generators, and main steam piping. Ammonia, 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, ammonia, and an alternate amine provide these conditions. Although ammonia is volatile and will not concentrate in the steam generator, it will reach an equilibrium level which will establish alkaline conditions in the steam generator. 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.

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, Rev. 4.

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, Rev. 4.

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 ammonia, 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 low 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 continuous 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. Increased silica levels can be indicative of a condenser leakage or makeup water contamination. Grab samples are analyzed for silica by spectrophotometry.

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 electrode.

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 unit shift supervisor (USS), 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.

10.4.6.2.2 Component Description

10.4.6.2.2.1 Condensate Filter Demineralizer. The condensate filter demineralizers are the powdered resin 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 Ecodex 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 Powdex resins. The active surface area is 1281 ft^2 and the Powdex 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 (Powdex vessel), 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 Powdex resin in the vessel effluent.

After the Powdex 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 Powdex vessel, new resin is coated on the clean Powdex 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 Ecodex is flushed to the backwash recovery tank by backwash water.

After an exhausted Powdex 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 Powdex vessel until the backwash water quality is good enough for recycling to the turbine cycle. If necessary, the exhausted Powdex 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 Powdex 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 pre-selected value between 25 and 35 psi due to buildup of suspended solids.

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, vertially 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

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 System Description

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, alternate amine, and ammonia into the condensate pump discharge downstream of the condensate demineralizers and additional hydrazine and ammonia 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 ammonia, 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 ammonia 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)

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.

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 pressure transmitter is located in the main feedwater header to provide the feedwater system pressure to the speed control system for the steam generator feedwater pump turbines. 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, alternate amine, and ammonia into the condensate system. The ammonia 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.

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

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. 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

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. Therefore, a safety-related volume of 340,000 gal in either CST is sufficient to supply the auxiliary feedwater pumps. 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. In accordance with latest Westinghouse criteria, the total flow to the intact steam generators is less than 2500 gal/min for all cases.

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).

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, the available net positive suction head (NPSH) to the motor- and turbine-driven auxiliary feedwater pumps is 35 ft, while the required NPSH is 15 ft and 18 ft,

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. Ammonia Storage Tank

Stainless steel, 12,000-gal-capacity tank used for bulk storage of ammonia.

B. Ammonia Storage Transfer Pumps

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

C. Ammonia and Hydrazine Day Tanks

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

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

Duplex unit, positive displacement, disc diaphragm type pump, with autopneumatic 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.

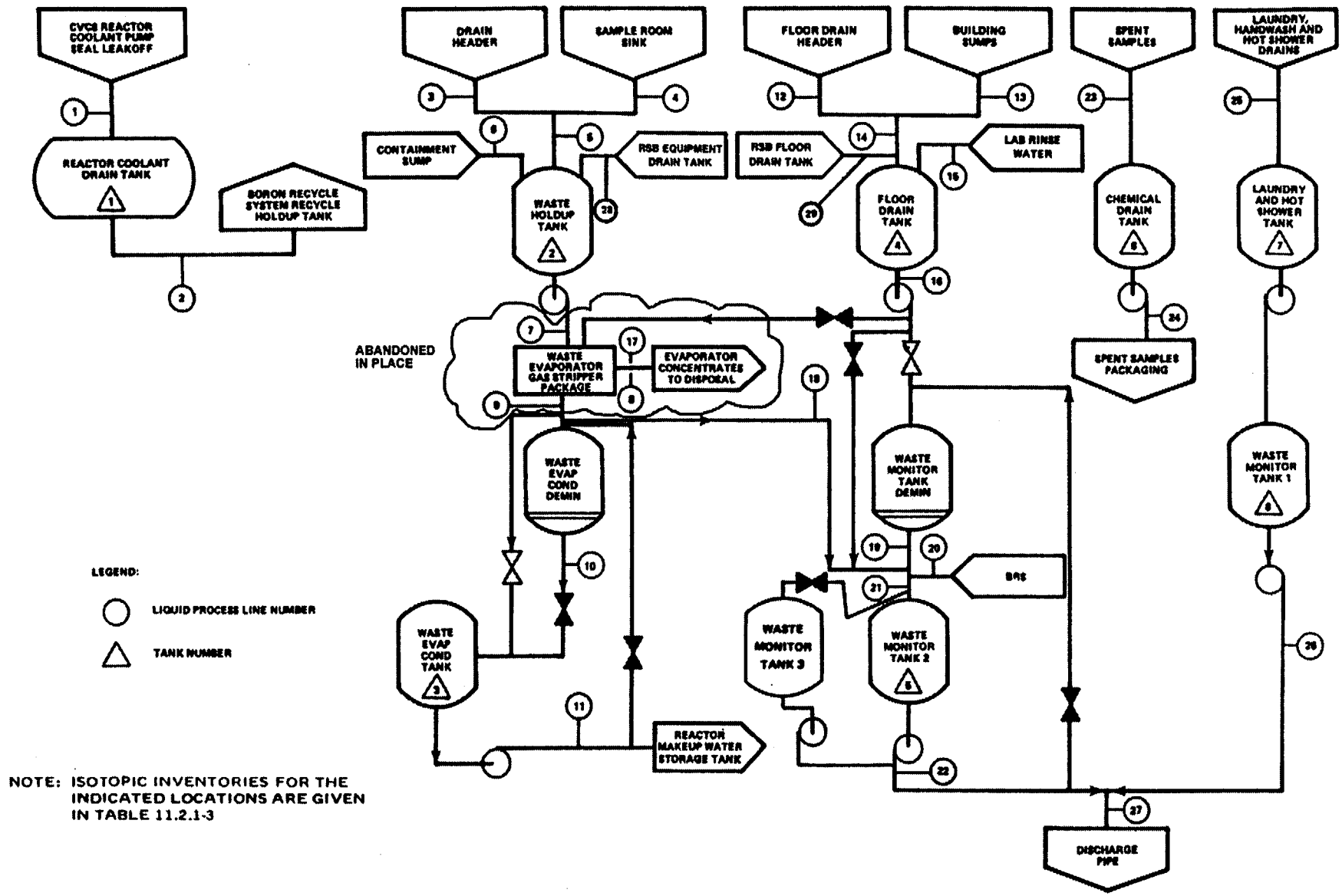
L. Alternate Amine Day Tank (Vendor Supplied Tote)

Stainless-steel, 400-gal tank with a positive displacement pump.

TABLE 10.4.10-1

CONDENSATE AND FEEDWATER CHEMICAL INJECTION SYSTEM

Major Component	Design Parameters
Ammonia storage tank	12,000 gal
Hydrazine storage tank	6000 gal
Day tanks	250 gal
Metering cylinders	10 liter
Ammonia storage transfer pump	43 gal/min at 178 ft head
Hydrazine storage transfer pump	43 gal/min at 110 ft head
Condensate chemical pump	12 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
Amine day tank with feed pump (Vendor Supplied Tote)	400 gal at 20 gal/day



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VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

LWPS PROCESS FLOW DIAGRAM

FIGURE 11.2.2-1

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 <u>System Operation</u>

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

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 Facilities

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.

Two buildings have been designed for handling dry active wastes (DAW) from plant operations:

A. The DAW processing building was designed to be used for sorting and compacting DAW. This building may also be used to store radioactively contaminated outage material and other radioactively contaminated items as needed.

B. The DAW storage building is used to store DAW. This building may also be used to store radioactively contaminated outage material and other radioactively contaminated items as needed.

11.4.2.5.1 Dry Active Waste Processing Building

DAW, outage equipment, and other radioactively contaminated equipment are transported to the processing facility 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 DAW processing building has separate areas for incoming radwaste containers, sorting and compaction of radioactive waste, handling of nonradioactive dry waste, and storage of empty containers. The DAW is sorted to separate usable material, radioactive waste, and nonradioactive waste. Radioactive waste is compacted into packaging that will comply with the criteria of 10 CFR part 71 to minimize the need for repackaging for shipment.

A sorting table and dry waste compactor are provided. The sorting table and trash compactor have designs that direct any airborne dusts created by the sorting or compaction operation through an exhaust fan and filter and then to the building ventilation system.

The building has lockable doors and is enclosed by a fence with lockable gates. The purpose of this fence is to prevent unauthorized personnel from entering a radioactive area. Security patrols verify building security.

The building is provided with automatic wet-pipe sprinkler systems designed for an Ordinary Group II Hazard. Ionization and photoelectric smoke detectors are provided. Smoke detectors actuation or sprinkler head water flow initiates local audible/visual alarms external to the building. A fire hose station with portable extinguisher is provided. The waste container materials used do not support combustion.

11.4.2.5.2 Dry Active Waste Storage Building

DAW, outage equipment, and other potentially contaminated radioactive material are transported to the storage facility 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. The waste is stored in the interior of the building. Lockable doors and a fence with lockable gates are provided.

The building is provided with automatic wet-pipe sprinkler system and smoke detectors. Fire alarms external to the building are provided. The waste container materials used do not support combustion.

This monitor has visual and audible alarms locally and remotely in the nonradioactive storage area. The remote alarm is located adjacent to the ventilation system controls. Upon alarm, the ventilation system exhaust can be manually isolated. This monitor shall be activated when DAW processing commences.

B. Liquid Process and Effluent Monitors

1. Deleted.

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 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) and the Comm Console PERMS viewer application in the control room.

   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 and

B) has two alarm setpoints; i.e., intermediate and high. ARE-13256, an inactive monitor, has only a high alarm. 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 are operator controlled through the IPC interface. 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 (ARB ARMS and N16 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 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.

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 nuclear plant general 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 review board 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. (See section 17.2.)

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 nuclear plant general 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 superintendent. The authority to prevent unsafe practices and to direct steps to prevent any unnecessary radiation exposures also lies within the health physics superintendent's responsibilities. The health physics supervisor will assist the health physics superintendent in ensuring that radiation exposures are maintained ALARA. The health physics supervisor will report to the health physics superintendent who serves as the radiation protection manager referred to in Regulatory Guide 8.8. The health physics and chemistry manager 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

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.

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 Area Radiation Monitoring

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 during processing of liquid radwaste. The design objectives of these ARMs are described in paragraph 12.3.4.1.10.

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, and AX6DD225).

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. 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, 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.

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. 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. Alternate Radwaste Building Area Monitor ARE-16851

To continuously indicate the radiation levels on the outside west wall of the ARB. A high alarm alerts personnel in the area of a deteriorating radiological condition.

J. Alternate Radwaste Building Area Monitor ARE-16852

To continuously indicate the radiation levels on the outside of the shield wall on the east side of the ARB. A high alarm alerts personnel in the area of a deteriorating radiological condition.

K. Alternate Radwaste Building Area Monitor ARE-16853

To continuously indicate the radiation levels in the interior of the ARB. A high alarm alerts personnel in the area of a deteriorating radiological condition.

L. Alternate Radwaste Building Area Monitor ARE-16854

To continuously indicate the radiation levels in the dress-out area of the ARB. 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, RE-0011, and ARE-16854, 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 (or alternate radwaste control room in the case of the ARB monitors) 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.

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 ARB Area Radiation Monitors

The design objectives of the ARB ARMs during normal liquid radwaste processing and anticipated operational occurrences are:

A. To warn of uncontrolled or inadvertent movement of radioactive material in and around the ARB.

B. To provide local and remote (in the ARB control room) indication (audible and visual) of ambient gamma radiation levels in and around the ARB.

C. To annunciate and warn of possible equipment malfunctions and radioactive leaks in the ARB.

D. To furnish information for making radiation surveys.

By meeting the above objectives, the ARMs 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.

12.4.2 RADIATION EXPOSURE OUTSIDE THE PLANT STRUCTURES

12.4.2.1 Direct Radiation Dose Estimates

The direct radiation from the containment and the auxiliary and turbine buildings is negligible compared to that from the alternate radwaste building and outside storage tanks. The principal sources of radioactivity not stored in the plant structures are the alternate radwaste building, 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 alternate radwaste building 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.

12.5 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 superintendent by the nuclear plant general manager, who has the overall radiological safety responsibility. Responsibilities of the health physics staff relative to radiation protection are given below.

The nuclear plant general manager will ensure that all station personnel support the health physics program. The nuclear plant general manager will participate, via review and approval, in the selection of goals and objectives for the health physics program. Through the nuclear plant general manager's review and controlling functions, support will be given to the efforts of the health physics and chemistry manager in achieving the objectives stated in paragraph 12.5.1.1.

The health physics superintendent has functional control of and is responsible for establishing the health physics program. The health physics superintendent 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 superintendent's qualifications are listed in subsection 13.1.3.

Additional health physics personnel titles include nuclear specialists, laboratory supervisors, health physics foremen, health physics technicians, dosimetry specialists, 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

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 performance team leaders or assistant performance 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 Nuclear Support (Vogtle) is responsible to GPC for assuring the availability of, for providing, or for securing adequate technical support for the VEGP. SNC Nuclear Support may use the services of Technical Services, SCS, Bechtel, Westinghouse, and others as appropriate. Portions of the technical support from these organizations may be fulfilled through the use of outside contractors.

13.1.1.1.3.1 Southern Company Services Support. Southern Company Services (SCS) 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. SCS and others provide for various assigned support activities, which normally include the following:

A. Architect-engineering services of SCS and others 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 safety evaluation and analysis.

C. Site assistance as required on safety evaluations and analyses which are not directly related to design; e.g., with respect to operational requirements, technical specification changes, etc.

13.1.1.1.3.2 Technical Services Support. Technical Services provides for various support activities, including the following:

A. Nuclear fuel procurement and contract management.

B. Nuclear fuel support, including core analysis and quality assurance.

C. Selected generic issues and projects.

D. Environmental, health physics, and plant chemistry support.

13.1.1.1.3.3 Nuclear, Mechanical, Structural, Electrical, Thermal-Hydraulic, Metallurgy and Materials, and Instrumentation and Control Engineering. Southern Company Services is the primary source of engineering in these disciplines.

13.1.1.1.3.4 Plant Chemistry. Plant chemistry support is the responsibility of Technical Services, as discussed in paragraph 13.1.1.1.3.2.

13.1.1.1.3.5 Health Physics. Health physics support is provided by Technical Services, as discussed in paragraph 13.1.1.1.3.2.

13.1.1.1.3.6 Engineering Support. Nuclear Engineering support (Vogtle) provides this support as discussed in paragraph 13.1.1.2.1.2.4.

13.1.1.1.3.7 Licensing Services Support. Nuclear Licensing Services support (Vogtle) provides this support, as discussed in paragraph 13.1.1.2.1.2.5.

13.1.1.1.3.8 Maintenance Support. Nuclear Maintenance and Support (Vogtle) provides this support, as discussed in paragraph 13.1.1.2.1.2.6.

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, which provide services in the areas of engineering, licensing, maintenance, administration, and quality assurance.

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Engineering support is provided primarily by the SCS Nuclear Plant Support (Vogtle). SNC Technical Services provides nuclear fuel contract administrative services, reload licensing, and operating licensing support.

The structures of the nuclear operations organization and the Vogtle organization are shown in figures 13.1.1-1 and 13.1.1-2 and are described in the following paragraphs. The Technical Services, Administrative Services and Human Resources organizations are also shown in figure 13.1.1-1 and are described in the following paragraphs.

13.1.1.2.1.1 President and CEO. President/CEO is responsible for all aspects of operation of Southern Company's nuclear plants including employment decisions. The president/CEO is also responsible for all technical and administrative support activities provided by SNC, SCS, and nonaffiliate contractors. The president/CEO directs the executive vice president. The president/CEO reports to the Board of Directors with respect to all matters.

13.1.1.2.1.2 Executive Vice President. The executive vice president reports to the president/CEO. This individual is responsible for the safe, reliable, and efficient operation of VEGP and Edwin I. Hatch Nuclear Plant (HNP). The executive vice president directs the vice president-project (Vogtle), the vice president-project (Hatch) and the vice president-project (Farley).

13.1.1.2.1.2.1 Vice President-Project (Vogtle). The vice president-project (Vogtle) reports to the executive vice president regarding operation issues and support matters. As vice president-project (Vogtle), he/she is responsible for operation and maintenance of VEGP. He/she directs the nuclear plant general manager, the nuclear support general manager (Vogtle), the safety audit and engineering review manager (Vogtle), and the administrative support supervisor.

13.1.1.2.1.2.2 Nuclear Support General Manager (Vogtle). The nuclear support general manager (Vogtle) reports to the vice president-project (Vogtle) and is responsible for corporate support in the areas of engineering, licensing, maintenance, planning and performance, and budgeting. The nuclear support general manager (Vogtle) directs the engineering manager (Vogtle), the licensing services manager (Vogtle), the nuclear maintenance and support manager (Vogtle), and the planning and performance supervisor (Vogtle).

13.1.1.2.1.2.3 Safety Audit and Engineering Review Manager (Vogtle). The responsibilities of the safety audit and engineering review manager (Vogtle) are described in section 17.2.

13.1.1.2.1.2.4 Engineering Manager (Vogtle). The engineering manager (Vogtle) reports to the nuclear support general manager (Vogtle). This individual is responsible for providing corporate support to VEGP in matters related to design control and engineering support for VEGP. Some specific responsibilities of the nuclear engineering corporate staff include:

A. Direct design engineering for plant modifications and provide project management for major improvement projects.

B. Assure that specialized engineering expertise is available as needed for normal operations and emergency situations.

C. Assess nuclear industry issues through contact with owners and ad hoc groups, and architect/engineer and nuclear utility interfaces. Review reports of industry experience for events at VEGP.

D. Assure that documentation and records of design activities are properly maintained.

13.1.1.2.1.2.5 Licensing Services Manager (Vogtle). The licensing services manager (Vogtle) reports to the nuclear support general manager (Vogtle). This individual is responsible for providing corporate support to VEGP in matters related to licensing, nuclear fuel, and emergency planning for VEGP. Some specific responsibilities of the licensing services corporate staff include:

A. Act as the primary interface with the Nuclear Regulatory Commission (NRC) for VEGP. Evaluate regulatory information and translate NRC requirements.

B. Ensure that specialized nuclear fuel expertise is available, as needed, including scheduling, procurement, conversion, enrichment, licensing, and core analysis.

C. Coordinate the planning and execution of emergency drills. Act as VEGP interface with federal, state, and local authorities.

D. Maintain the Vogtle Final Safety Analysis Report (FSAR), Technical Specifications, Emergency Plan, Security Plan, and other licensing documents.

13.1.1.2.1.2.6 Nuclear Maintenance and Support Manager (Vogtle). The nuclear maintenance and support manager (Vogtle) reports to the nuclear support general manager (Vogtle). This individual is responsible for providing corporate support to VEGP in matters related to operations, maintenance, testing, and inspection of VEGP. Some specific responsibilities of the nuclear maintenance and support corporate staff include:

A. Evaluating preventive and corrective maintenance programs and contracted maintenance services.

B. Assessing maintenance-related developments and industry trends for application or impact on VEGP.

C. Assisting in implementation of maintenance planning and scheduling processes. This includes outage management and long-term planning.

D. Coordinating inservice inspection and testing programs.

13.1.1.2.1.3 Vice President and Corporate Counsel. The vice president and corporate counsel reports to the president/CEO. This individual is responsible for the legal activities provided by SNC for the VEGP. The vice president and corporate counsel directs the corporate secretary.

