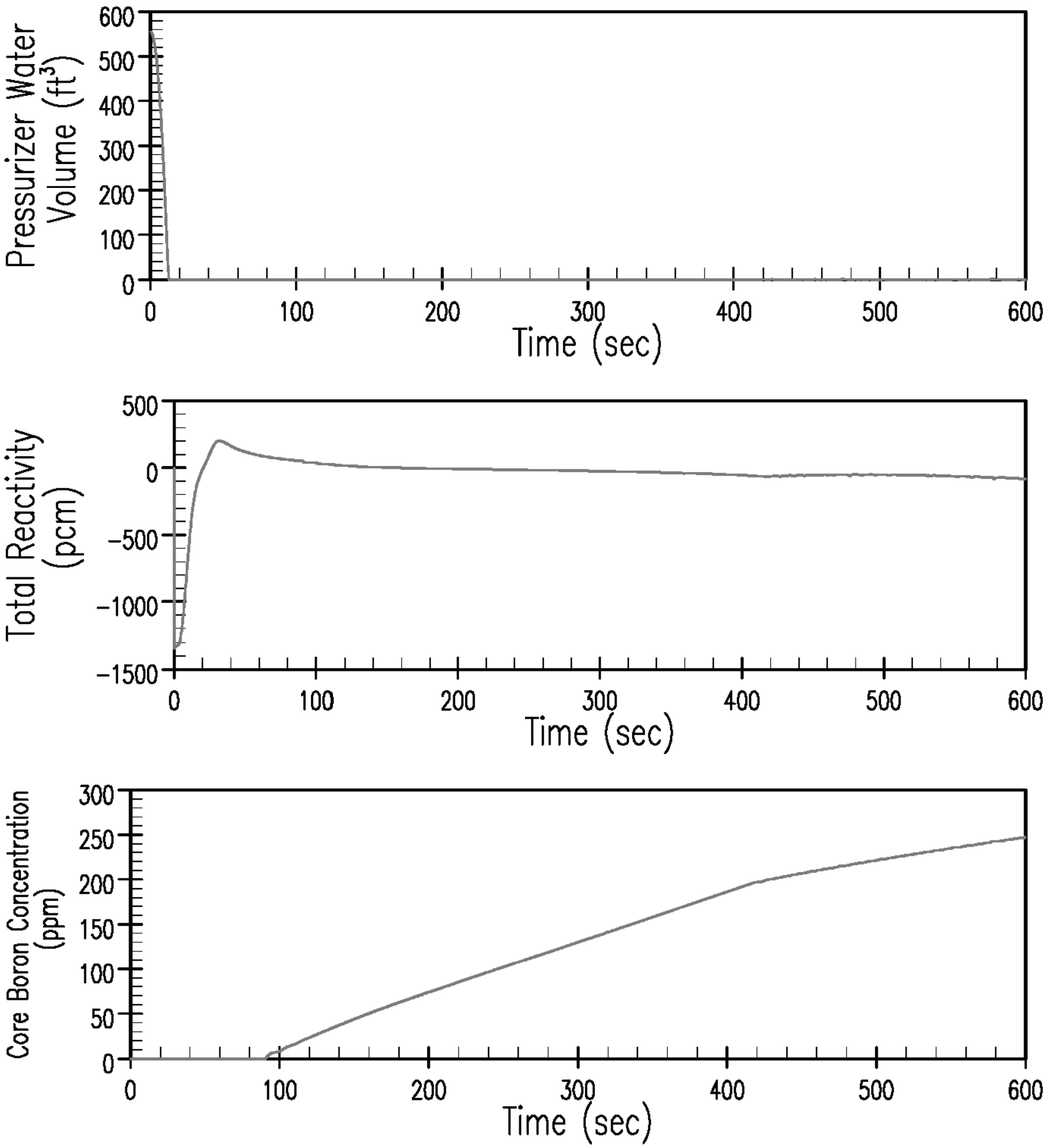
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

1.4 ft^2 STEAM LINE RUPTURE AT HOT ZERO POWER WITH OFFSITE POWER AVAILABLE. SG PRESSURE, AVERAGE CORE MODERATOR TEMPERATURE AND VESSEL INLET TEMPERATURE TRANSIENTS