13.1.1.2.1.3.1 Vice President-Technical Services. The vice president-technical services reports directly to the president/CEO and provides unique expertise in areas that perform best when functionally grouped and provides an in-focus response to generic regulatory or industry issues. Organizations contained within Technical Services are Nuclear Fuel; Inspection and Testing Services; Corporate Quality Services; and Regulatory, Engineering, and Environmental Services.

13.1.1.2.1.3.2 Vice President-Administrative Services. The vice president-administrative services reports directly to the president/CEO. The vice president-administrative services is responsible for providing corporate administrative services including financial services, public affairs, corporate security, nuclear operations related support functions (e.g., access authorization and fitness for duty), material services and general services. Reporting directly to the vice president-administrative services are the treasurer, comptroller, and chief

financial officer; manager-material services; manager-safety and health; supervisor-general services; manager-public affairs; and manager-security.

13.1.1.2.2 Quality Assurance

The safety audit and engineering review manager (SAERM) is responsible functionally to the vice president-project (Vogtle). The SAERM is authorized to manage the QA program for design, testing, operation, and maintenance and to ensure its implementation in accordance with the requirements of section 17.2. The SAER organization is composed of a staff at the corporate headquarters and a staff at the VEGP plant site.

The SAER organization does not provide technical support (as defined in subsection 13.1.1 of Regulatory Guide 1.70) for the operation of VEGP. The activities of the SAER organization are fully described in section 17.2.

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, which provides the line responsibility for the operation of the VEGP, and the support organization is shown in figure 13.1.1-1. 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 Vogtle Project 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.

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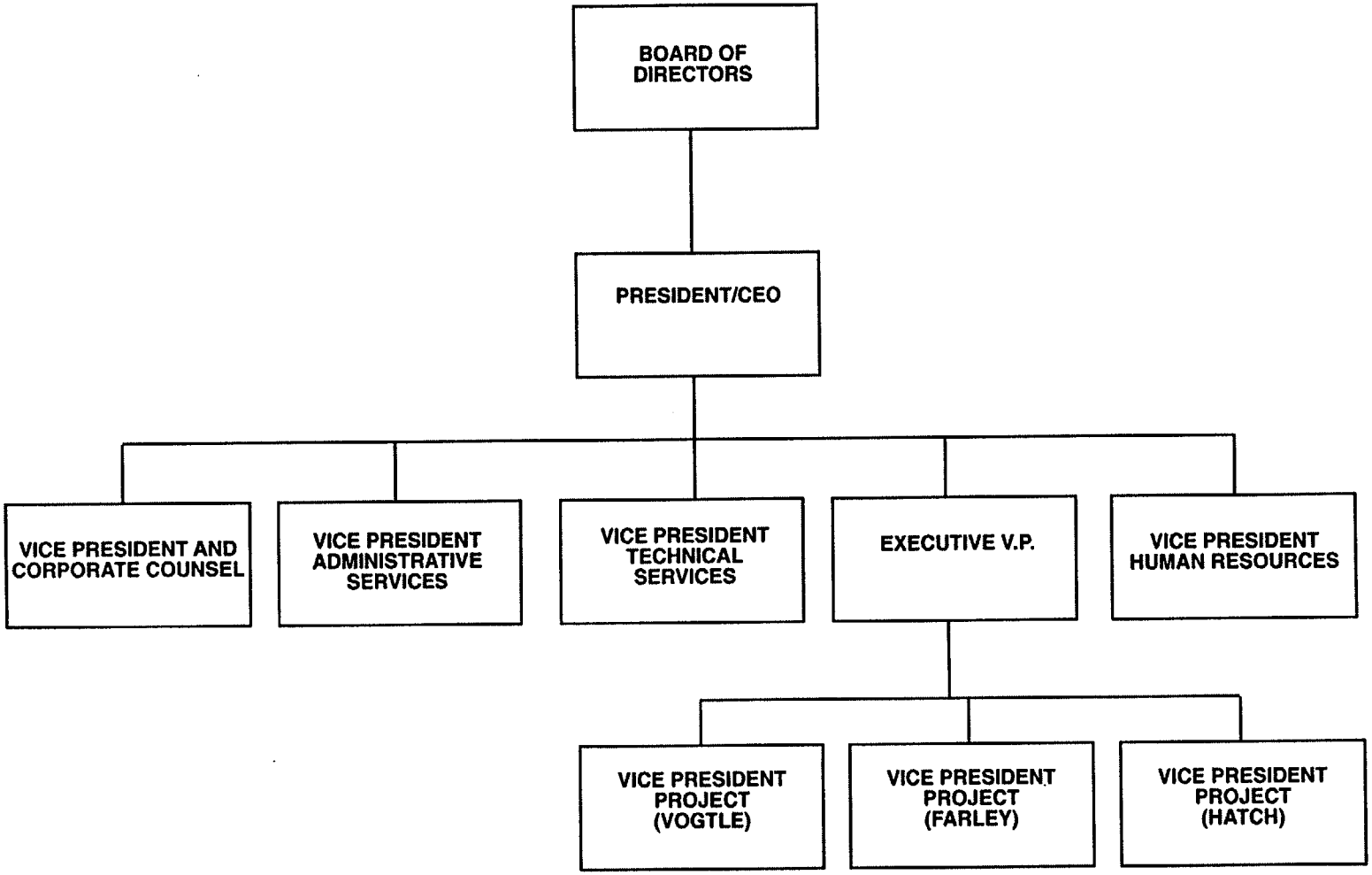
The operating organization for the VEGP is described in subsection 13.1.2. The company technical support organizations for operation, modification, and maintenance are described in paragraph 13.1.1.2 and shown in figures 13.1.1-1 and 13.1.1-2. The operating and support organization described herein provide assurance of safe operation of VEGP Units 1 and 2 in compliance with regulatory requirements.



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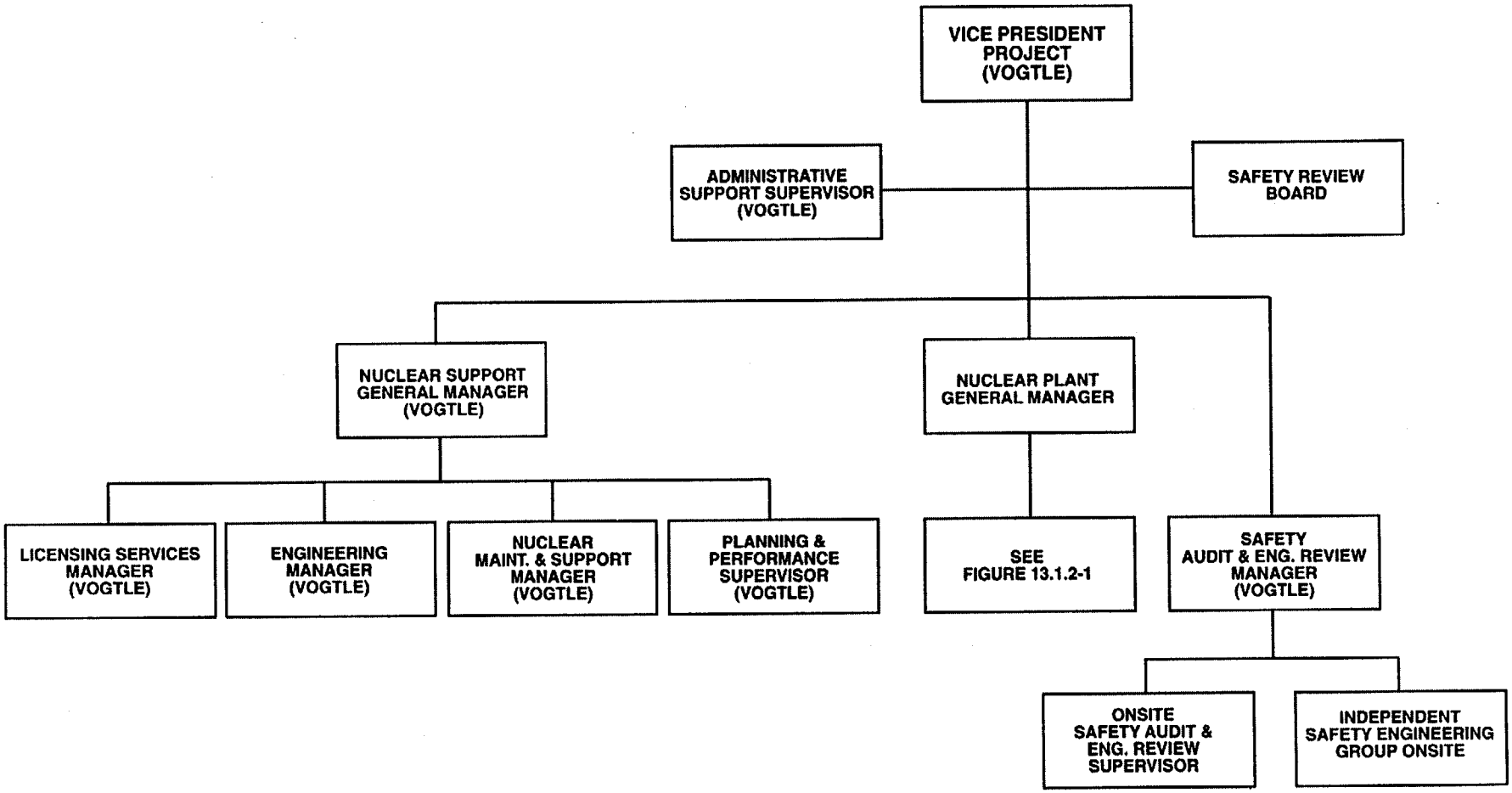
VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

NUCLEAR OPERATIONS ORGANIZATION
VEGP UNITS 1 AND 2

FIGURE 13.1.1-1



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VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

SNC TECHNICAL SUPPORT ORGANIZATION
VEGP UNITS 1 AND 2

FIGURE 13.1.1-2

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 shown in figure 13.1.2-1.

13.1.2.2 Plant Personnel Responsibilities and Authorities

13.1.2.2.1 Overall Plant Management

The nuclear plant general manager (NPGM) is responsible for direct management of the plant, including industrial relations, planning, coordination, direction of operation, training, maintenance, refueling, and technical activities. The NPGM is responsible for:

- Compliance with the requirements of the operating license, Technical Specifications, and quality assurance program, and
- Approval, prior to implementation, of each proposed test, experiment, or modification to systems or equipment that affect nuclear safety.

In the NPGM's absence, the plant support assistant general manager assumes responsibility for the plant. The NPGM will designate in writing other qualified personnel to assume overall plant responsibility in his absence. (See paragraph 13.1.2.2.2 for succession of responsibility for overall plant operation.)

The NPGM reports to the vice president-project (Vogtle). The NPGM has access to the advice and services of technical specialists in the corporate office and outside expertise as necessary.

The organizations reporting to the NPGM are:

A. The plant operations assistant general manager is responsible for the operation of Units 1 and 2. Reporting to the plant operations assistant general manager are the operations manager, the maintenance manager, the outage and planning manager, and the health physics and chemistry manager.

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B. The plant support assistant general manager is responsible for supporting Units 1 and 2. Reporting to this plant support assistant general manager are the engineering support manager, the plant training and emergency preparedness manager, the administration manager, and the plant modifications and maintenance support manager.

Responsibilities of the individuals reporting to the plant operations assistant general manager are as follows:

C. The operations manager is responsible for operating the plant safely and efficiently.

D. The maintenance manager is responsible for directing the performance teams, for performance of preventive maintenance, for repair of plant equipment, and for assigned quality control measures.

E. The health physics and chemistry manager is responsible for the radiation protection program and for chemistry and radiochemical activities at VEGP, and for ensuring the building and grounds upkeep.

F. The outages and planning manager is responsible for planning, scheduling, and coordinating outages, and for developing plans, which include both corrective and planned maintenance and technical specification surveillance.

Responsibilities of the individuals reporting to the plant support assistant general manager are as follows:

G. The engineering support manager is responsible for systems engineering, discipline engineering, design changes and modifications, nuclear safety and compliance, reactor engineering, plant performance, and implementation of the quality control inspection program.

H. The plant training and emergency preparedness manager is responsible for the training and retraining programs, the training simulator, and the managing of the onsite emergency preparedness effort.

I. The plant administration manager is responsible for overseeing plant security, administration and managing the computerized information system.

J. The plant modifications and maintenance support manager is responsible for providing direction to other plant departments in technical matters and plant design. The

plant modifications and maintenance support manager is also responsible for directing the improvement projects team, the plant modification program, and providing maintenance support utilizing contractor craft personnel.

13.1.2.2.2 Operations Supervision and Shift Organization

The operations manager is responsible to plant management for the operation of Unit 1 and Unit 2. The operations manager shall possess a senior reactor operator (SRO) license. Reporting to the operations manager are the unit superintendents and shift superintendents. The unit superintendents direct day-to-day planning, provide technical support, and coordinate training of operations personnel.

The shift superintendent is responsible for seeing that plant operations are conducted in accordance with appropriate standing orders, unit operating procedures, and technical specifications. The shift superintendent'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 superintendent shall possess an SRO license.

A shift technical advisor reports to the shift superintendent. The shift technical advisor provides perspectives in assessment of plant conditions and evaluation of the safety of the plant.

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 superintendent, 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 superintendent. 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.

Reporting to the shift supervisors are the shift support supervisors, plant operators, and plant equipment 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 radiochemical 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. Nuclear plant general manager (NPGM).

B. Plant operations assistant general manager.

C. Plant support assistant general manager.

D. Operations manager.

E. Licensed manager/superintendent designated by NPGM.

F. Shift superintendent (SRO).

G. Shift supervisor (SRO).

13.1.2.2.3 Engineering, Health Physics, Chemistry, and Maintenance Supervision

The plant modifications and maintenance support manager, and the engineering support manager report to the plant support assistant general manager and supervise the plant modifications and maintenance support and engineering support staffs. The functions of their staffs are to monitor and enhance plant performance, ensure regulatory compliance, provide engineering support for plant operation, maintenance support, and implement the plant modification program and the quality control program.

The nuclear safety and compliance (NSAC) supervisor reports to the engineering manager. The nuclear safety and compliance supervisor is involved in day-to-day safety-related activities such as maintaining an NRC open item list from inspections and ensures these items are resolved.

Plant engineering supervisors reporting to the plant modifications and maintenance support manager and engineering support manager determine when to call consultants and contractors for dealing with complex problems beyond the scope of available capabilities. The supervisors correspond closely to that identified as "engineer in charge" by ANSI 18.1-1971.

The health physics-superintendent reports to the health physics and chemistry manager and is responsible for the radiation protection program. The health physics-superintendent also ensures that appropriate monitoring devices and protective clothing are available and is responsible for radiation monitoring devices used by personnel entering the plant and for the maintenance of all required radiation exposure records of plant support and visiting personnel. Reporting to the health physics superintendent is the health physics staff.

The chemistry superintendent reports to the health physics and chemistry manager and is responsible for chemical and radiochemical activities at the plant. The chemistry superintendent is responsible for administrative control of effluent releases from the plant to ensure that the releases are maintained ALARA and within the required limits.

The chemistry supervisors work under the direction of the chemistry superintendent and are responsible for performing chemical and radiochemical sample analyses, and for monitoring and maintaining the plant makeup water demineralizer to ensure proper water quality. Laboratory personnel advise operations staff on the operation of plant systems to maintain water quality within the specified limits.

The maintenance manager directs and plans maintenance activities with the assistance of other departments. Reporting to him are an assistant performance team manager and performance team leaders. The performance team leaders and assistant performance team leaders direct the performance team technical staff, electricians, I&C technicians, and mechanics in the upkeep of equipment. The performance team leaders and assistant performance team leaders are responsible for directing technicians in testing, calibration, surveillance checks, and repair of plant instrumentation and control systems.

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13.1.2.2.4 Quality Assurance and Quality Supervision

The NPGM is responsible for the development and implementation of the quality assurance program to comply with the OQAM with the exception of the controls assigned to the safety audit and engineering review (SAER) organization (chapter 17). Reporting to the engineering support manager, the designated engineering supervisor will provide for inspection activities required by the quality assurance program. Personnel performing quality control inspections have procedural authority to stop work and to control further processing, use, or installation of nonconforming items.

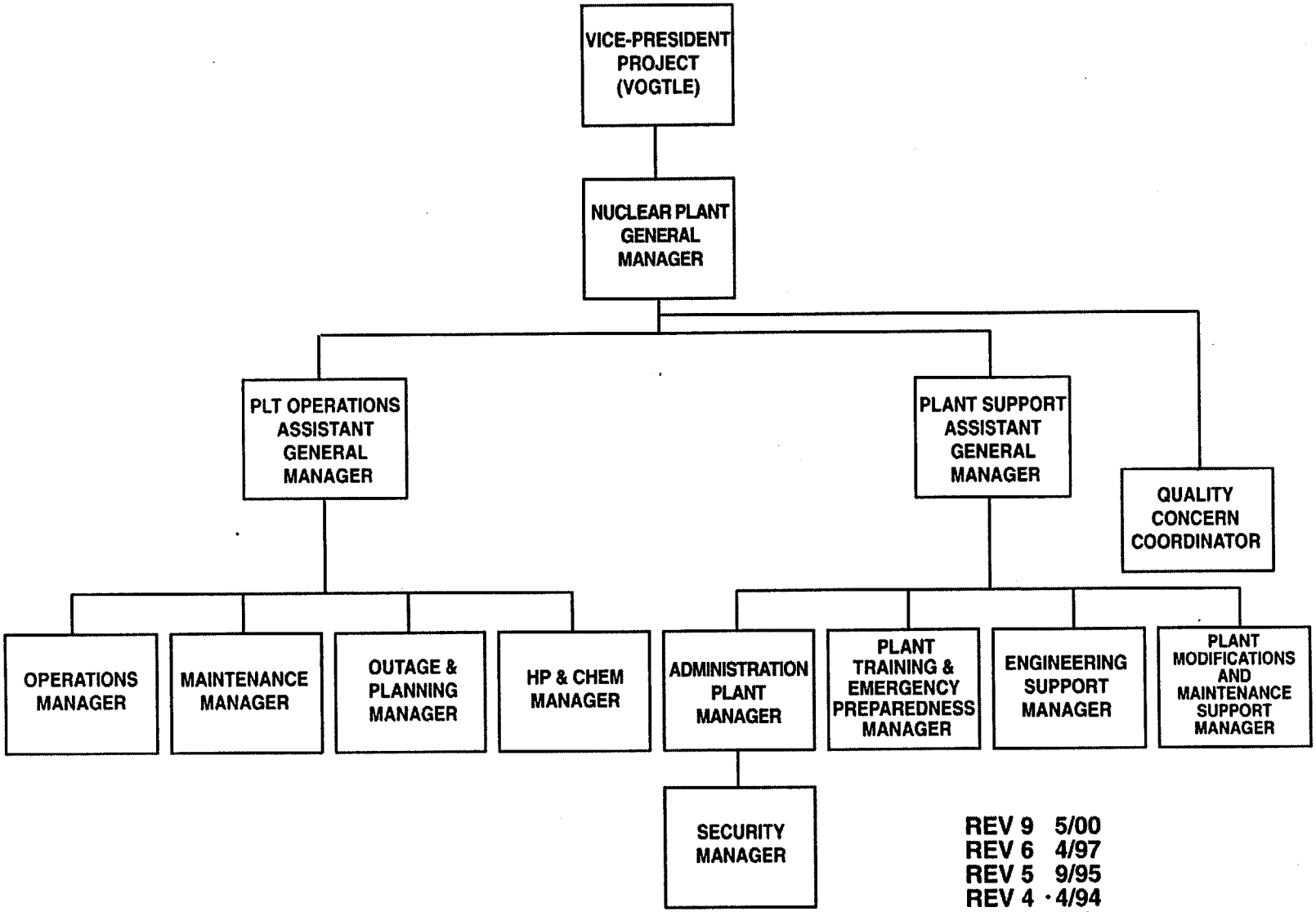
The safety audit and engineering review manager (SAERM), as described in section 17.2 of the FSAR, is responsible to the vice president-project (Vogtle) for managing activities of the SAER organization. The SAER organization will provide a comprehensive independent audit of safety-related activities to verify that they are in compliance with the quality assurance program. The quality assurance program during operations is discussed in section 17.2.

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 superintendent is responsible for implementing the radiation protection program in the absence of the health physics superintendent or his designated alternate.



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NUCLEAR OPERATIONS ORGANIZATION
VEGP UNITS 1 AND 2

FIGURE 13.1.2-1

TABLE 13.1.3-1 (SHEET 1 OF 2)

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	Nuclear plant general manager	10	3				X[a]	4		4	
							X[a]	4		4	
Assistant plant manager	Plant operations/plant support assistant general manager	10	3				X[a]	4		4	
Operation manager	Operations manager	8	3				X	2 or	2	2	
Technical manager	Engineering support manager		1	7				4		4	
Maintenance manager	Maintenance manager	7	1					2 or	2	2	
Training manager(b)	Plant training and emergency preparedness manager		2	4				4			
Superintendents or supervisors not requiring NRC license											
Radiochemistry	Chemistry superintendent or plant chemist		5					2 or	2	4	1 year experience in radiochemistry.
Other supervisors	Material supervisor			4							High school diploma or equivalent required.
	Laboratory supervisor			4							High school diploma or equivalent required.
	Performance team leader			7							Knowledge of area of responsibility and high school diploma or equivalent required.

TABLE 13.1.3-1 (SHEET 2 OF 2)

ANSI N18.1 Title	VEGP Title	Recommended Experience (years)				License		Suggested Education (Years)			Remarks
		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	
Foreman	Laboratory foreman			4							High school diploma or equivalent required.
	I&C assistant performance team leader			5				2 or	2	4	6 months experience in nuclear I&C.
	Assistant performance team leader			4							Knowledge of area of responsibility and high school diploma or equivalent required.
Technical support personnel											
Engineer-in-charge	Plant engineering supervisor			3	4						
ANSI 3.1 Title											
Radiation protection manager	Health physics and chemistry manager or health physics/chemistry superintendent or plant health physicist		3[c]	4				4			

a. Senior reactor operator license is not required for the nuclear plant general manager or the plant operations/plant support assistant general manager, but the nuclear plant general manager or either of the plant operations/plant support assistant general managers will have the background required to sit for examination.

b. Required by NUREG-0737.

c. Either the health physics/chemistry manager or health physics/chemistry superintendent or 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.

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

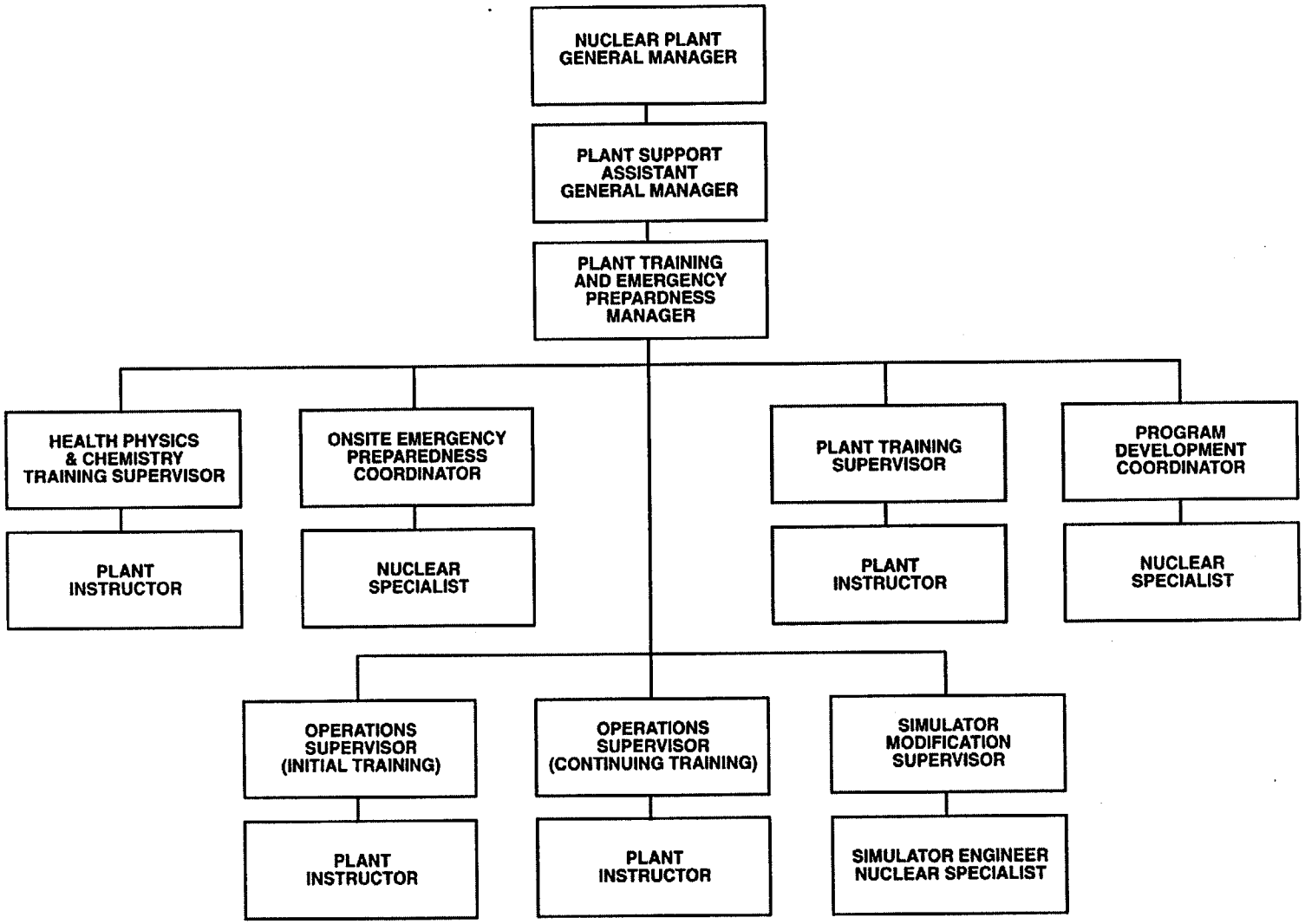
Records to document course completion and documentation of qualification are retained as quality records in accordance with Regulatory Guide 1.88, subject to the conditions specified in paragraph 1.9.88.1 and in accordance with paragraph 17.2.17.1, Records Retention.

13.2.1.2 Simulator

The VEGP simulator conforms to the guidance given in Regulatory Guide 1.149, Revision 1. The certification of the VEGP simulator as a plant-referenced simulator 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. Instructors are qualified and participate in requalification training in accordance with the processes delineated for accredited programs. In addition, the fire protection training instructor also meets the requirements outlined in paragraph 9B.C.3.d(2). The VEGP training department organization is shown in figure 13.2.1-1.



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VEGP TRAINING ORGANIZATION

FIGURE 13.2.1-1

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

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 ANSI N45.2.6-1978 as described in subsection 1.9.58 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. QC inspection personnel and the engineering supervisor responsible for inspection activites must receive additional training in plant systems. QC inspection personnel will complete the plant systems training within 1 year of assignment to their position and the engineering supervisor must complete this training prior to being qualified.

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 intolicensed personnel training, senior reactor operator certified personnel training, and shift technical advisor training. Other personnel, including the superintendents in health physics and chemistry, and team leaders and assistant team leaders for I&C technicians in the maintenance department, as well as the nuclear plant general manager and the plant operations assistant general manager, complete training in mitigating core damage commensurate with their responsibilities.

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 following the recommendations of Regulatory Guide 1.33, Rev.2 1978, Quality Assurance Program Requirements (Operations), and ANSI N18.7-1976, Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants.

The nuclear plant general manager (NPGM) 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 NPGM 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 NPGM approval as described above, this program is also implemented through standing committees, as referenced below.

13.4.1 ONSITE REVIEW

Refer to paragraph 17.2.1.2.1 for discussion related to the Plant Review Board (PRB), which performs the onsite review function.

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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 control 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 as described in Appendix A of Regulatory Guide 1.33, Rev. 2.

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 affects an unreviewed safety question. The acceptability of procedure reviews is assured by the review and approval by the responsible VEGP department heads. Nuclear

Operations department heads shall further ensure that procedures described in paragraph 17.2.1.2.1 are forwarded to the Plant Review Board (PRB) for additional review.

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 Regulatory Guide 1.58. 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 nuclear plant general manager (NPGM) has ultimate responsibility for all plant procedures. Specific procedures for which NPGM approval is required are discussed in paragraph 17.2.5.1.2. Nuclear Operations 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 paragraph 17.2.5.1.2.

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 SAER group performs various audits (described in section 17.2) 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.

operation during plant transients or emergency conditions. Other provisions establish the shift superintendent 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 superintendent 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 NPGM annually reviews the administrative duties of the shift superintendent 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 guidance provided in NUREG-0737, item I.A.1.3, as revised by NRC Generic Letter 82-12. These procedures limit the working hours of the plant staff in performance of safety-related functions; e.g., licensed senior reactor operators, licensed reactor operators, key health physics technicians, key auxiliary operators, and key maintenance personnel. These procedures provide for maintaining adequate shift coverage without routine heavy use of overtime. The objective is to have operating personnel work a nominal 40-h week while the plant is operating, although this work week may be adjusted to accomodate a 12-h shift schedule. However, in the event that unforeseen problems require substantial amounts of overtime, or during extended periods of shutdown for refueling, major maintenance or major plant modification, (on a temporary basis) these procedures shall provide the following guidelines:

- An individual should not be permitted to work more than 16 h straight, excluding shift turnover time;
- An individual should not be permitted to work more than 16 h in any 24-h period, nor more than 24 h in any 48-h period, nor more than 72 h in any 7-day period, all excluding shift turnover time.
- A break of at least 8 h should be allowed between work periods, including shift turnover time.
- Except during extended shutdown periods, the use of overtime should be considered on an individual basis and not for the entire staff on a shift.

These procedures require that use of overtime that would exceed the above-stated guidelines be authorized by the applicable department superintendent or higher level of management, with the basis for granting the

exception being documented. Controls are included in these procedures such that individual excess overtime shall be reviewed monthly by the NPGM or his designee.

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 ANSI 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.

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 17.2.17.1.3.o. 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 nuclear plant general manager.

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.

LIST OF FIGURES (Continued)

15.4.2-6 Uncontrolled RCCA Bank Withdrawal from Full Power with Minimum Reactivity Feedback (2 pcm/s Withdrawal Rate)

15.4.2-7 Minimum DNBR vs. Reactivity Insertion Rate for Rod Withdrawal from 100 Percent Power

15.4.2-8 Minimum DNBR vs. Reactivity Insertion Rate for Rod Withdrawal from 60 Percent Power

15.4.2-9 Minimum DNBR vs. Reactivity Insertion Rate for Rod Withdrawal from 10 Percent Power

15.4.3-1 Nuclear Power Transient and Core Heat Flux Transient for Dropped RCCA

15.4.3-2 Pressurizer Pressure Transient and Core Average Temperature Transient for Dropped RCCA

15.4.4-1 Improper Startup of an Inactive Reactor Coolant Pump

15.4.4-2 Improper Startup of an Inactive Reactor Coolant Pump

15.4.4-3 Improper Startup of an Inactive Reactor Coolant Loop

15.4.4-4 Improper Startup of an Inactive Reactor Coolant Pump

15.4.4-5 Improper Startup of an Inactive Reactor Coolant Pump

15.4.7-1 Representative % Change in Local Assy. Avg. Power for Interchange Between Region 1 and Region 3 Assy.

15.4.7-2 Representative % Change in Local Assy. Avg. Power for Interchange Between Region 1 and Region 2 Assy. with BP Rods Retained by the Region 2 Assy.

15.4.7-3 Representative % Change in Local Assy. Avg. Power for Interchange Between Region 1 and Region 2 Assy. with the BP Rods Transferred to Region 1 Assy.

15.4.7-4 Representative % Change in Local Assy. Avg. Power for Enrichment Error (Region 2 Assy. Loaded into Core Central Position)

15.4.7-5 Representative % Change in Local Assy. Avg. Power for Loading Region 2 Assy. into Region 1 Position Near Core Periphery

LIST OF FIGURES (Continued)

15.4.8-1 Nuclear Power Transient BOL Full Power

15.4.8-2 Hot Spot Fuel and Clad Temperature vs. Time BOL Full Power

15.4.8-3 Nuclear Power Transient EOL Zero Power

15.4.8-4 Hot Spot Fuel and Clad Temperature vs. Time EOL Zero Power

15.4.9-1 Steam Line Break Coincident with Control Rod Withdrawal: Steam Flow and Steam Pressure

15.4.9-2 Steam Line Break Coincident with Control Rod Withdrawal: RCS Pressure and Core Average Temperature

15.4.9-3 Steam Line Break Coincident with Control Rod Withdrawal: Core Heat Flux and Reactivity

15.5.1-1 Inadvertent Operation of ECCS at Power

15.5.1-2 Inadvertent Operation of ECCS During Power Operation

15.5.1-3 Inadvertent Operation of ECCS During Power Operation

15.6.1-1 Nuclear Power and DNBR Transients for Inadvertent Opening of a Pressurizer Safety Valve

15.6.1-2 Pressurizer Pressure Transients and Core Average Temperature Transient for Inadvertent Opening of a Pressurizer Safety Valve

15.6.3-1 Pressurizer Level

15.6.3-2 RCS Pressure

15.6.3-3 Secondary Pressure

15.6.3-4 Intact Loop T_{hot} and T_{cold}

15.6.3-5 Ruptured Loop T_{hot} and T_{cold}

15.6.3-6 Differential Pressure

LIST OF FIGURES (Continued)

15.6.3-7 Primary to Secondary Break Flow

15.6.3-8 Ruptured SG Water Volume

15.6.3-9 Ruptured SG Water Mass

15.6.3-10 Ruptured SG Atmospheric Mass Releases

15.6.3-11 Intact SG Atmosopheric Mass Release

15.6.3-12 Iodine Transport Model

15.6.3-13 Break Flow Flashing Fraction

15.6.5-1 Sequence of Events for Large Break LOCA Analysis

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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 (Also refer to subsection 15.4.1)	0.50	-	Lower (see figure 15.0.4-1)	0 and 3579[a,f]
	Excessive increase in secondary steam flow	LOFTRAN	0.0 and 0.50	-	Lower and upper (See figure 15.0.4-1)	3579[a,f]
	Inadvertent opening of a steam generator relief or safety valve	LOFTRAN	Function of moderator density (see figure 15.1.4-1)	-	-2.9 pcm/°F	0 (subcritical)
	Steam system piping failure	LOFTRAN, THINC	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 3579[a,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)	3579[a,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)	3657[c,d,e]
	Loss of normal feedwater flow	LOFTRAN	Lower figure 15.0.4-2	+7	Upper (see figure 15.0.4-1)	3657[c,d,e]
	Feedwater system pipe break	LOFTRAN	Lower figure 15.0.4-2	-	Upper (see figure 15.0.4-1)	3657[c,d,e]

TABLE 15.0.3-2 (SHEET 3 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)
	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	3585[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)	3565[b,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)	3585[c,f]
	Steam generator tube failure	LOFTTR2	Figure 15.0.4-2	-	Upper (see figure 15.0.4-1)	3650.6[a,d]
	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 subsection 15.6.5 references[b]	-	See subsection 15.6.5 references	3636[b,d]

a. Minimum pump heat of 14 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.

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.

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	2.5[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. From time setpoint is reached to turbine trip.

c. From time setpoint is reached to feedwater isolation.

d. Low-low level setpoint assumed for 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.

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15.0.9 FISSION PRODUCT INVENTORIES

15.0.9.1 Inventory in the Core

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 stated in TID-14844[1]: 100 percent of the noble gases and 50 percent of the halogens at a core power level of 3565 MWt.

The time dependent fission product inventories in the reactor core are calculated by the ORIGEN code[2] using a data library based on ENDF/B-IV[3,4]. Core inventories are shown in table 15.0.9-1. Table 15.0.9-1 is based on discharge region average burnups of 36,000 MWd/MTU. Calculations performed for the VANTAGE 5 fuel design[5] considered fuel enrichments of 4.5 wt/% and lead rod average burnups of 60,000 MWd/MTU (discharge batch average burnups of 45,000 MWd/MTU). The VANTAGE 5 analysis calculations showed the variations in core inventories to be insignificant. This finding is consistent with the current NRC position regarding extended fuel burnups[6,7]. Generic calculations performed for the VANTAGE+ fuel design[8] considered fuel enrichments up to 5.0 wt percent U-235, discharge batch average burnups of 54,000 MWd/MTU, and lead rod burnups up to 75,000 MWd/MTU. As was the case with the VANTAGE 5 analysis, the VANTAGE+ analysis also showed that variation in core inventories, as compared to values based on 36,000 MWd/MTU, were insignificant. The NRC approved this conclusion with regard to operating and storing fuel, setting limits for the storage of fuel with enrichments up to a maximum of 5.0 wt percent U-235[9] in the Unit 1 and Unit 2 spent fuel racks. Thus, the use of table 15.0.9-1 core inventories in calculating the radiological consequences of accidents remains valid for LOPAR, VANTAGE 5, and VANTAGE+ fuel assembly designs to 60,000 MWd/MTU lead rod average burnups.

15.0.9.2 Inventory in the Fuel Pellet Clad Gap

The radiation sources associated with a gap activity release accident are based on the assumption that the fission products in the space between the fuel pellets and the cladding of all fuel rods in the core are released as a result of cladding failure.

The gap activities were determined using the model suggested in Regulatory Guide 1.25. Specifically, 10 percent of the iodine and noble gas activity (except Kr-85, I-127, and I-129, which are 30 percent) is accumulated in the fuel clad gap. The gap activities are shown in table 15.0.9-1. With the exception of the fuel handling accident, these gap fractions were used in

calculating radiological consequences of the accident analyses. The gap fractions assumed for the handling accident are discussed in subsection 15.7.4.

REFERENCES

1. DiNunno, J. J., et al., "Calculation of Distance Factors for Power and Test Reactor Sites," TID-14844, March 1962.

2. Bell, M. J., "ORIGEN - The ORNL Isotope Generation and Depletion Code," ORNL-4628, May 1973.

3. "ORIGEN Yields and Cross Sections - Nuclear Transmutation and Decay Data From ENDF/B-IV," Radiation Shielding Information Center, Oak Ridge National Laboratory, RSIC-DLC-38, September 1975.

4. England, T. R., and Schauter, R. E., "ENDF/B-IV Fission-Product Files: Summary of Major Nuclide Data," ENDF-223, October 1975.