FIGURE 15.1.5-3



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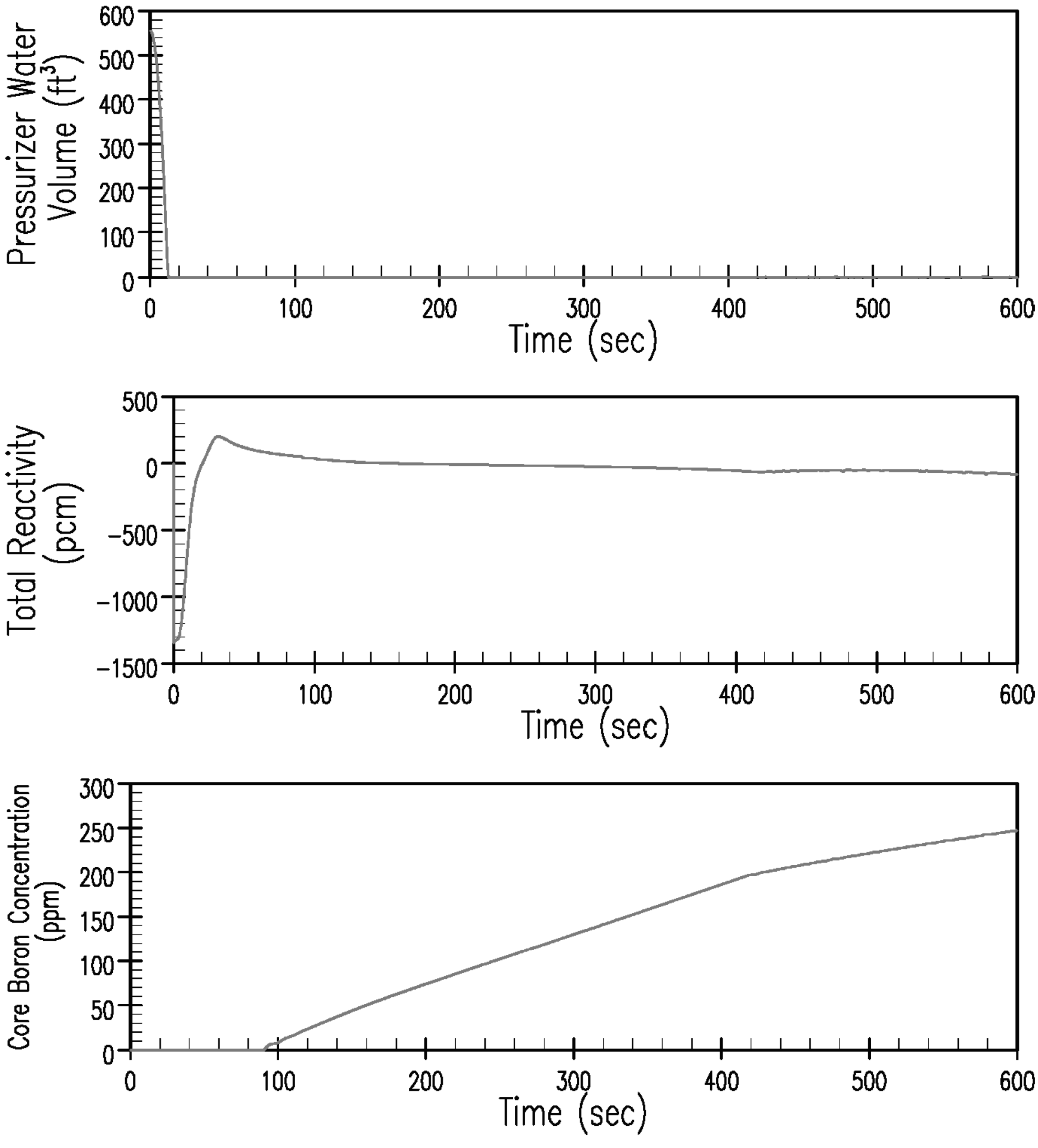
UNIT 1 AND UNIT 2

1.4 ft^2 STEAM LINE RUPTURE AT HOT ZERO POWER WITH OFFSITE POWER AVAILABLE. PRESSURIZER WATER VOLUME, REACTIVITY, AND CORE BORON TRANSIENTS