5. Westinghouse Letter 90GP*-G-0099, "Vogtle 1 and 2 Final VANTAGE 5 RTSR and Licensing Submittal," October 2, 1990.

6. Federal Register/Vol. 53. No. 39/Monday, February 29, 1988/pages 6040-6043.

7. Baker, D. A., et al., "Assessment of the Use of Extended Burnup Fuel in Light Water Power Reactors," NUREG/CR-5009, February 1988.

8. Davidson, S. L., and Ryan, T. L., "VANTAGE+ Fuel Assembly Reference Core Report," WCAP-12610-P-A (proprietary), April 1995.

9. C. K. McCoy to NRC, "Vogtle Electric Generating Plant Response to Request for Additional Information Revised Request to Revise Technical Specifications Credit for Boron and Enrichment Increase for Fuel Storage," LCV-0849-E, August 8, 1997.

BIBLIOGRAPHY

Meek, M. E. and Rider, B. F., "Summary of Fission Product Yields for U-235, U-238, Pu-239, and Pu-241 at Thermal Fission Spectrum and 14 Mev Neutron Energies," APED-5398, March 1968.

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 stablized 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.

malfunction occurs which results in a step increase in flow to one steam generator from zero to 225 percent of the nominal full load value for one steam generator.

C. For the feedwater regulating valve accident at no-load conditions (Mode 3), a feedwater regulating valve malfunction occurs which results in a step increase in flow to one steam generator from zero to 50 percent of the nominal full load value for one steam generator. The flow is limited by a 6-inch pipe in the line from the condensate pump to the steam generators through which all the feedwater to the steam generators must pass.

D. For the no-load condition, feedwater temperature is at a conservatively low value of 32°F.

E. No credit is taken for the heat capacity of the RCS and steam generator thick metal in attenuating the resulting plant cooldown.

F. 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, trips the main feedwater pumps, and trips the turbine. The turbine trip signal initiates a reactor trip above the P-9 setpoint. (See section 7.2.)

G. The analysis assumes the RCS flow equivalent to the operation of four RCPs.

H. The analysis considers cases with and without automatic rod control. The automatic rod control cases assume that the rods can automatically insert and withdraw.

The analysis uses RTDP methodology in the determination of initial reactor power, pressure, and RCS temperature. (See tables 15.0.3-2 and 15.0.3-3.)

Normal reactor control systems are not required to function. The reactor protection system may function to trip the reactor due to overpower or high-high steam generator water level conditions. 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

In case of an accidental full opening of one feedwater regulating valve with the reactor at zero power (Mode 3) and the above mentioned assumptions, the maximum reactivity insertion rate is

less than the maximum reactivity insertion rate analyzed in subsection 15.4.1 for uncontrolled rod cluster control assembly bank withdrawal from a subcritical or low-power startup condition; therefore, this section does not present the results of the analysis. Note that the analysis in subsection 15.4.1 conservatively assumes only two RCPs in operation for the DNBR calculation. This assumption bounds the feedwater malfunction event with respect to the DNBR.

The analysis of the Mode 2 case uses the TWINKLE, FACTRAN, and THINC codes to evaluate the resulting reactivity insertion rate as described in subsection 15.4.1. The Mode 2 analysis assumes all four RCPs to be in operation. The resulting minimum calculated DNBR is above the limit value and is less limiting than the analysis presented in subsection 15.4.1; therefore, this section does not present the results of the analysis.

Note that if the incident occurs with the unit just critical at no-load, the reactor may trip by the power range high neutron flux trip (low setting) set at approximately 25-percent nominal full power.

The full power case (maximum reactivity feedback coefficients, manual rod control) results in the greatest power increase. Assuming automatic rod control (insertion and withdrawal) results in a less severe transient.

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. A turbine trip is also generated from the high-high steam generator water level condition. The turbine trip 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 figure 15.1.2-1 and figure 15.1.2-2.) 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 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 power are above the limit value. Additionally, for an excessive feedwater addition in Mode 3, the resulting reactivity insertion rate is less than the value used in the rod withdrawal from subcritical analysis (subsection 15.4.1). For an excessive feedwater addition in Mode 2, the resulting reactivity insertion rate was analyzed and the minimum DNBR is above the limit value and is less limiting than the analysis presented in subsection 15.4.1.

REFERENCE

1. Burnett, T. W. T., et al., "LOFTRAN Code Description," WCAP-7907-P-A (proprietary), WCAP-7907-A (nonproprietary), April 1984.

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
	Minimum DNBR occurs	83.0
	High-high steam generator water level signal generated	86.2
	Turbine trip occurs due to high-high steam generator level	88.7
	Reactor trip occurs	90.7
	Feedwater isolation valves close automatically	93.2
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 reached (approximate time only)	50
3. Automatic rod control (minimum moderator feedback)	10-percent step load increase	0.0
	Equilibrium conditions reached (approximate time only)	125

TABLE 15.1.2-1 (SHEET 2 OF 3)

Accident	Event	Time (s)
4. Automatic rod control (maximum moderator feedback)	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
Steam system piping failure		
1. Case 1 (offsite power available)	Steam line ruptures	0
	Pressurizer empty	16.6
	Criticality attained	21.2
	SI initiation (low steam line pressure)	27.7
	Boron reaches core	77.2
2. Case 2 (concurrent loss of offsite power)	Steam line ruptures	0
	Pressurizer empty	19.0
	Criticality attained	25.2
	SI initiation (low steam line pressure)	42.7
	Boron reaches core	94.8

TABLE 15.1.2-1 (SHEET 3 OF 3)

	Accident	Event	Time (s)
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 DNER)	32.2

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.

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.

15.1.5 STEAM SYSTEM PIPING FAILURE

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.

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 major rupture of a 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 5 s) 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 5 s of actuation following a large break in the steam line. 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, THINC, 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 15.1.4-1.

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 determined that the reactivity employed in the kinetics analysis was always larger than the reactivity calculated, including the above local effects for the limiting conditions. These results verify conservatism; i.e., underprediction of negative reactivity feedback from power generation.

C. Minimum capability for injection of boric acid (2400 ppm) solution corresponding to the most restrictive single failure in the SI system. 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 the 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, which 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 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 12-s delay is assumed to start the diesels and to load the necessary SI equipment onto them.

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, and are different for each case studied.

The core parameters used for each of the two cases correspond to values determined from the respective transient analysis.

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. The MSIVs are designed to be fully closed in less than 5 s from receipt of a closure signal.

As shown in figure 15.1.5-2, the core attains criticality with the RCCAs inserted (with the design shutdown assuming one stuck RCCA) before boron solution at 2400 ppm 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.

Figures 15.1.5-5 through 15.1.5-8 show the salient parameters 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 12 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 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. The elevated appearance rates are assumed to return to normal once the reactor coolant iodine level increases to 100 μCi/g 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 8 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.

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-A, 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.

at beginning of life since this results in the minimum value of moderator temperature coefficient. This assumption maximizes the power rise and minimizes the tendency of increased moderator temperature to flatten the power distribution.

15.4.3.2.1.2 Results

A. One or More Dropped RCCAs

Single or multiple dropped RCCAs within the same group result in a negative reactivity insertion. The core is not adversely affected during this period, since power is decreasing rapidly. Power may be reestablished either by reactivity feedback or control bank withdrawal.

Following a dropped rod event in manual rod control, the plant will establish a new equilibrium condition. The equilibrium process without control system interaction is monotonic, thus removing power overshoot as a concern and establishing the automatic rod control mode of operation as the limiting case.

For a dropped RCCA event in the automatic rod control mode (insertion and withdrawal), the rod control system detects the drop in power and initiates control bank withdrawal. Power overshoot may occur due to this action by the automatic rod controller after which the control system will insert the control bank to restore nominal power. Figures 15.4.3-1 and 15.4.3-2 show a typical transient response to a dropped RCCA (or RCCAs) in automatic control. Uncertainties in the initial condition are included in the DNB evaluation as described in reference 1. In all cases, the minimum DNBR remains above the limit value.

Following plant stabilization, the operator may manually retrieve the RCCA by following approved operating procedures.

B. Dropped RCCA Bank

A dropped RCCA bank results in a negative reactivity insertion greater than 500 pcm. The core is not adversely affected during the insertion period, since

power is decreasing rapidly. The transient will proceed as described in part A; however, the return to power will be less due to the greater worth of the entire bank. The power transient for a dropped RCCA bank is symmetric. Following plant stabilization, normal procedures are followed.

C. Statically Misaligned RCCA

The most severe misalignment situations with respect to DNBR at significant power levels arise from cases in which one RCCA is fully inserted or where bank D is fully inserted with one RCCA fully withdrawn. Multiple independent alarms, including a bank insertion limit alarm, alert the operator well before the transient approaches the postulated conditions. The bank can be inserted to its insertion limit with any one assembly fully withdrawn without the DNBR falling below the limit value.

The insertion limits in the Core Operating Limits Report (COLR) may vary from time to time depending on several limiting criteria. The full-power insertion limits on control bank D must be chosen to be above that position which meets the minimum DNBR and peaking factors. The full power insertion limit is usually dictated by other criteria. Detailed results will vary from cycle to cycle depending on fuel arrangements.

For this RCCA misalignment, with bank D inserted to its full-power insertion limit and one RCCA fully withdrawn, DNBR does not fall below the limit value. The analysis of this case assumes that the initial reactor power, pressure, and RCS temperature are at their nominal values, with the increased radial peaking factor associated with the misaligned RCCA.

For RCCA misalignments with one RCCA fully inserted, the DNBR does not fall below the limit value. The analysis of this case assumes that initial reactor power, pressure, and RCS temperatures are at their nominal values, with the increased radial peaking factor associated with the misaligned RCCA.

DNB does not occur for the RCCA misalignment incident, thus there is no reduction in the ability of the primary coolant to remove heat from the fuel rod. The peak fuel temperature corresponds to a linear heat generation rate based on the radial peaking factor penalty associated

The analysis identified four events considered to be the most likely initiators:

1. Demineralizer outlet isolation valve open during resin flushing.

2. Valve 226 open following BTRS demineralizer flushing operation.

3. Failure to secure chemical addition.

4. Boric acid flow control valve (FV-110A) fails closed during makeup.

Initiator 4 was found to be the most limiting event for Modes 3, 4, and 5a. For Mode 5b, initiator 3 was considered to allow the addition of small amounts of unborated chemical solution into the RCS for water chemistry control. The maximum flowrate possible through this flow path is approximately 3 percent of that associated with the limiting flow path for Modes 3, 4, and 5a. The parameters used in the calculation of time available for operator response are listed in table 15.4.6-1. Conservative values of boron worth (pcm/ppm), as a function of RCS boron concentration, were assumed in the analysis.

Since the active volumes considered are so small in Mode 5b, it was determined that the same valves locked closed in refueling (valves 175 and 183, and, except when required for small chemical additions, valves 176 and 177) would need to be locked closed in Mode 5b. (See paragraph A.)

C. Dilution During Full Power Operation, Including Startup.

For the dilution during startup (Mode 2), the analysis assumes an initial maximum critical boron concentration of 1800 ppm based on the rods being inserted to the insertion limits. The analysis assumes the minimum change in the boron concentration from this initial condition to a hot zero power critical condition to be 300 ppm. The analysis also assumes full rod insertion to occur due to reactor trip, minus the most reactive stuck rod. The analysis assumes the dilution flow to be the combined capacity of the two primary water makeup pumps (approximately 242 gal/min) and a minimum RCS water volume of 9583 ft^3. This volume corresponds to the active volume of the RCS minus the pressurizer and accounts for 10-percent steam generator tube plugging.

During power operation (mode 1), the plant operates under either manual or automatic rod control. While the plant is in manual control, the analysis assumes the dilution flow to be a maximum of 242 gal/min, which is the combined capacity of the two primary water makeup pumps. While in automatic control, the maximum letdown flow (approximately 125 gal/min) limits the dilution flow. The analysis assumes an initial maximum critical boron concentration, corresponding to the rods inserted to the insertion limits at hot full power, of 1800 ppm. The analysis also assumes the minimum change in the boron concentration from this initial condition to a hot zero power critical condition to be 300 ppm. The analysis assumes full rod insertion to occur due to reactor trip, minus the most reactive stuck rod. The analysis uses a minimum water volume of 9583 ft^3 in the RCS, corresponding to the active volume of the RCS minus the pressurizer volume and accounts for 10-percent tube plugging.

No single active failure in any plant systems or equipment will adversely affect the consequences of the accident.

15.4.6.2.2 Results

The calculated sequence of events is shown in table 15.4.1-1.

A. Dilution During Refueling

Since the maximum flowrate associated with the available dilution flow paths in mode 6 is very small, the total time from initiation of event to the eventual complete loss of shutdown margin is significantly large compared to the minimum required operator action time. Therefore, a considerable amount of time is available for the operator to initiate and terminate procedures for RCS water chemistry adjustments before potential loss of shutdown becomes a concern. Additionally, assuming the availability of one high flux at shutdown (HFAS) alarm set at not more than 2.3 times background, it is shown that the technical specification shutdown margin requirement for mode 6 is sufficient to ensure that the operator has 30 minutes from the time of alarm to terminate the dilution before shutdown margin is lost.

B. Dilution During Cold Shutdown, Hot Shutdown, and Hot Standby

For dilution during cold shutdown, hot shutdown, and hot standby, the Core Operating Limits Report (COLR) provides the required shutdown margin as a function of RCS boron concentration. The specified shutdown margin ensures that the operator has 15 min from the time of the HFAS

15.4.9 STEAMLINE BREAK WITH COINCIDENTAL ROD CLUSTER CONTROL ASSEMBLY WITHDRAWAL AT POWER

The automatic rod withdrawal capability of the rod control system is disabled for Vogtle Unit 1 and Unit 2. Physically disabling the automatic rod withdrawal capability eliminates the possibility that a steam line break event will result in a consequential and coincidental rod withdrawal. The analysis presented in this section is retained for historical purposes.

15.4.9.1 Introduction

The coincidental and consequential occurrence of an uncontrolled RCCA bank withdrawal at power following steamline break event is one of four potential interaction scenarios resulting from adverse environmental conditions (either inside or outside of containment) following a high energy line break; these scenarios are identified in "IE Information Notice 79-22." The premise of this concern is that during a high energy line break (such as steamline rupture), certain sensors used in the control systems could be exposed to an adverse environment. If the equipment is not qualified for the adverse environment, a control system malfunction might occur.

The automatic rod control system is one of the control systems that could malfunction. The rod control system relies on the measurement of T_{avg}, nuclear power, and turbine impulse pressure to determine if control rod motion is required. A small steamline rupture may occur outside containment near the turbine impulse pressure transmitters or inside containment in the vicinity of the excore detectors, thus exposing equipment used in the rod control system to an adverse environment. If this equipment is not properly qualified for these conditions, a consequential RCCA withdrawal following a steamline rupture may occur.

The steamline break affects the rod control system (via either an inside containment break near the excore detectors or an outside containment break near the turbine impulse transmitters) and causes the control rods to withdraw following the initiation of the transient. This causes an increase in reactor power and core heat flux to the point at which an OPΔT trip setpoint is reached. This trip terminates the most adverse part of the transient. The steamline break causes increased heat removal and subsequent decrease in primary pressure simultaneous with the increase in reactor power. Secondary pressure also decreases until the low steamline pressure setpoint is reached, initiating steamline isolation and safety injection actuation.

Because of the lower RCS pressure coincident with the increase in reactor power, the consequences at the point of peak heat flux may be more adverse than the RCCA bank withdrawal at power transient analyzed in the FSAR.

The most limiting part of this transient pertinent to this study is immediately before reactor trip (i.e., rod motion). The most limiting case is that for the largest steamline break that trips on OPΔT prior to reaching a reactor trip on a safety injection signal (e.g., low steamline pressure). Therefore, the analysis assumes the largest steamline break size for which a low steamline pressure signal will not occur prior to the OPΔT reactor trip, and the analysis terminates 5 seconds after reactor trip. "Steam System Piping Failure" presented in subsection 15.1.5 bounds the return to power following reactor trip and steamline isolation. If the low steamline pressure setpoint is reached, a reactor trip on safety injection actuation would result and terminate the event. Therefore, like the analysis performed for "Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power" (subsection 15.4.2), to demonstrate protection by the ΔT trips, only the applicable range of these trips needs to be considered. Also note that no credit is taken in the steamline break with coincident rod withdrawal at power analysis for a reactor trip via the high neutron flux overpower protection signal, since this trip function may be inoperable due to adverse environmental conditions associated with a steamline break inside containment.

The performance of the analysis for a steamline break with coincident withdrawal of the control rods due to an adverse environment demonstrates that the corresponding minimum DNBR does not decrease below the appropriate safety analysis limit DNBR value, and no fuel or clad damage occurs. Additionally, no system overpressurization is expected since the steamline break results in an RCS depressurization as described above.

This is an ANS Condition III/IV incident.

15.4.9.2 Analysis of Effects and Consequences

15.4.9.2.1 Method of Analysis

The analysis of this transient uses the LOFTRAN computer code (reference 1). The following assumptions were made for this transient:

A. The analysis employs RTDP methodology in determining initial conditions of maximum core power, reactor coolant average temperature, and minimum reactor coolant pressure.

B. For end of life shutdown margin and equilibrium xenon conditions, the analysis assumes the most reactive RCCA stuck in its fully withdrawn position for conditions following reactor trip.

C. The analysis uses a negative moderator coefficient corresponding to the end of life unrodded core. This maximizes the reactivity insertion caused by the cooldown during the steamline break.

D. The analysis assumes the reactor trip setpoint on OPΔT at a conservative value. The ΔT trip includes all adverse instrumentation and setpoint errors; the delays for trip actuation are at the maximum values.

E. The analysis bases the RCCA trip insertion characteristic on the assumption that the highest worth assembly is stuck in its fully withdrawn position.

F. A spectrum of break sizes are analyzed. The limiting break is the largest break size for which a low steam line pressure signal will not occur and a reactor trip occurs on OPΔT.

G. The calculation of the steam flow during a steamline break uses the Moody Curve for f L/D = 0.

H. A conservatively large reactivity insertion rate is used.

No single active failure in any plant systems or equipment will adversely affect the consequences of the accident.

15.4.9.2.2 Results

The minimum DNBR occurred with beginning of life reactivity coefficients and a 0.7 ft break area. The calculated sequence of events for the limiting case is shown in table 15.4.1-1.

Figures 15.4.9-1, 15.4.9-2 and 15.4.9-3 show transient conditions following the steam line rupture with coincident RCCA bank withdrawal.

The steamline break affects the turbine impulse transmitters and causes the control rods to withdraw at the initiation of the transient. This causes an increase in reactor power and core heat flux to the point at which the OPΔT trip setpoint is reached. The reactor trip terminates the most adverse part of the transient. The steamline break causes increased heat removal and subsequent decrease in primary pressure simultaneous with the increase in reactor power. If the transient extends beyond post-reactor trip, secondary pressure will decrease until the low steamline pressure setpoint is reached, initiating steamline isolation and safety injection actuation.

The analysis of the steamline break with coincident RCCA bank withdrawal demonstrates that the DNBR limit is met. The most limiting part of this transient pertinent to this study was immediately before reactor trip (i.e., rod motion). The transient for the steamline break presented in subsection 15.1.5 bounds the return to power following reactor trip and steamline isolation. The other FSAR steamline break analysis assumed a larger break size and initial conditions corresponding to no-load temperatures (i.e., less stored energy in the RCS and reactor fuel). The DNBR is always greater than the limit value. Figure 15.4.9-3 shows the DNBR as a function of time for this transient.

15.4.9.3 Conclusions

The analysis demonstrates that the DNBR does not decrease below the limit value and no fuel or clad damage occurs. Additionally, no system overpressurization will occur; thus, all applicable safety criteria are met. As stated in the results, the large steamline break analysis presented in subsection 15.1.5 bounds the return to power following a reactor trip and steamline isolation; therefore, there is adequate protection to ensure plant safety for this transient.

B. Fuel cladding integrity shall be maintained by ensuring that the minimum departure from nucleate boiling ratio (DNBR) remains above the 95/95 DNBR limit for PWRs.

C. An incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently.

To address criterion C, the analysis historically used the more restrictive criterion that a water-solid pressurizer condition be precluded when the pressurizer is at or above the set pressure of the pressurizer safety valves (PSVs). This addressed any concerns regarding subcooled water relief through the plant PSVs. The current analysis conservatively predicts the minimum time to reaching a pressurizer water solid condition and the resulting water relief characteristics (if applicable). An evaluation of the PSV operability for temporary water relief under the specific conditions of this transient was performed. The evaluation demonstrated that a more serious plant condition will not result following an inadvertent ECCS actuation by confirming that the RCS pressure boundary remains intact for the post-transient plant shutdown.

The inadvertent ECCS actuation at power event is analyzed to determine both the minimum DNBR value and maximum pressurizer volume (or minimum time to a pressurizer water-solid condition and subsequent water relief). The most limiting case with respect to DNB is a minimum reactivity feedback condition with the plant assumed to be in manual rod control. Because of the power and temperature reduction during the transient, operating conditions do not approach the core limits.

For maximizing the potential for pressurizer filling, the most limiting case is a maximum reactivity feedback condition with an immediate reactor trip, and subsequent turbine trip, on the initiating SI signal. The transient results are presented for each case.

The analysis assumptions are as follows:

A. Initial Operating Conditions

The DNB case is analyzed with the Revised Thermal Design Procedure as described in WCAP-11397-P-A (reference 2). Initial reactor power, RCS pressure, and temperature are assumed to be at the nominal full power values. Uncertainties in initial conditions are included in the limit DNBR as described in reference 2.

VEGP has a vessel average temperature window from 570°F to 588.4°F; therefore, cases at each end of this window are analyzed for pressurizer filling. The initial conditions for these cases assume maximum uncertainties on power (+2 percent), vessel average temperature (-5°F),and pressurizer pressure (-50 psia).

B. Moderator and Doppler Coefficients of Reactivity

The minimum feedback case (DNB) assumes a positive (+7 pcm/°F) moderator temperature coefficient and a low absolute value Doppler power coefficient at beginning of life (BOL). The maximum feedback case (pressurizer filling) assumes a large (absolute value) negative moderator temperature coefficient and a most-negative Doppler power coefficient.

C. Reactor Control

For the DNB case (without direct reactor trip on SI) the reactor is assumed to be in manual rod control.

D. Pressurizer Pressure Control

For the DNB case, the pressurizer heaters are inoperable. This assumption yields a higher rate of pressurizer decrease. The pressurizer spray portion of the pressurizer pressure control system is assumed available in order to minimize the RCS pressure. The PORVs are also assumed operable. PORVs reduce RCS pressure which is conservative for DNB analyses.

For the pressurizer filling cases, the following pressurizer pressure control system assumptions are made:

- Normal operation of the pressurizer spray is assumed. Spray actuates as a result of the pressure increase. Spray is assumed to be fully effective in condensing steam and thus maintaining a lower pressure until the water level increases to the point where the spray nozzle is submerged (i.e., when the pressurizer is nearly filled). This maximizes the ECCS injection flow.

- Operation of the pressurizer heaters minimizes the time to fill the pressurizer and subsequent water relief through the PSV(s). This is important since the maximum number of water relief cycles supported in the valve operability evaluation is three. Operation of the heaters, however, increases the temperature of the water that is relieved. Colder water relief temperatures are more limiting with respect to valve operability. Therefore, to ensure

that both criteria are conservatively addressed, cases have been analyzed assuming both normal heater operation and no heater operation.

- PORVs are not assumed as an automatic pressure control system for the pressurizer filling case. Automatic actuation of the PORVs would directly mitigate the event consequences by preventing water relief through the PSVs. Operator action to make one PORV available following an acceptable delay is credited in the analysis to mitigate the event. Should the PORVs fail due to water relief, the block valves would be available to isolate the RCS.

E. ECCS Injection

The inadvertent ECCS analysis models a maximum ECCS flow rate that bounds operation at the original design which included a positive displacement pump and the plant configuration with the replacement centrifugal pump. Safety injection (SI) is actuated at time zero, with flow injected to the RCS from two high-head centrifugal charging pumps plus the normal charging pump. Both high-head charging and SI pumps automatically start on an SI signal, and the associated alternate minimum flow protection lines receive a signal to open at high RCS back pressures. The failure of the normal charging pump to be stripped from the bus was taken as the single failure since it results in higher flow rates than the failure to open one safety-related miniflow path. The analysis also assumes zero injection line purge volume for calculation simplicity; thus, the boration transient begins immediately in the analysis.

F. Turbine Load

For the DNB case (without direct reactor trip/turbine trip on SI), the turbine load remains constant until the governor drives the throttle valve wide open. After the throttle valve is full open, turbine load decreases as steam pressure drops.

G. Reactor Trip

Reactor trip is initiated by a low pressurizer pressure signal at 1935 psia for the DNB case. The pressurizer filling case assumes reactor trip on the initiating SI signal.

H. Decay Heat

Core residual heat generation is based on the 1979 version of ANS 5.1 (reference 3). ANSI/ANS-5.1-1979 is a conservative representation of the decay energy release rates. Long-term operation at the initial power level preceding the trip is assumed.

I. Pressurizer Safety Valves

The safety valves open at a pressure of 2425 psia which corresponds to a tolerance of -2 percent relative to the set pressure of 2475 psia. The valves are assumed to close at a pressure of 2300 psia, which corresponds to a blowdown of 5 percent below the opening pressure of 2425 psia.

J. Operator Actions

An operator action, to make one PORV available for water relief, was assumed at 590 seconds in the pressurizer fill case initialized from the low end of the vessel average window (570°F), and 625 seconds was assumed for the pressurizer fill case initialized from the high end of the vessel average window (588.4°F).

15.5.1.2.2 Results

The transient responses for the DNB and limiting pressurizer filling cases are shown in figure 15.5.1-1. Table 15.5.1-1 shows the calculated sequence of events.

For the DNB case, nuclear power starts decreasing immediately due to boron injection, but steam flow does not decrease until later in the transient when the turbine throttle valve is wide open. The mismatch between load and nuclear power causes T_{avg}, pressurizer water level, and pressurizer pressure to drop. The reactor trips and control rods start moving into the core when the pressurizer pressure reaches the pressurizer low pressure trip setpoint. The DNBR increases throughout the transient.

For the pressurizer filling case, reactor trip occurs at event initiation followed by a rapid initial cooldown of the RCS. Coolant contraction results in a short-term reduction in pressurizer pressure and water level. The combination of the RCS heatup, due to residual RCS heat generation, and ECCS injected flow causes the pressure and level transients to rapidly turn around. Pressurizer water level then increases throughout the transient.

In the case initialized from the low end of the vessel average window (570°F), the analysis assumes that at 590 seconds the operator manually opens a PORV. For the case initialized from the high end of the vessel average window (588.4°F), this operator action is assumed at 625 seconds. In each of these cases, once the PORV is fully open, the RCS begins to depressurize to below the pressure where the safety valves reseat.

15.5.1.3 Conclusions

Results of the analysis show that spurious ECCS operation without immediate reactor trip does not present any hazard to the integrity of the RCS with respect to DNBR. The minimum DNBR is never less than the initial value. Thus, there will be no cladding damage and no release of fission products to the RCS. If the reactor does not trip immediately, the low pressurizer pressure reactor trip will provide protection. This trips the turbine and prevents excess cooldown, which expedites recovery from the incident.

With respect to pressurizer filling, the pressurizer may reach a water-solid condition. However, the resulting potential water relief will not impair PSV operability. The RCS pressure boundary will therefore remain intact and the event will not generate a more serious plant condition.

REFERENCE

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.

3. ANSI/ANS-5.1-1979, "Decay Heat Power in Light Water Reactors," August 29, 1979.

TABLE 15.5.1-1 (Sheet 1 of 2)

TIME SEQUENCE OF EVENTS FOR INCIDENTS WHICH RESULT IN AN INCREASE IN REACTOR COOLANT INVENTORY
(Accident: Inadvertent operation of ECCS during power operation)

Case	Event	Time(s)	
DNBR case:	SI pumps begin injecting borated water	0.0	
	Low pressurizer pressure reactor trip setpoint reached	45.5	
	Rods begin to drop	47.5	
	Minimum DNBR occurs	(a)	
Pressurizer filling case:		Pressurizer Heaters	
		On	Off
A nominal T_{AVG} = 570.7°F	SI actuation, reactor trip	0.0	0.0
Operator action to manually open PORV at 590 s	Pressurizer fills with water	448.0	471.0
	PSV opens - cycle #1	495.3	518.7
	PSV closes - cycle #1	499.2	522.7
	PSV opens - cycle #2	529.4	553.0
	PSV closes - cycle #2	533.3	556.9
	PSV opens - cycle #3	561.0	584.4
	PSV closes - cycle #3	564.8	588.0
	Time of last PSV cycle (minimum final water relief temperature)	565.0 (633.8°F)	588.0 (628.4°F)
	Operator action to open PORV	590.0	590.0

TABLE 15.5.1-1 (Sheet 2 of 2)

Accident	Event	Time(s) Pressurizer Heaters On	Time(s) Pressurizer Heaters Off
A nominal T_{AVG} = 588.4°F	SI actuation, reactor trip	0.0	0.0
Operator action to manually open PORV at 625 s	Pressurizer fills with water	480.0	501.5
	PSV opens - cycle #1	529.4	550.8
	PSV closes - cycle #1	533.2	554.4
	PSV opens - cycle #2	564.5	586.4
	PSV closes - cycle #2	568.3	590.3
	PSV opens - cycle #3	597.9	620.9
	PSV closes - cycle #3	601.7	624.8
	Time of last PSV cycle (minimum final water relief temperature)	602.0 (636.9°F)	625.0 (631.8°F)
	Operator action to open PORV	625.0	625.0

a. DNBR does not decrease below its initial value.

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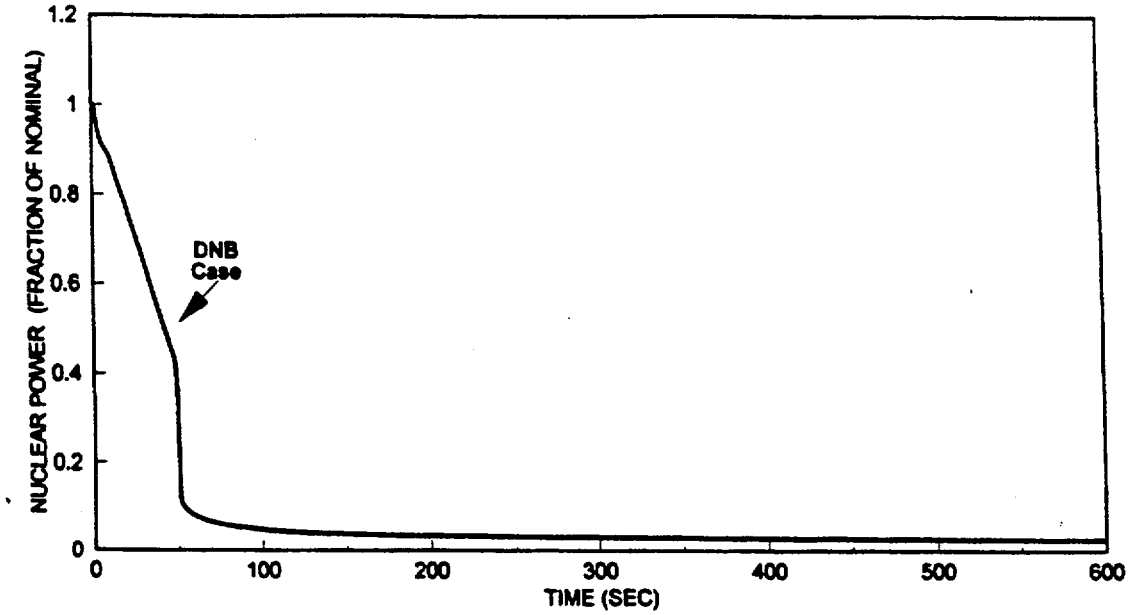

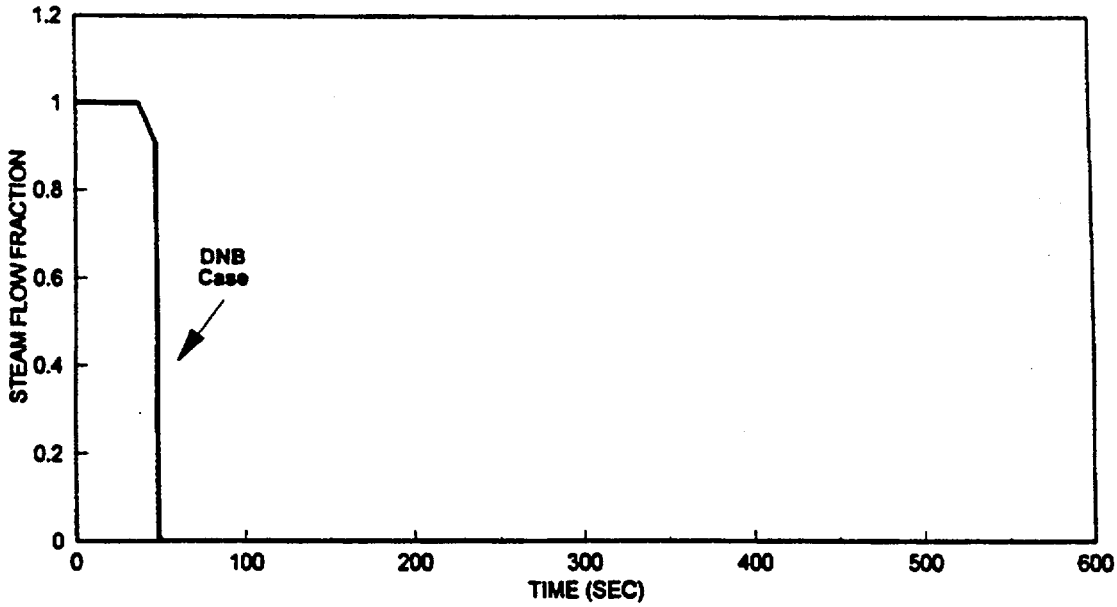

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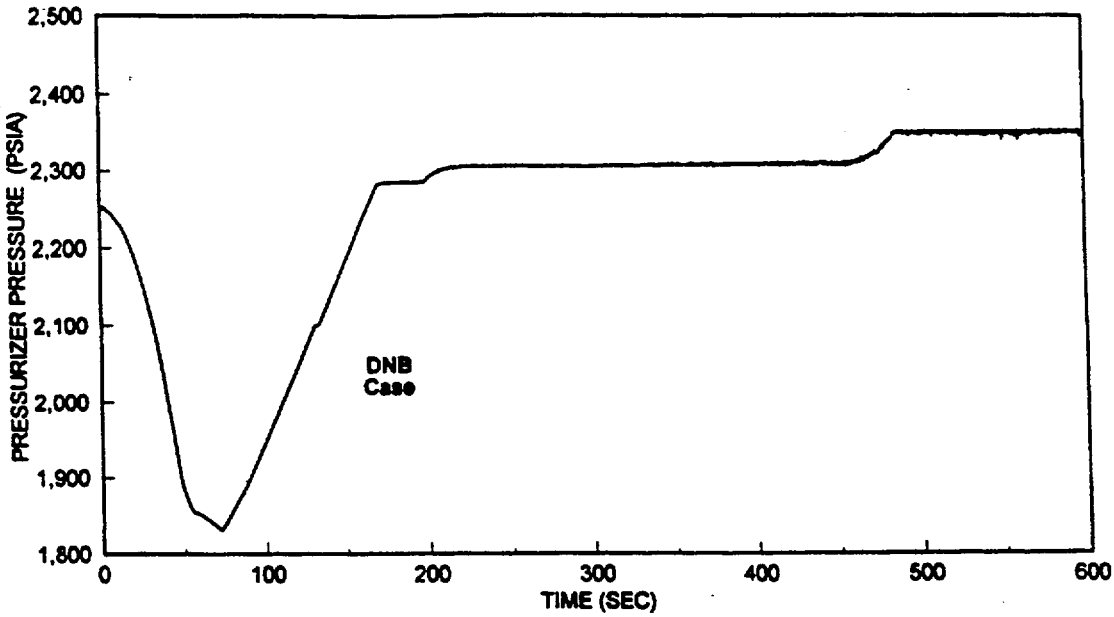
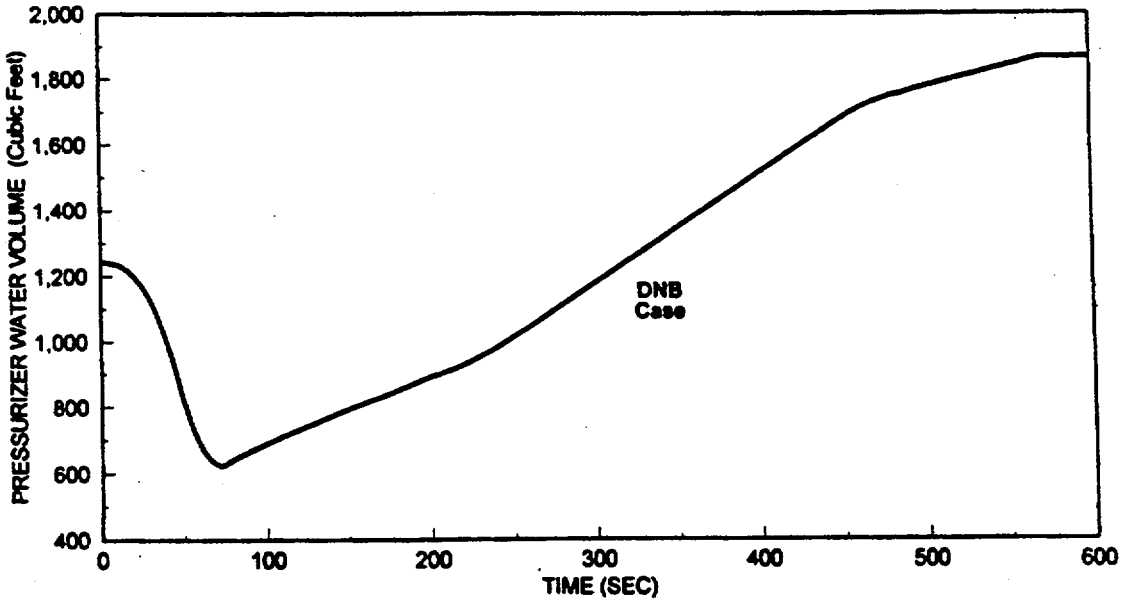
REV 5 9/95
REV 3 12/92

VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2	INADVERTENT OPERATION OF ECCS AT POWER
	FIGURE 15.5.1-1 (SHEET 1 OF 9)





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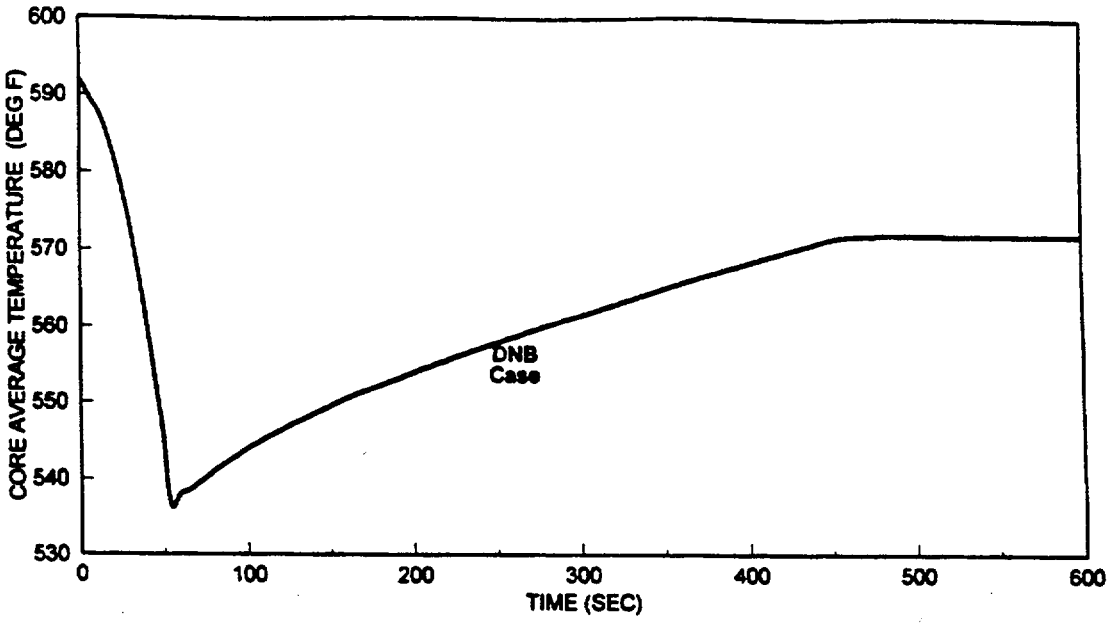
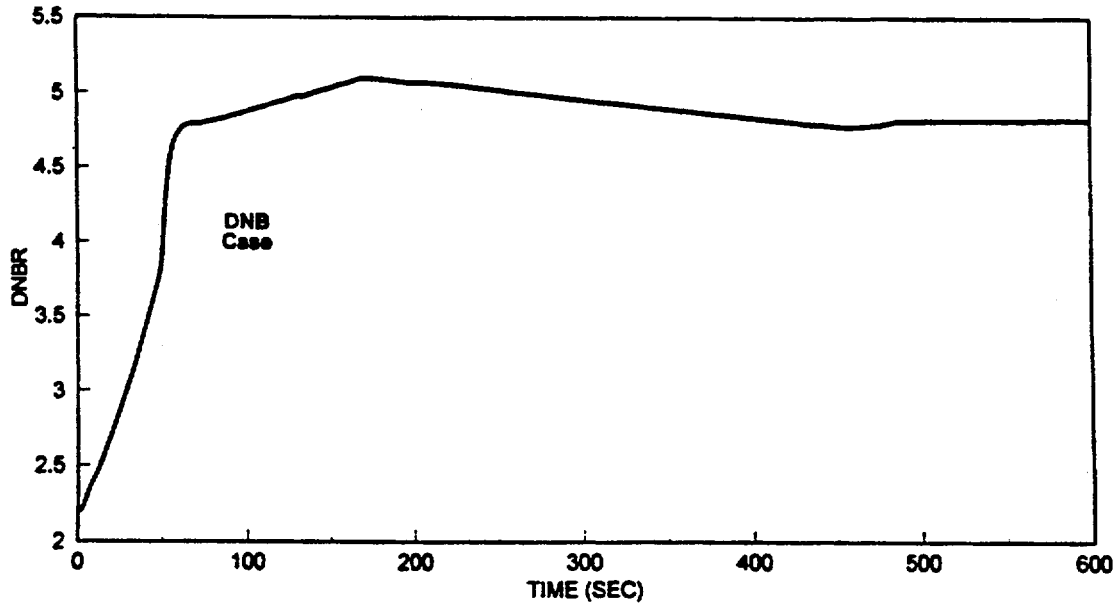
REV 5 9/95
REV 3 12/92

VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2	INADVERTENT OPERATION OF ECCS AT POWER
	FIGURE 15.5.1-1 (SHEET 2 OF 9)





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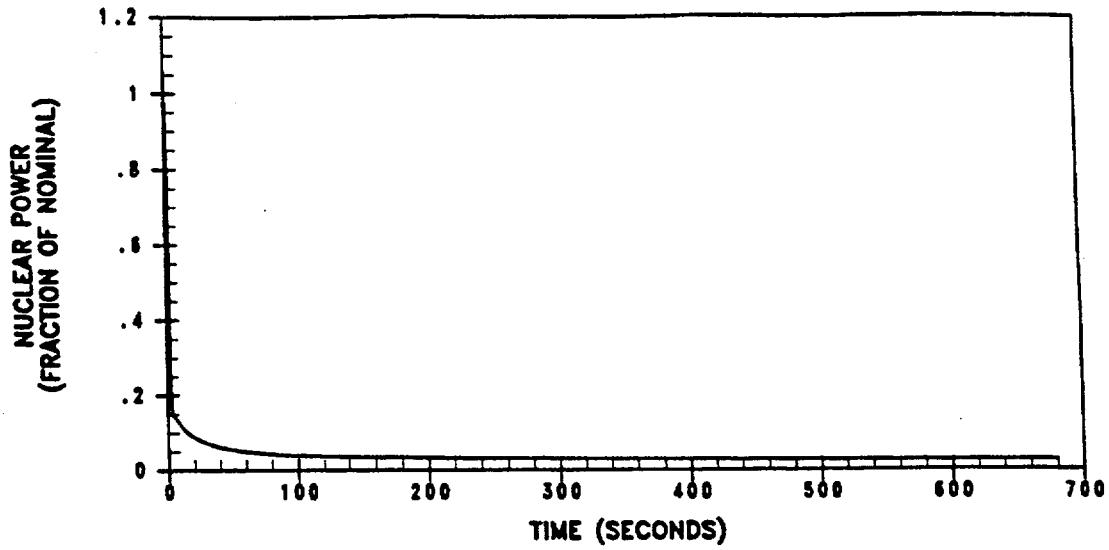
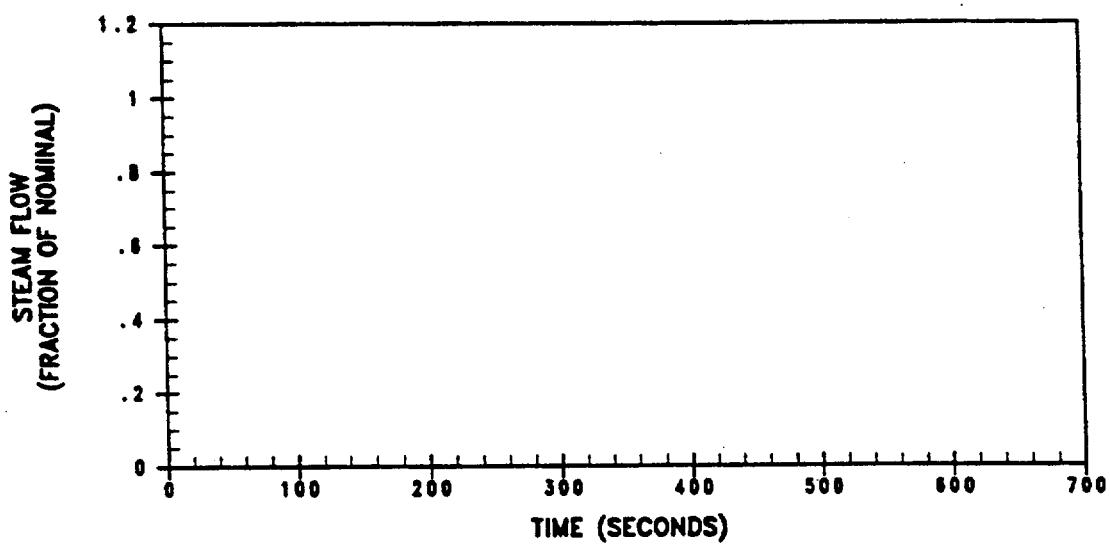
VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2	INADVERTENT OPERATION OF ECCS AT POWER
	FIGURE 15.5.1-1 (SHEET 3 OF 9)





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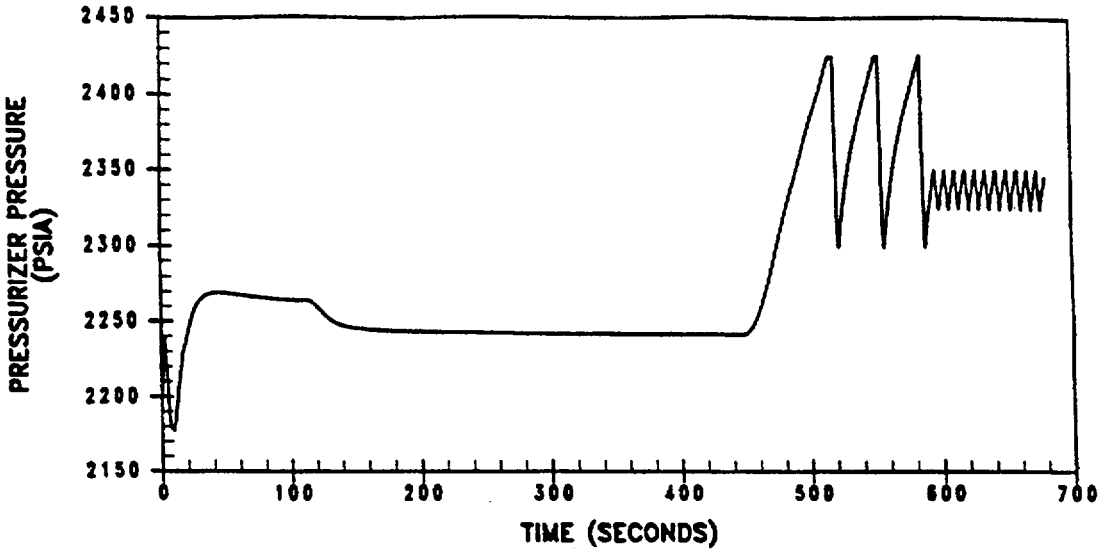
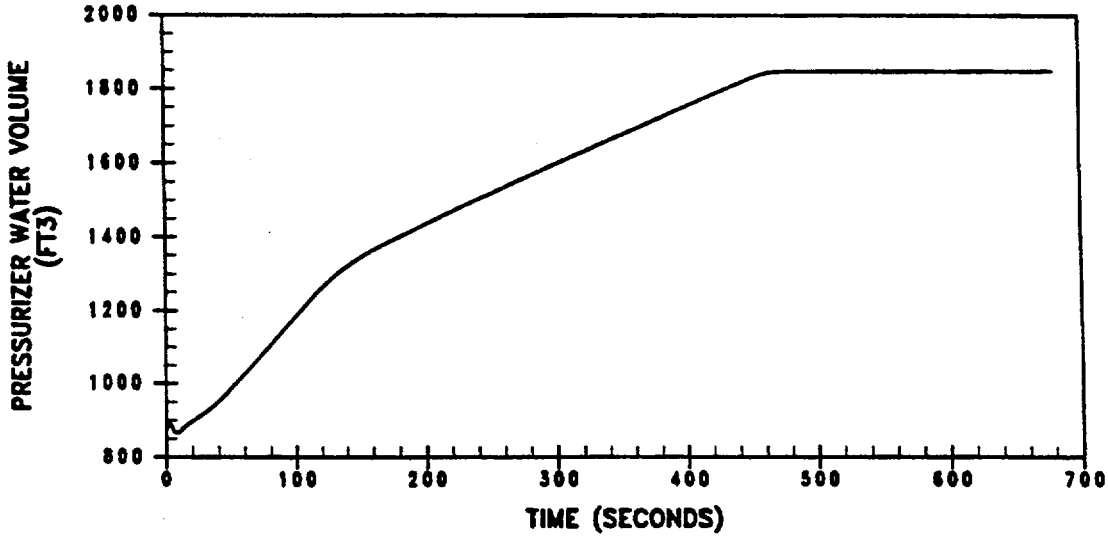
VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2	INADVERTENT OPERATION OF ECCS AT POWER - NUCLEAR POWER AND STEAM FLOW AS A FUNCTION OF TIME (TAVG=570.7 ^{o}F, W/O Pressurizer Heaters)
	FIGURE 15.5.1-1 (SHEET 4 OF 9)





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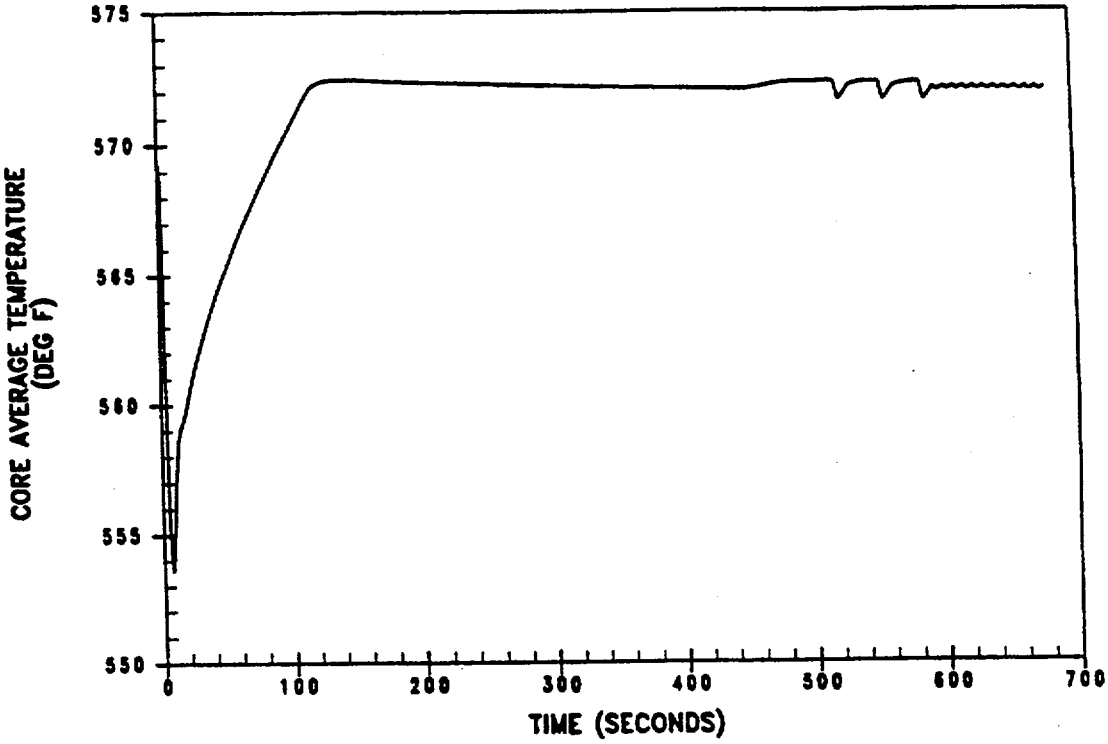
VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2	INADVERTENT OPERATION OF ECCS AT POWER - PRESSURIZER PRESSURE AND WATER VOLUME AS A FUNCTION OF TIME (TAVG=570.7 °F, W/O Pressurizer Heaters)
	FIGURE 15.5.1-1 (SHEET 5 OF 9)



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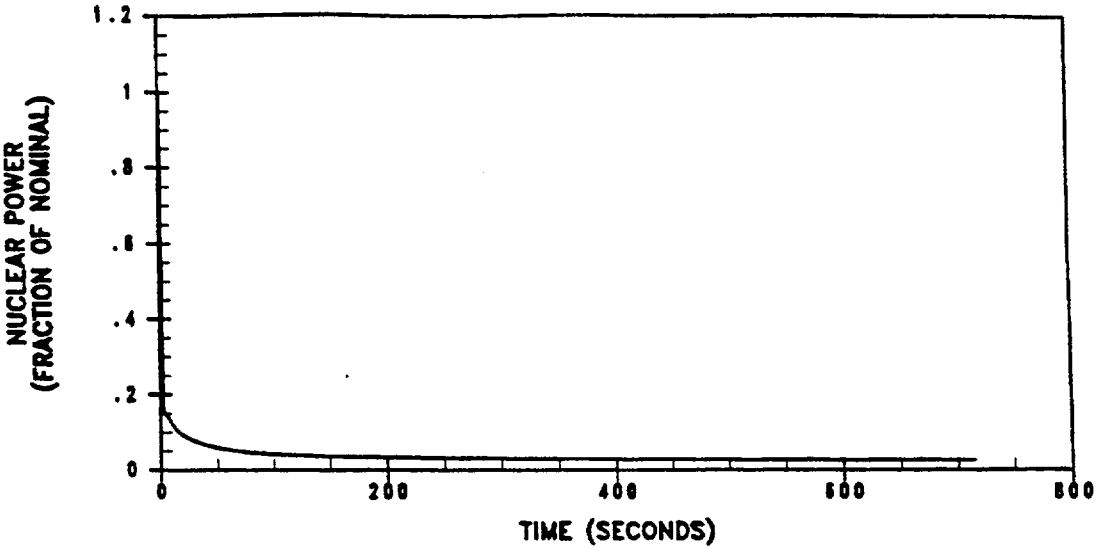
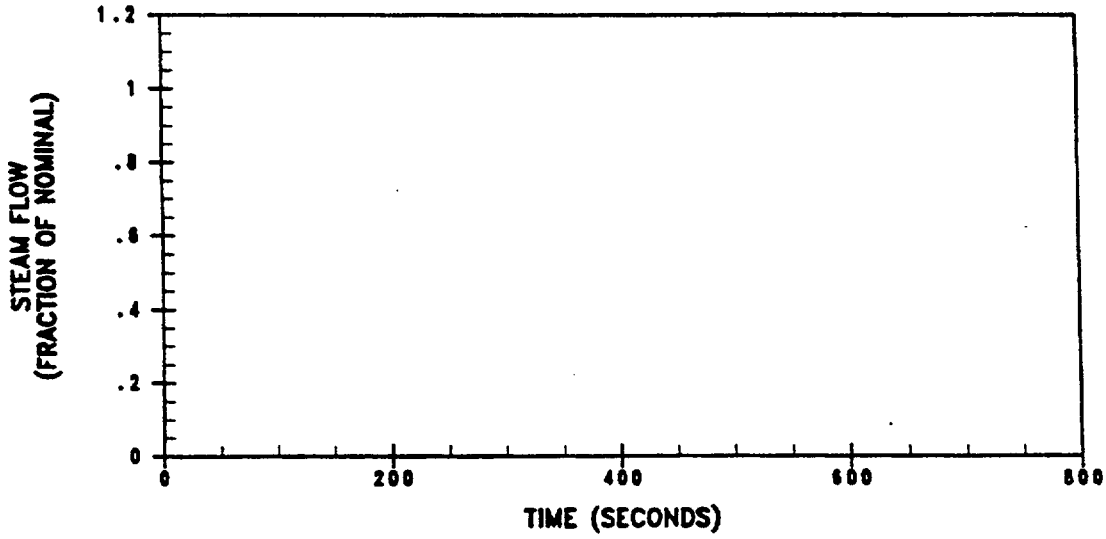
VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2	INADVERTENT OPERATION OF ECCS AT POWER - CORE AVERAGE TEMPERATURE AS A FUNCTION OF TIME (TAVG=570.7 °F, W/O Pressurizer Heaters) FIGURE 15.5.1-1 (SHEET 6 OF 9)