FIGURE 15.1.5–4



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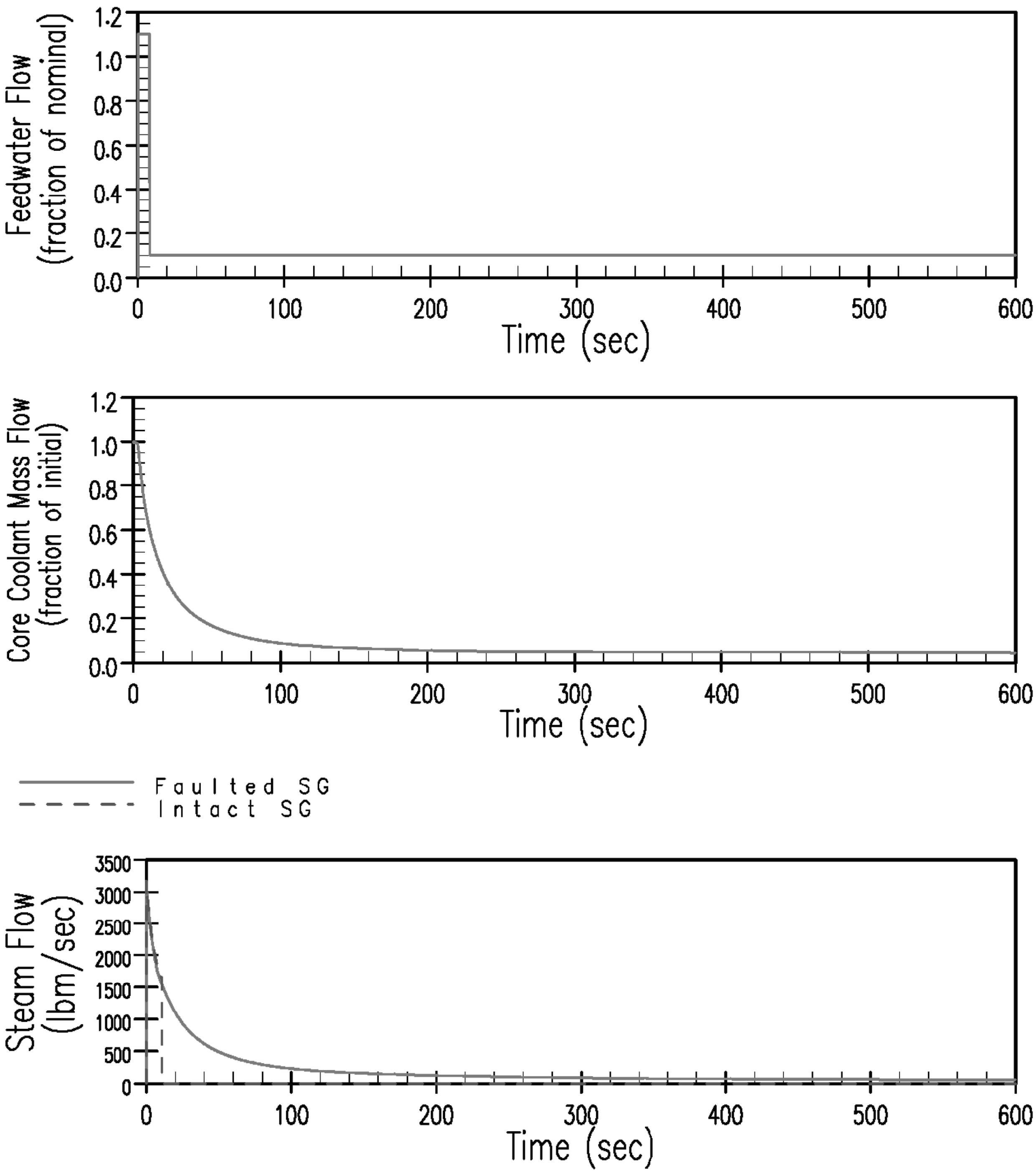
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

1.4 ft^2 STEAM LINE RUPTURE AT HOT ZERO POWER WITH OFFSITE POWER AVAILABLE.
NUCLEAR POWER, CORE HEAT FLUX, AND PRESSURIZER PRESSURE TRANSIENTS