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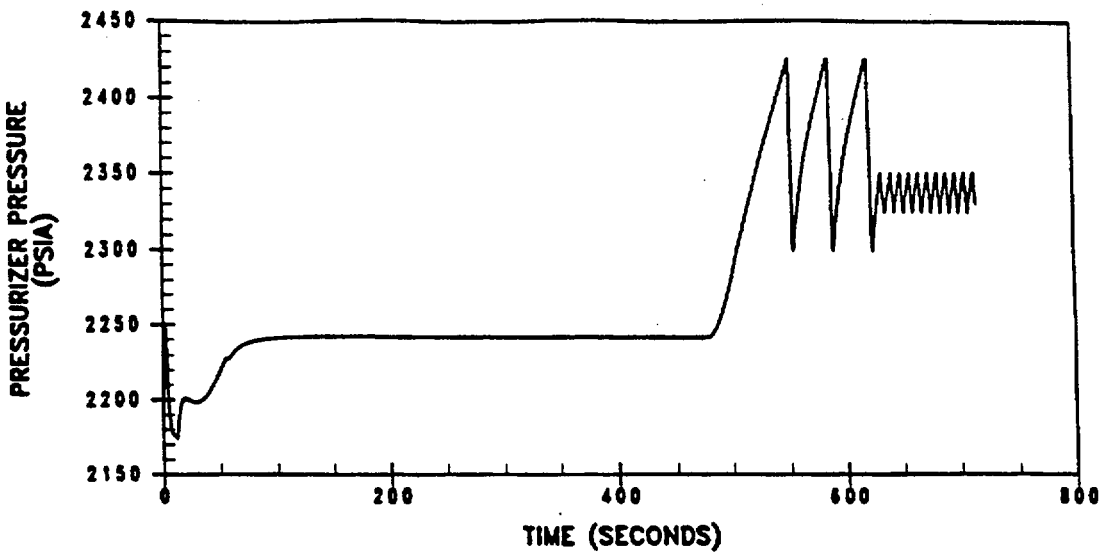
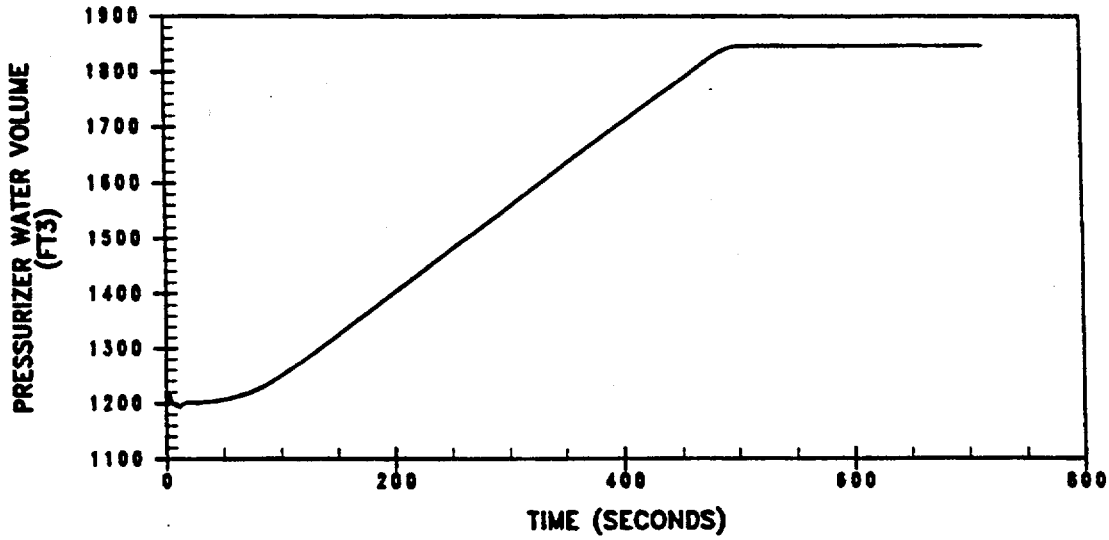
VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2	INADVERTENT OPERATION OF ECCS AT POWER - NUCLEAR POWER AND STEAM FLOW AS A FUNCTION OF TIME (TAVG=588.4 ^{o}F, W/O Pressurizer Heaters)
	FIGURE 15.5.1-1 (SHEET 7 OF 9)





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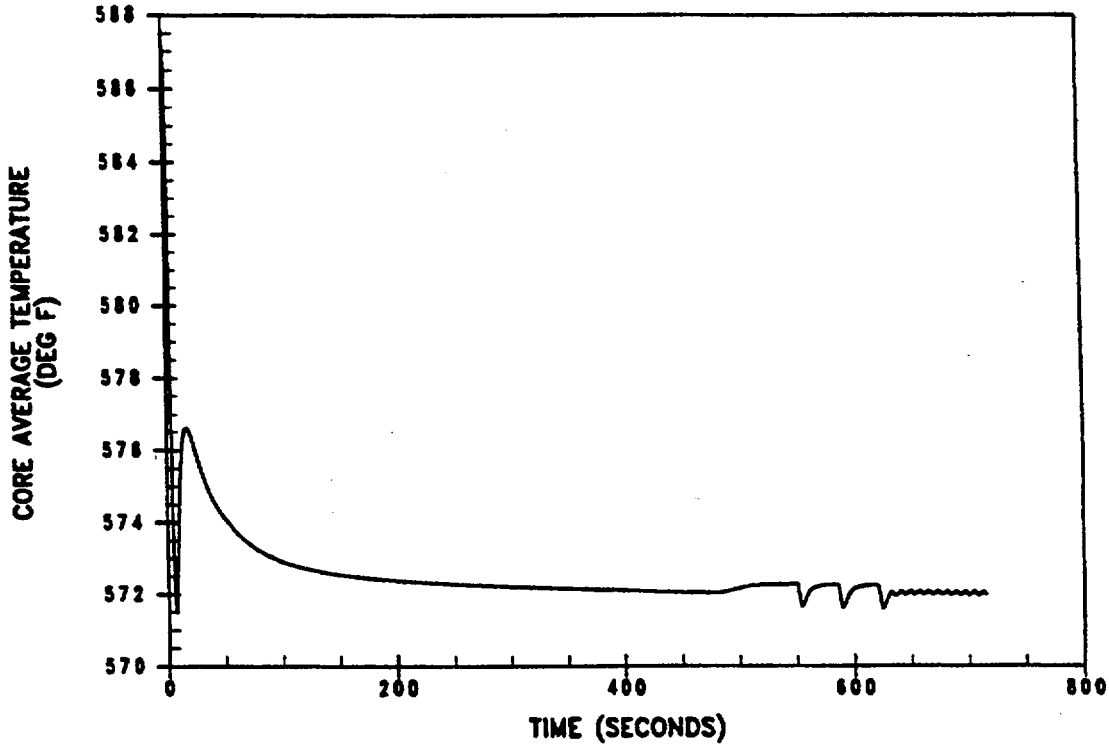
VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2	INADVERTENT OPERATION OF ECCS AT POWER - PRESSURIZER PRESSURE AND WATER VOLUME AS A FUNCTION OF TIME (TAVG=588.4 °F, W/O Pressurizer Heaters)
	FIGURE 15.5.1-1 (SHEET 8 OF 9)



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VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2	INADVERTENT OPERATION OF ECCS AT POWER - CORE AVERAGE TEMPERATURE AS A FUNCTION OF TIME (TAVG=588.4 ^{o}F, W/O Pressurizer Heaters)
	FIGURE 15.5.1-1 (SHEET 9 OF 9)

15.6.2 BREAK IN INSTRUMENT LINE OR OTHER LINES FROM REACTOR COOLANT PRESSURE BOUNDARY THAT PENETRATE CONTAINMENT

The reactor coolant system (RCS) wide-range pressure instrument lines penetrate the containment. In addition, there are the sample lines from the hot legs of reactor coolant loops 1 and 4, from the steam and liquid space of the pressurizer, and from the 3-in. chemical and volume control system (CVCS) letdown line penetrating the containment. The sample lines are provided with isolation valves on both sides of the containment wall and are designed in accordance with the requirements of General Design Criterion (GDC) 55.

A break in one of the RCS wide-range pressure instrument lines at a point outside the containment but before the hydraulic isolator combined with the failure of the instrument bellows would result in a nonisolable release of reactor coolant outside the containment. While the release would continue until the plant is shut down and the RCS is depressurized, the release rate never exceeds 0.106 lb/s.

The most severe pipe rupture outside containment, with regard to radioactivity release during normal plant operation, occurs in the CVCS. This would be a complete severance of the 3-in. letdown line just outside containment but between the outboard letdown isolation valve and letdown heat exchanger at rated power condition. (See drawings 1X4DB114, 1X4DB115, 1X4DB116-1, 1X4DB116-2, 1X4DB117, and 1X4DB118.) The occurrence of a complete severance of the letdown line would result in a loss of reactor coolant at the rate of about 19 lb/s. Since the release rate is within the capability of the reactor makeup system, it would not result in engineered safety features system actuation. Frequent operation of the automatic reactor makeup system in addition to the low pressurizer pressure deviation alarm will provide the operator some indication of the loss of reactor coolant. After the 30 min, the operator is assumed to isolate the letdown line.

15.6.2.1 Assumptions - Instrument Line Break

The major assumptions and parameters used in the analysis are provided in table 15.6.2-1 and summarized below:

A. The reactor coolant iodine activity is based on a pre-existing iodine spike of 60-μCi/g dose equivalent I-131. (See table 15A-6.)

B. The noble gas activity in the reactor coolant is based on 1-percent fuel defects. (See table 11.1-2.)

C. Reactor shutdown is assumed to be initiated 8 h after the break occurs.

D. One residual heat removal system train is assumed to be available. Time to cool the RCS to <212°F is 24 h.

E. A total of 12,200 lb of reactor coolant is spilled (based on a release for 32 h).

F. All of the noble gases in the spilled reactor coolant are released to the environment.

G. The fraction of the spill assumed to flash is 58 percent. All of the iodine activity in the flashed fraction of the spill is assumed to be released. Ten percent of the iodine in the unflashed fraction is assumed to be released.

H. No credit is taken for mixing and holdup of the releases within the auxiliary building nor are the auxiliary building normal exhaust filters credited with reducing the release. That is, the release is modeled as being direct to the environment.

I. No credit is taken for ground deposition or decay in transit to the exclusion area boundary or outer boundary of the low population zone.

15.6.2.2 Assumptions - Letdown Line Break

The major assumptions and parameters used in the analysis are provided in table 15.6.2-2 and summarized below:

A. The reactor coolant iodine activity is based on a pre-existing iodine spike of 60-μCi/g dose equivalent I-131. (See table 15A-6.)

B. The noble gas activity in the reactor coolant is based on 1-percent fuel defects. (See table 11.1-2.)

C. A total of 34,200 lb of reactor coolant is spilled (based on a release for 30 min).

D. All of the noble gases in the spilled reactor coolant are released to the environment.

E. The fraction of the spill assumed to flash is 8.6 percent. All of the iodine activity in the flashed fraction of the spill is assumed to be released. Ten percent of the iodine in the unflashed fraction is assumed to be released.

15.6.3 STEAM GENERATOR TUBE FAILURE

15.6.3.1 Identification of Causes and Accident Description

The accident examined is the complete severance of a single steam generator tube. This event is considered an American Nuclear Society (ANS) Condition IV event, a limiting fault. (See subsection 15.0.1.) The accident is assumed to take place at full power with the reactor coolant contaminated with fission products corresponding to continuous operation with a limited number of defective fuel rods. The accident leads to an increase in contamination of the secondary system due to leakage of radioactive coolant from the reactor coolant system (RCS). In the event of a coincident loss of offsite power or failure of the condenser steam dump system, discharge of activity to the atmosphere takes place via the steam generator power-operated relief valves (and safety valves if their setpoint is reached).

Complete severance of a steam generator tube is considered a somewhat conservative assumption since the Inconel-600 tube material is highly ductile. The more probable mode of tube failure would be one or more minor leaks of undetermined origin. Activity in the steam and power conversion system is subject to continual surveillance, and an accumulation of minor leaks which exceed the limits established in the Technical Specifications is not permitted during the unit operation.

The operator is expected to determine that a steam generator tube rupture has occurred, to identify and isolate the faulted steam generator, and to complete the required recovery actions to stabilize the plant and terminate the primary to secondary break flow. These actions should be performed on a restricted time scale in order to minimize contamination of the secondary system and ensure termination of radioactive release to the atmosphere from the faulted unit. Consideration of the indications provided at the control board, together with the magnitude of the break flow, leads to the conclusion that the recovery procedure can be carried out on a time scale which ensures that break flow to the secondary system is terminated before water level in the affected steam generator rises into the main steam pipe. Sufficient indications and controls are provided to enable the operator to carry out these functions satisfactorily.

If normal operation of the various plant control system is assumed, the following sequence of events is initiated by a tube rupture.

A. Pressurizer low-pressure and low-level alarms are actuated and charging pump flow increases in an attempt to maintain pressurizer level. On the secondary side, steam flow/feedwater flow mismatch occurs, since feedwater flow to the affected steam generator is reduced as a result of primary coolant break flow to that unit.

B. The main steamline radiation monitors, the condenser air ejector radiation monitor, and/or the steam generator blowdown liquid monitor will alarm, indicating a sharp increase in radioactivity in the secondary system. The high radiation level alarm from the steam generator blowdown process monitor automatically isolates the system and terminates discharge. The high radiation level alarm from the air ejector monitor automatically diverts the air ejector and steam seal exhauster blower discharges through a filtration unit.

C. The decrease in RCS pressure due to continued loss of reactor coolant inventory leads to a reactor trip signal on low pressurizer pressure or OTΔT. Resultant plant cooldown following reactor trip leads to a rapid decrease in RCS pressure and pressurizer level, and a safety injection signal initiated by low pressurizer pressure follows soon after reactor trip. The safety injection signal automatically terminates normal feedwater supply and initiates auxiliary feedwater addition.

D. The reactor trip automatically trips the turbine, and if offsite power is available, the steam dump valves open, permitting steam dump to the condenser. In the event of a coincident loss of offsite power, the steam dump valves automatically close to protect the condenser. The steam generator pressure rapidly increases, resulting in steam discharge to the atmosphere through the steam generator power-operated relief valves (and safety valves if their setpoint is reached).

E. Following reactor trip and safety injection actuation, the continued action of the auxiliary feedwater supply and borated safety injection flow (supplied from the refueling water storage tank) provides a heat sink which absorbs some of the decay heat. This reduces the amount of steam bypass to the condenser, or in the case of loss of offsite power, steam relief to the atmosphere.

leakage. Primary to secondary leakage will continue after SI flow is stopped until RCS and ruptured steam generator pressures equalize. Charging flow, letdown, and pressurizer heaters will then be controlled to prevent repressurization of the RCS and reinitiation of leakage into the ruptured steam generator.

Following SI termination, the plant conditions will be stabilized, the primary to secondary break flow will be terminated, and all immediate safety concerns will have been addressed. At this time, a series of operator actions are performed to prepare the plant for cool down to cold shutdown conditions. Subsequently, actions are performed to cool down and depressurize the RCS to cold shutdown conditions and to depressurize the ruptured steam generator.

15.6.3.2 Analysis of Effects and Consequences

An SGTR results in the leakage of contaminated reactor coolant into the secondary system and subsequent release of a portion of the activity to the atmosphere. Therefore, an analysis must be performed to assure that the offsite radiological consequences resulting from an SGTR are within the allowable guidelines. One of the major concerns for an SGTR is the possibility of steam generator overfill since this could potentially result in a significant increase in the offsite radiological consequences. Therefore, an analysis was performed to demonstrate margin to steam generator overfill as documented in reference 1, assuming the limiting single failure relative to overfill. The analysis assumes that one of the steam generator atmospheric relief valves (ARVs) is out of service, as allowed by the Technical Specifications. The analysis was performed assuming one operable ARV was on the ruptured steam generator and the other two operable ARVs were on intact steam generators. The limiting single failure for the margin to overfill analysis results in the loss of control room (CR) control of the ARVs on the intact steam generators. Action outside the CR is required to open an intact steam generator ARV to perform the plant cooldown. The analysis assumed that a single intact steam generator ARV is used for the cooldown, and it is not opened until 19 minutes after the ruptured steam generator is isolated. The results of this analysis demonstrated that there is margin to steam generator overfill for VEGP. An analysis was also performed to determine the offsite radiological consequences as documented in reference 1, assuming the limiting single failure relative to offsite doses without steam generator overfill. Since steam generator overfill does not occur, the results of this analysis represent the limiting consequences for an SGTR for VEGP.

A thermal and hydraulic analysis was performed to determine the plant response for a design basis SGTR, and to determine the integrated primary to secondary break flow and the mass releases

from the ruptured and intact steam generators to the condenser and to the atmosphere. This information was then used to calculate the quantity of radioactivity released to the environment and the resulting radiological consequences.

15.6.3.3 Thermal and Hydraulic Analysis

The plant response following an SGTR was analyzed with the LOFTTR2 (reference 1) program until the primary to secondary break flow is terminated. The reactor protection system and the automatic actuation of the engineered.safeguards systems were modeled in the analysis. The major operator actions which are required to terminate the break flow for an SGTR were also simulated in the analysis.

Analysis Assumptions

The accident modeled is a double-ended break of one steam generator tube located at the top of the tube sheet on the outlet (cold leg) side of the steam generator. It was assumed that the reactor is operating at full power at the time of the accident and the secondary level was assumed to correspond to operation at the nominal steam generator level minus an allowance for uncertainties. It was also assumed that a loss of offsite power occurs at the time of reactor trip and the highest worth control assembly was assumed to be stuck in its fully withdrawn position at reactor trip.

Other important analysis assumptions include:

A. NSSS power = 3579 MWt* 1.02 (uncertainty) = 3650.6 MWt.

B. Average RCS temperature = 588.4°F.

C. RCS pressure = 2250 psia - 50 psia (uncertainty) = 2200.

D. Thermal design flow = 374400 gal/min.

E. Pressurizer level = 62% (includes uncertainty).

F. SG tube plugging = 10%.

G. Auxiliary feed flow = 2110 gal/min for overfill analysis, 1746 gal/min for dose analysis.

H. Auxiliary feed flow delay time = 90 seconds for dose analysis, 30 seconds for overfill analysis.

The limiting single failure for offsite doses was assumed to be the failure of the PORV on the ruptured steam generator. Failure of this PORV in the open position will cause an uncontrolled depressurization of the ruptured steam generator which will

increase primary to secondary leakage and the mass release to the atmosphere. It was assumed that the ruptured steam generator PORV fails open when the ruptured steam generator is isolated, and that the PORV is isolated by locally closing the associated block valve.