FIGURE 15.1.5–5



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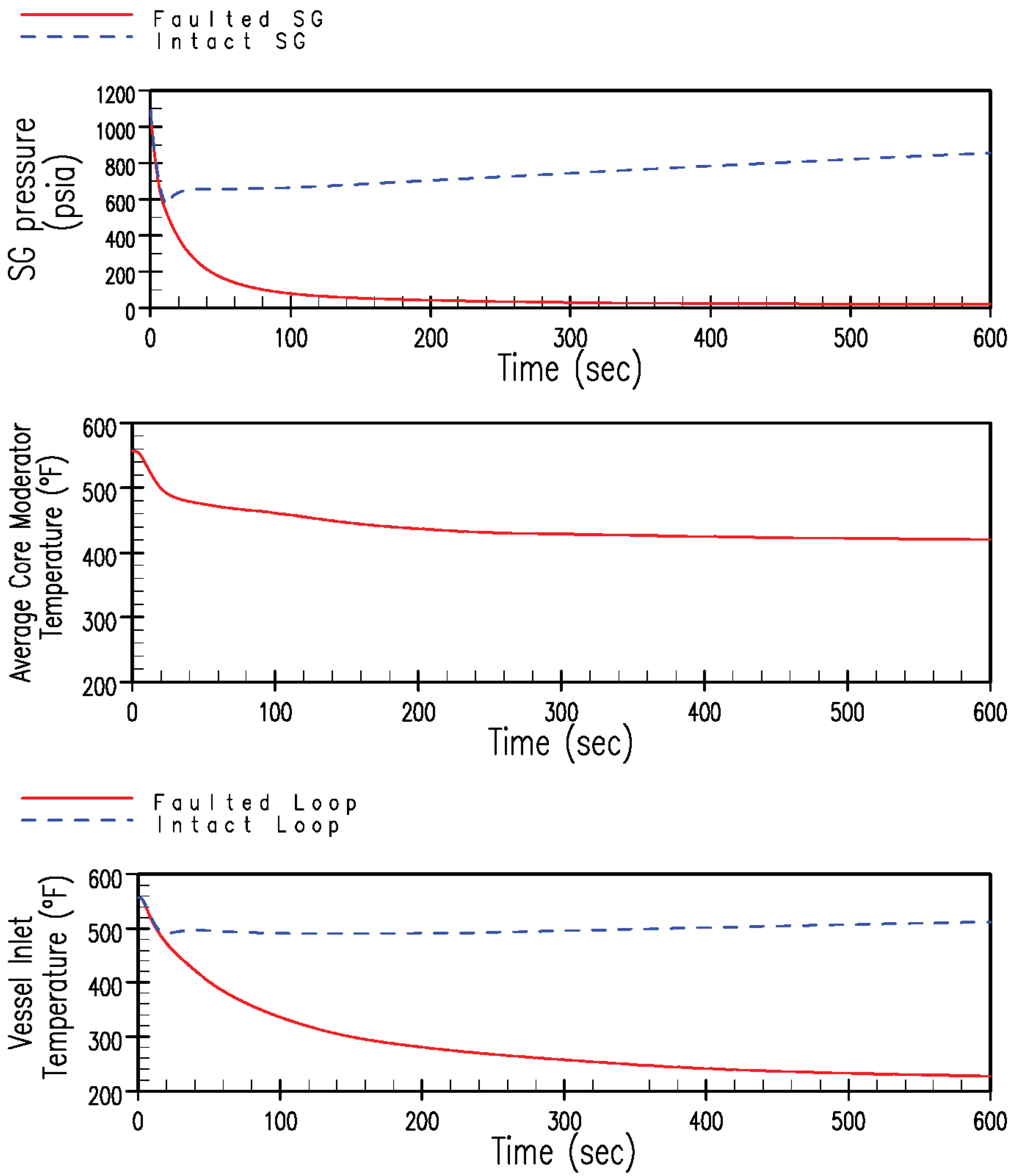
VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

1.4 ft² STEAM LINE RUPTURE AT HOT ZERO POWER WITHOUT OFFSITE POWER AVAILABLE. FEED FLOW, CORE FLOW, AND STEAM FLOW TRANSIENTS (UNIT 1 ONLY)

FIGURE 15.1.5–6



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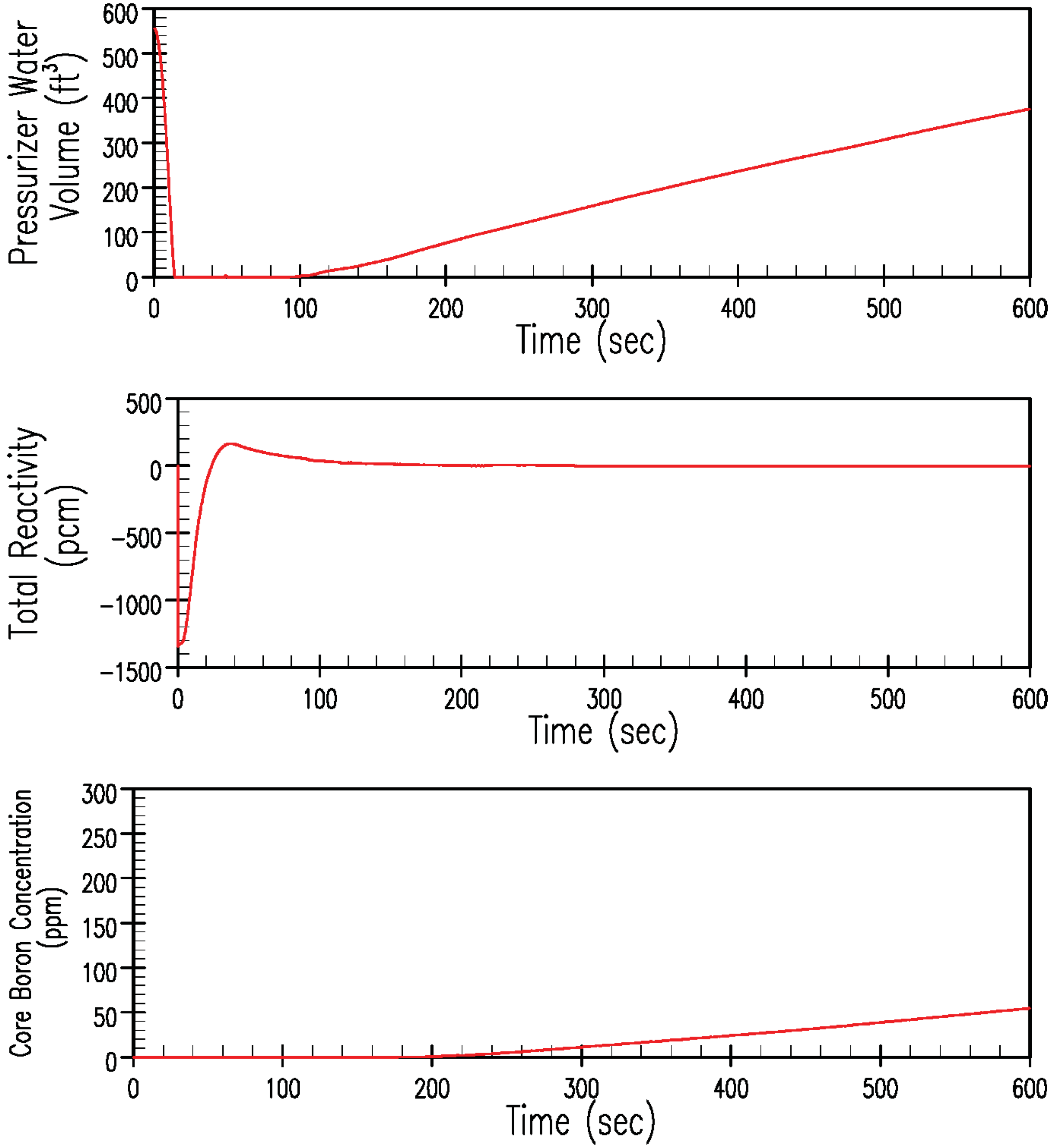
SOUTHERN COMPANY
Energy to Serve Your World®

VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

1.4 ft^2 STEAM LINE RUPTURE AT HOT ZERO POWER WITHOUT OFFSITE POWER AVAILABLE.
SG PRESSURE, AVERAGE CORE MODERATOR TEMPERATURE, AND VESSEL INLET TEMPERATURE TRANSIENTS

FIGURE 15.1.5–7



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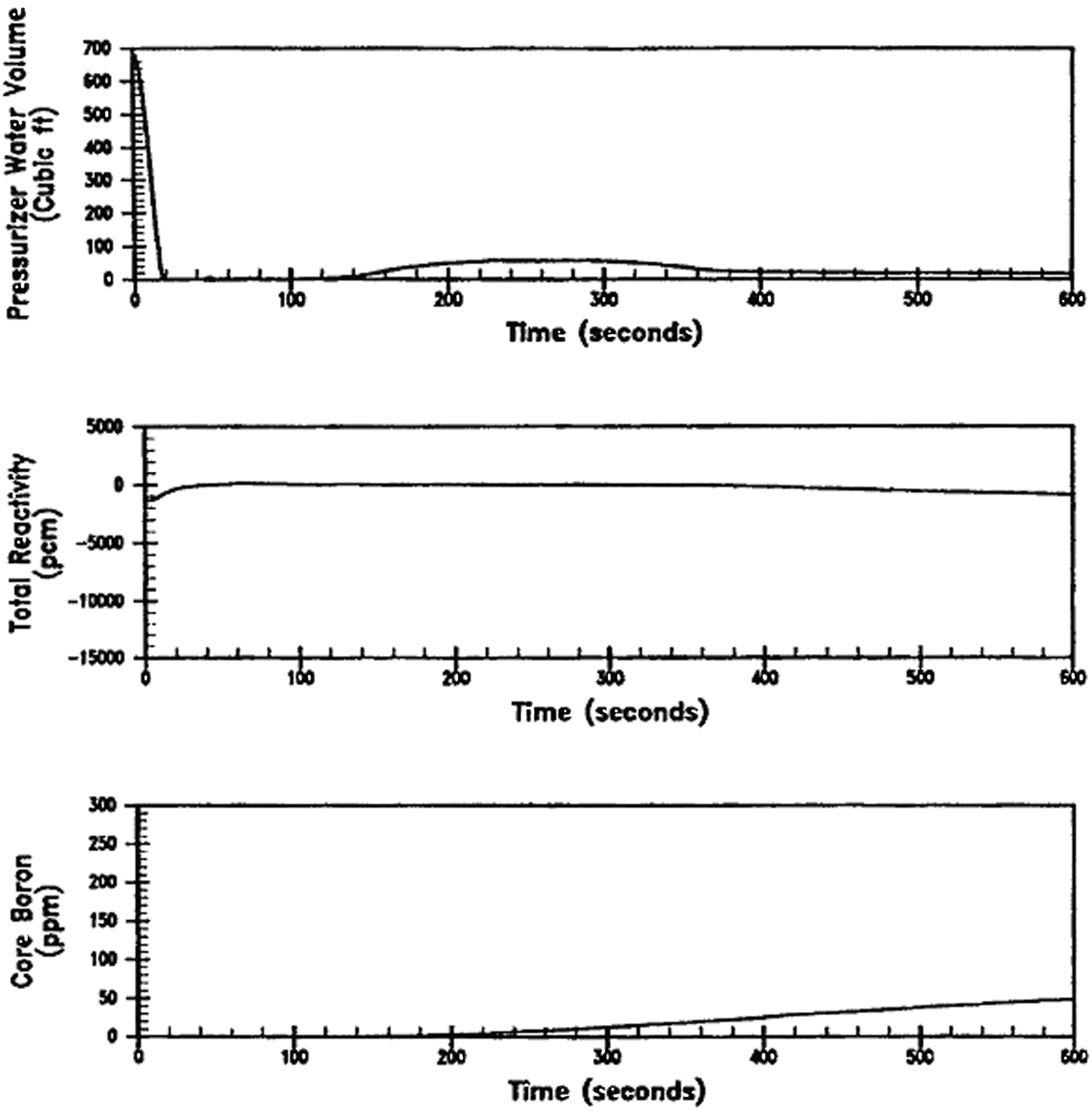
SOUTHERN COMPANY *Energy to Serve Your World®*	VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2	1.4 ft^2 STEAM LINE RUPTURE AT HOT ZERO POWER WITHOUT OFFSITE POWER AVAILABLE. PRESSURIZER WATER VOLUME, REACTIVITY, AND CORE BORON TRANSIENTS FIGURE 15.1.5–8



(Historical)

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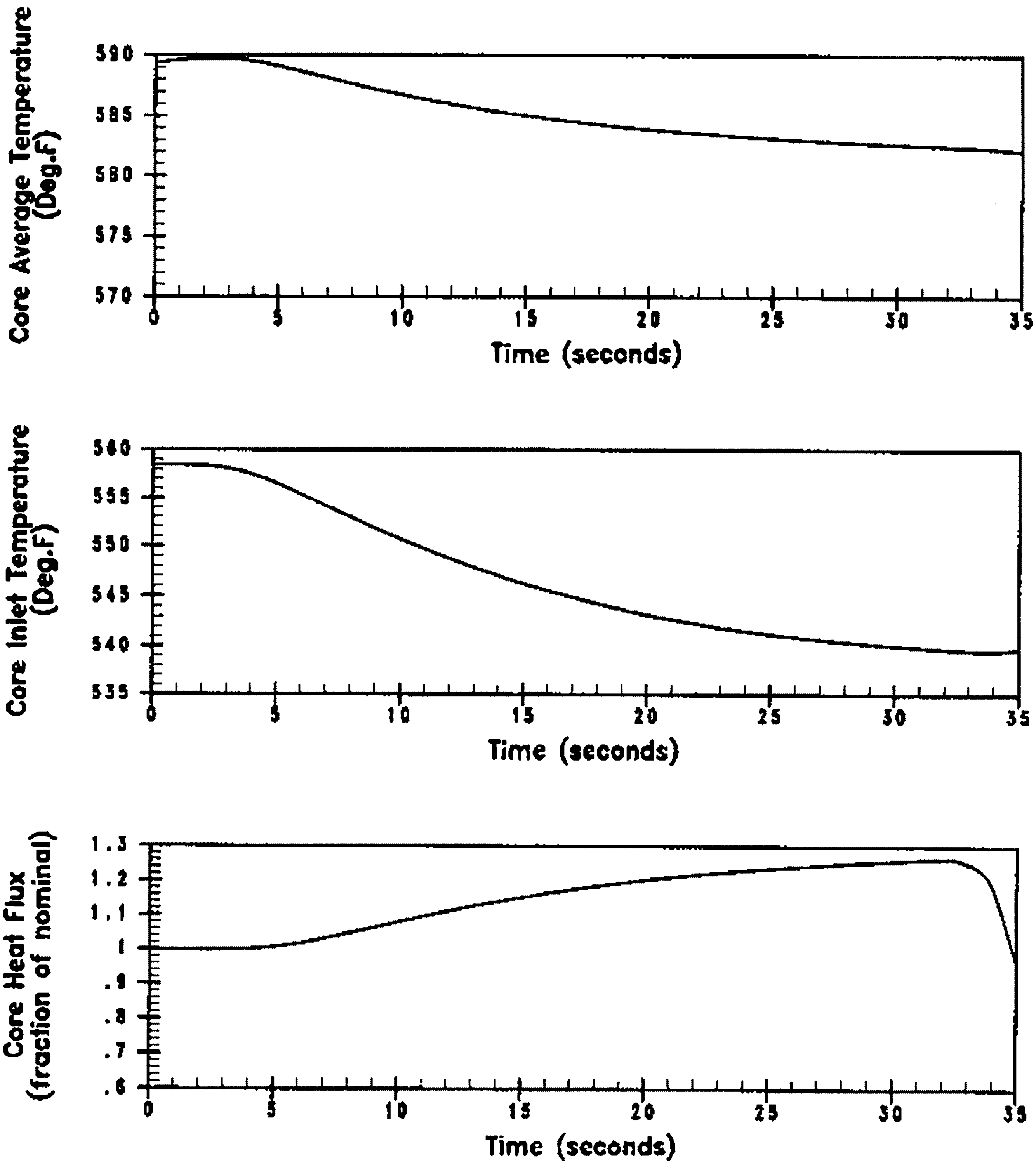
VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

NUCLEAR POWER, PRESSURIZER PRESSURE AND PRESSURIZER WATER VOLUME TRANSIENTS FOR 0.8 FT^2 STEAM LINE BREAK AT FULL POWER

FIGURE 15.1.5–9



(Historical)

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SOUTHERN COMPANY
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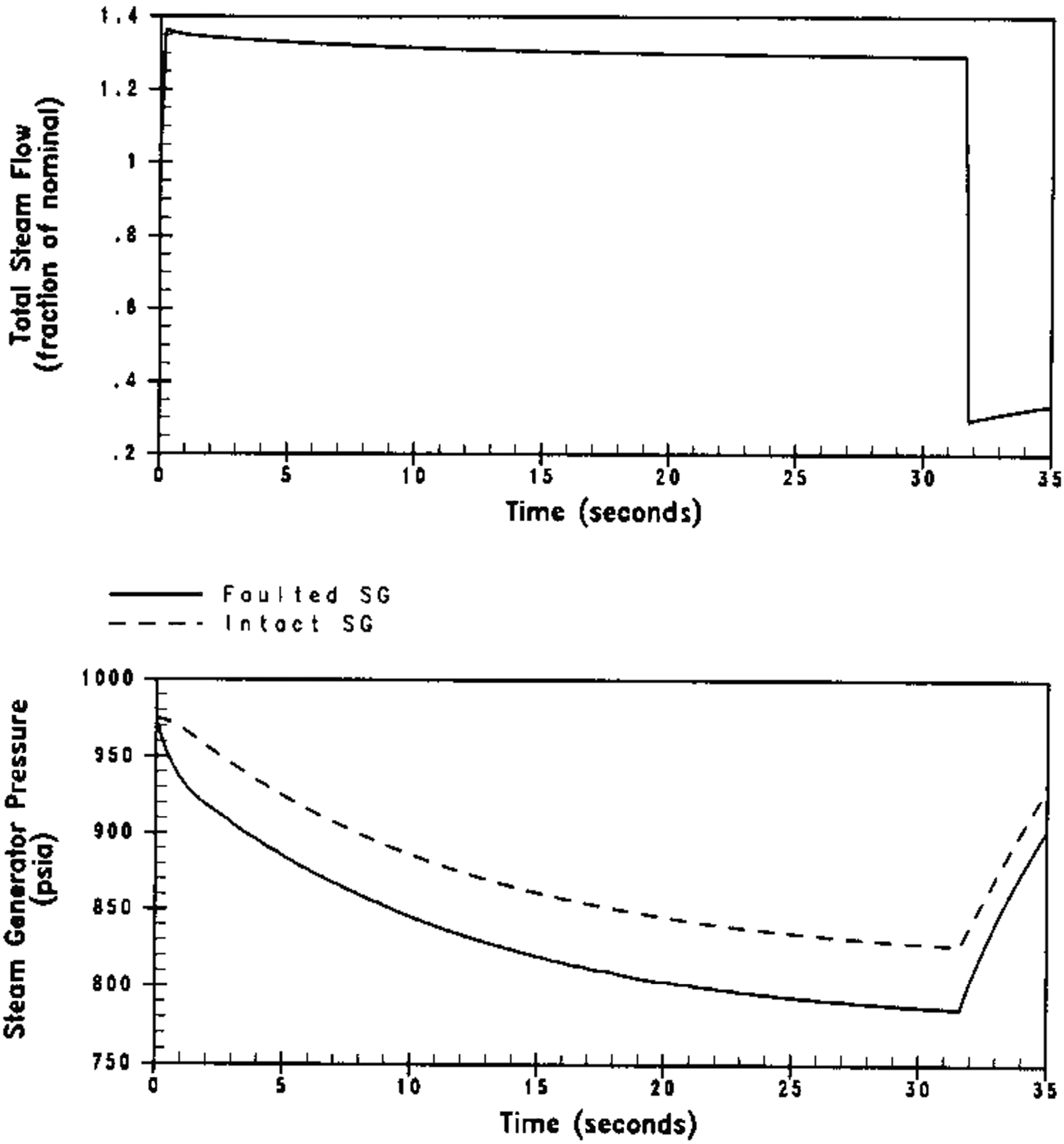
VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

CORE AVERAGE TEMPERATURE, CORE INLET TEMPERATURE, AND CORE HEAT FLUX TRANSIENTS FOR 0.8 ft^2 STEAM LINE BREAK AT FULL POWER

FIGURE 15.1.5–10



(Historical)

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SOUTHERN COMPANY
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VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

TOTAL STEAM FLOW AND STEAM GENERATOR PRESSURE TRANSIENTS FOR 0.8 FT^2 STEAM LINE BREAK AT FULL POWER

FIGURE 15.1.5–11