The offsite radiological analysis also assumed that one of the intact steam generator ARVs was inoperable. The analysis assumed that only two of the three intact steam generator ARVs were available for the cooldown.

The major operator actions required for the recovery from an SGTR are discussed in paragraph 15.6.3.1, and these operator actions were simulated in the analysis. The operator action times used for the analysis were established in reference 2 and are presented in table 15.6.3-1. It is noted that the PORV on the ruptured steam generator was assumed to fail open at the time the ruptured steam generator was isolated. Before proceeding with the recovery operations, the failed-open PORV on the ruptured steam generator was assumed to be isolated by locally closing the associated block valve. It was assumed that the ruptured steam generator PORV is isolated at 16 minutes after the valve was assumed to fail open. After the ruptured steam generator PORV was isolated, an additional delay time of 9 minutes (table 15.6.3-1) was assumed for the operator action time to initiate the RCS cooldown.

Transient Description

The LOFTTR2 analysis results are described below. The sequence of events for this transient is presented in table 15.6.3-2.

Following the tube rupture, reactor coolant flows from the primary into the secondary side of the ruptured steam generator since the primary pressure is greater than the steam generator pressure. In response to this loss of reactor coolant, pressurizer level decreases as shown in figure 15.6.3-1. The RCS pressure also decreases as shown in figure 15.6.3-2, as the steam bubble in the pressurizer expands. As the RCS pressure decreases due to the continued primary to secondary leakage, automatic reactor trip occurs on an OTΔT trip signal.

After reactor trip, core power rapidly decreases to decay heat levels. The turbine stop valves close, and steam flow to the turbine is terminated. The steam dump system is designed to actuate following reactor trip to limit the increase in secondary pressure, but the steam dump valves remain closed due to the loss of condenser vacuum resulting from the assumed loss of offsite power at the time of reactor trip. Thus, the energy

transfer from the primary system causes the secondary side pressure to increase rapidly after reactor trip until the steam generator PORVs (and safety valves if their setpoints are reached) lift to dissipate the energy, as shown in figure 15.6.3-3. The main feedwater flow will be terminated and AFW flow will be automatically initiated following reactor trip and the loss of offsite power.

The RCS pressure decreases more rapidly after reactor trip as energy transfer to the secondary shrinks the reactor coolant and the tube rupture break flow continues to deplete primary inventory. Pressurizer level also decreases more rapidly following reactor trip. The decrease in RCS inventory results in a low pressurizer pressure SI signal. After SI actuation, the SI flowrate exceeds the tube rupture break flowrate, and the pressurizer level begins to increase. This also results in an increase in the RCS pressure which trends toward the equilibrium value where the SI flowrate equals the break flowrate.

Since offsite power is assumed lost at reactor trip, the RCPs trip, and a gradual transition to natural circulation flow occurs. Immediately following reactor trip, the temperature differential across the core decreases as core power decays (see figures 15.6.3-4 and 15.6.3-5); however, the temperature differential subsequently increases as natural circulation flow develops. The cold leg temperatures trend toward the steam generator temperature as the fluid residence time in the tube region increase. The intact steam generator loop temperatures continue to slowly decrease due to the continued AFW flow until operator actions are taken to control the AFW flow to maintain the specified level in the intact steam generators. The ruptured steam generator loop temperatures also continue to slowly decrease until the ruptured steam generator was isolated and the PORV was assumed to fail open.

Major Operator Actions

1. Identify and Isolate the Ruptured Steam Generator

   The ruptured steam generator was assumed to be identified and the MSIV isolated at 20 minutes after the initiation of the SGTR or when the narrow range level recovers to 33 percent, whichever time is greater. However, at-power testing at VEGP with steam generator narrow range lower level tap relocation has shown that the steam generator narrow range level will not drop below 33 percent following a reactor trip. Therefore, it was conservatively assumed that the ruptured steam generator is isolated at 20 minutes. Isolation of AFW flow to the faulted steam generator takes place 7 minutes after event initiation. The ruptured steam generator PORV was assumed to fail open at the time of MSIV isolation, and the failure was simulated at 1202 seconds. The failure causes the ruptured steam generator to rapidly

depressurize as shown in figure 15.6.3-3, which results in an increase in primary to secondary leakage. The depressurization of the ruptured steam generator increases the break flow and energy transfer from primary to secondary, which results in a decrease in the ruptured loop temperatures, as shown in figure 15.6.3-5. As noted previously, the intact steam generator loop temperatures also decrease, as shown in figure 15.6.3-4, until the AFW flow to the intact steam generators is throttled. After this time, the heat transfer to the intact steam generators decreases and the temperature differential across the intact steam generators decreases. The decrease in the RCS temperatures results in an initial decrease in the pressurizer level and RCS pressure. However, the increased SI flow subsequently causes the pressurizer level and RCS pressure to increase again, as shown in figures 15.6.3-1 and 15.6.3-2, respectively. It was assumed that the time required for the operator to identify that the ruptured steam generator PORV is open and to locally close the associated block valve is 16 minutes. Thus, at 2162 seconds the depressurization of the ruptured steam generator was terminated.

2. Cool Down the RCS to establish Subcooling Margin

After the ruptured steam generator PORV block valve was closed, a 9-minute operator action time was imposed prior to initiation of cooldown. The depressurization of the ruptured steam generator affects the RCS cooldown target temperature since the temperature is dependent upon the pressure in the ruptured steam generator. Since offsite power was lost, the RCS was cooled by dumping steam to the atmosphere using the two available, intact steam generator PORVs. The cooldown was continued until RCS subcooling at the ruptured steam generator pressure was 20°F plus an allowance of 24°F for instrument uncertainty. Because of the lower pressure in the ruptured steam generator, the associated temperature the RCS must be cooled to is also lower, which has the net effect of extending the time for cooldown. The cooldown was initiated at 2702 seconds and was completed at 3498 seconds.

The reduction in the intact steam generator pressures required to accomplish the cooldown is shown in figure 15.6.3-3, and the effect of the cooldown on the RCS temperature is shown in figure 15.6.3-4. The pressurizer level and RCS pressure also decrease during this cooldown process due to shrinkage of the reactor coolant, as shown in figures 15.6.3-1 and 15.6.3-2.

3. Depressurize to Restore Inventory

After the RCS cooldown, a 5-minute operator action time was assumed prior to depressurization. The RCS was depressurized at 3902 seconds to assure adequate coolant

inventory prior to terminating SI flow. With the RCPS stopped, normal pressurizer spray is not available and, thus, the RCS was depressurized by opening a pressurizer PORV. The depressurization was continued until any of the following conditions were satisfied: RCS pressure is less than the ruptured steam generator pressure and pressurizer level is greater than the allowance of 9 percent or pressurizer level uncertainty, or pressurizer level is greater than 69 percent, or RCS subcooling is less than the 24-°F allowance for subcooling uncertainty. The effect of the RCS depressurization on the RCS pressure and the differential pressure between the RCS and the ruptured steam generator is shown in figures 15.6.3-2 and 15.6.3-6. The RCS depressurization reduces the break flow as shown in figure 15.6.3-7 and increases SI flow to refill the pressurizer as shown in figure 15.6.3-1.

4. Terminate SI to Stop Primary to Secondary Leakage

The previous actions have established adequate RCS subcooling, verified a secondary side heat sink, and restored the reactor coolant inventory to ensure that SI flow is no longer needed. When these actions have been completed, the SI flow must be stopped to prevent repressurization of the RCS and to terminate primary to secondary leakage. The SI flow is terminated at this time if RCS subcooling is greater than the 24°F allowance for uncertainty, minimum AFW flow is available or at least one intact steam generator level is in the narrow range, the RCS pressure is increasing, and the pressurizer level is greater than the 9-percent allowance for uncertainty. To assure that the RCS pressure is increasing, SI was not terminated until the RCS pressure increases by at least 50 psi.

After depressurization was completed, an operator action time of 3 minutes was assumed prior to SI termination. Since the above requirements are satisfied, SI termination was performed at this time. After SI termination, the RCS pressure decreases as shown in figure 15.6.3-2. The differential pressure between the RCS and the ruptured steam generator also decreases as shown in figure 15.6.3-6. Figure 15.6.3-7 shows that the primary to secondary leakage continues after the SI flow is stopped until the RCS and ruptured steam generator pressures equalize.

The ruptured steam generator water volume is shown in figure 15.6.3-8. It is noted that the water volume in the ruptured steam generator is significantly less than the total steam generator volume of 5904 ft^3 when the break flow is terminated. The mass of water in the ruptured steam generator is also shown as a function of time in figure 15.6.3-9.

1. Design Basis Analytical Assumptions

The major assumptions and parameters used in the analysis are itemized in table 15.6.3-4.

2. Source Term Calculations

The radionuclide concentrations in the primary and secondary system, prior to and following the SGTR, are determined as follows:

a. The iodine concentrations in the reactor coolant will be based upon preaccident and accident-initiated iodine spikes.

   i. Accident-Initiated Spike - The initial primary coolant iodine concentration is 1 μCi/g of Dose Equivalent (D.E.) I-131. Following the primary system depressurization associated with the SGTR, an iodine spike is initiated in the primary system which increases the iodine release rate from the fuel to the coolant to a value 500 times greater than the release rate corresponding to the initial primary system iodine concentration.

   ii. Preaccident Spike - A reactor transient has occurred prior to the SGTR and has raised the primary coolant iodine concentration from 1 to 60 μCi/gram of D.E. I-131.

b. The initial secondary coolant iodine concentration is 0.1 μCi/gram of D.E. I-131.

c. The chemical form of iodine in the primary and secondary coolant is assumed to be elemental.

d. The initial noble gas concentrations in the reactor coolant are based upon 1-percent fuel defects.

3. Dose Calculations

The iodine transport model utilized in this analysis was proposed by Postma and Tam (reference 3). The model considers break flow flashing, steaming, and partitioning. The model assumes that a fraction of the iodine carried by the break flow becomes airborne immediately due to flashing and atomization. The fraction of primary coolant iodine which is not assumed to become airborne immediately mixes with the secondary water and is assumed to become airborne at a rate proportional to the steaming rate and the iodine partition coefficient. This analysis conservatively assumes an iodine partition coefficient of 100 between the

steam generator liquid and steam phases. Droplet removal by the dryers is conservatively assumed to be negligible. The iodine transport model is illustrated in figure 15.6.3-12.

The offsite radiological analysis did not consider steam generator tube uncovery in the calculation, since the steam generator tube uncovery issue was investigated and closed by the Westinghouse Owners Group (WOG). Reference 7 documents the Westinghouse position on the issue that the affect of tube uncovery on the limiting SGTR transient is essentially negligible and need not be considered in the analysis. Reference 8 documents the NRC agreement on this issue.

The following assumptions and parameters were used to calculate the activity released to the atmosphere and the offsite doses following an SGTR.

a. The mass of reactor coolant discharged into the secondary system through the rupture and the mass of steam released from the ruptured and intact steam generators to the atmosphere are presented in table 15.6.3-3.

b. The time dependent fraction of rupture flow that flashes to steam and is immediately released to the environment is presented in figure 15.6.3-13. The break flow flashing fraction was conservatively calculated assuming that 100 percent of the break flow comes from the hot leg side of the steam generator, whereas the break flow actually comes from both the hot leg and the cold leg sides of the steam generator.

c. The total primary to secondary leak rate is assumed to be 1.0 gal/min as allowed by the Technical Specifications. The leak rate is assumed to be 0.70 gal/min to the three intact steam generators and 0.30 gal/min to the ruptured steam generator.

d. The iodine partition factor between the liquid and steam of the ruptured and intact steam generators is assumed to be 100.

e. No credit was taken for radioactive decay during release and transport, or for cloud depletion by ground deposition during transport to the site boundary or outer boundary of the low population zone.

f. Short-term atmospheric dispersion factors (χ/Qs) for accident analysis and breathing rates are provided in table 15.6.3-8. The breathing rates were obtained from NRC Regulatory Guide 1.4, (reference 4).

4. Offsite Thyroid Dose Calculation Model

Offsite thyroid doses are calculated using the equation:

$$D_{Th} = \sum_i \left[DCF_i \left(\sum_j (IAR)_{ij} \quad (BR)_j \quad (\chi/Q)_j \right) \right]$$

where:

$(IAR)_{ij}$ = integrated activity of iodine nuclide i released during the time interval j in Ci[(a)]

$(BR)_j$ = breathing rate during time interval j in meter^3/second (table 15.6.3-8)

$(\chi/Q)_j$ = atmospheric dispersion factor during time interval j in second/meter^3 (table 15.6.3-8)

$(DCF)_i$ = thyroid dose conversion factor via inhalation for iodine nuclide i in rem/Ci (table 15.6.3-9)

D_{Th} = Thyroid dose via inhalation in rem

Offsite whole-body gamma doses are calculated using the equation:

$$D\gamma = 0.25 \sum_i \left[\bar{E}\gamma_i \left(\sum_j (IAR)_{ij} \quad (\chi/Q)_j \right) \right]$$

Where:

$(IAR)_{ij}$ = integrated activity of noble gas nuclide i released during time interval j in Ci[(a)]

$(\chi/Q)_j$ = atmospheric dispersion factor during time interval j in seconds/m^3

$\bar{E}\gamma_i$ = average gamma energy for noble gas nuclide i in Mev/dis (table 15.6.3-10)

$D\gamma$ = whole body gamma dose due to immersion in rem

a. No credit is taken for cloud depletion by ground deposition or by radioactive decay during transport to the exclusion area boundary or to the outer boundary of the low-population zone.

Offsite beta-skin doses are calculated using the equation:

$$D\beta = 0.23 \sum_i \left[\overline{E}\gamma_i \left(\sum_j (IAR)_{ij} \quad (\chi/Q)_j \right) \right]$$

where:

$(IAR)_{ij}$ = integrated activity of noble gas nuclide i released during time interval j in Ci[a]

$(\chi/Q)_j$ = atmospheric dispersion factor during time interval j in seconds/m^3

$\overline{E}\beta_i$ = average beta energy for noble gas nuclide i in Mev/dis (table 15.6.3-10)

$D\beta$ = beta-skin dose due to immersion in rem

5. Results

Thyroid, whole-body gamma, and beta-skin doses at the Exclusion Area Boundary and Low Population Zone are presented in table 15.6.3-11. All doses are well within the allowable guidelines as specified by Standard Review Plan 15.6.3 and 10 CFR 100.

a. No credit is taken for cloud depletion by ground deposition or by radioactive decay during transport to the exclusion area boundary or to the outer boundary of the low-population zone.

REFERENCES

1. Westinghouse Electric Company, "SGTR Reanalysis Results for Revised Operator Action Times SECL 98-124 Rev. 1." Letter from J. L. Tain to J. B. Beasley dated July 20, 1999, GP-16989.

2. Southern Nuclear Operating Company, Inc., ELV-06244-A, memo from L. A. Ward (Southern Nuclear) to J. L. Tain (W) dated August 18, 1998.

3. Postma, A. K. and Tam, P. S., "Iodine Behavior in a PWR Cooling System Following a Postulated Steam Generator Tube Rupture," NUREG-0409.

4. NRC Regulatory Guide 1.4, Rev. 2, "Assumptions Used for Evaluating the Potential Radiological Consequences of a LOCA for Pressurized Water Reactors," June 1974.

5. International Commission on Radiological Protection, "Limits for Intakes of Radionuclides by Workers," ICRP Publication 30, Volume 3 No. 1-4, 1979.

6. Bell, M. J., "ORIGEN - The ORNL Isotope Conversion and Depletion Code," ORNL-4628, 1973.

7. WOG-92-25, "Westinghouse Owners Group Steam Generator Tube Uncovery Issue," March 31, 1992.

8. Attachment to WOG-93-066, letter to Lawrence A. Walsh (Chairman WOG) from Robert C. Jones (NRC), "Westinghouse Owners Group - Steam Generator Tube Uncovery Issue," March 10, 1993.

TABLE 15.6.3-1

OPERATOR ACTION TIMES FOR DESIGN BASIS SGTR ANALYSIS

Action	Time (min)
Identify and isolate AFW flow to ruptured steam generator	7
Isolate ruptured steam generator	20 min or LOFTTR2 calculated time to recover to 33-percent narrow range level in the ruptured SG, whichever is longer[a]
Operator action time to initiate cooldown	19 (overfill analysis) 9 (dose analysis)
Cooldown	Calculated by LOFTTR2
Operator action time to initiate depressurization	5
Depressurization	Calculated by LOFTTR2
Operator action time to initiate SI termination	3
SI termination and pressure equalization	Calculated time for SI termination and equalization of RCS and ruptured SG pressures

a. At-power testing at VEGP with steam generator narrow range lower level tap relocation has shown that the steam generator narrow range level will not drop below 33 percent following a reactor trip. Therefore, it was conservatively assumed that the ruptured SG is isolated at 20 minutes.

TABLE 15.6.3-2

SEQUENCE OF EVENTS

Event	Time (s)
SG tube rupture	0
Reactor trip	43.8
SI actuated	357
AFW flow isolated to ruptured steam generator	420
Ruptured SG isolated	1200
Ruptured SG PORV fails open	1202
Ruptured SG PORV block valve closed	2162
RCS cooldown initiated	2702
RCS cooldown terminated	3498
RCS depressurization initiated	3800
RCS depressurization terminated	3902
SI terminated	4082
Break flow terminated	5412

TABLE 15.6.3-3

MASS RELEASES RESULTS
TOTAL MASS FLOW (lb)

	0 - 2 h	2 - 8 h
Ruptured SG		
- Condenser	49,900	0
- Atmosphere	129,000	42,400
- Feedwater	75,400	0
Intact SGs		
- Condenser	148,400	0
- Atmosphere	530,900	1,071,400
- Feedwater	747,200	1,164,300
Break Flow	206,200	0

TABLE 15.6.3-4 (SHEET 2 OF 3)

F. Initial steam generator mass (each), grams	4.2×10^7
G. Offsite power	Lost at time of reactor trip
H. Primary-to-secondary leakage duration for intact SG, h	8
I. Species of iodine	100-percent elemental
II. Activity Release Data	
A. Ruptured steam generator	
1. Rupture flow	See table 15.6.3-3
2. Rupture flow flashing fraction	See figure 15.6.3-13
3. Iodine scrubbing efficiency	See figure 15.6.3-15
4. Total steam release, lb	See table 15.6.3-3
5. Iodine partition factor	100
B. Intact steam generators	
1. Total primary-to-secondary leakage, gal/min	0.7
2. Total steam release, lb	See table 15.6.3-3
3. Iodine partition factor	100

TABLE 15.6.3-5

IODINE SPECIFIC ACTIVITIES
IN THE PRIMARY AND SECONDARY COOLANT
BASED ON 1, 60, AND 0.1 μCi/gram OF D.E. I-131

	Specific Activity (μ Ci/gm)		
	Primary Coolant		Secondary Coolant
Nuclide	1 μCi/gm	60 μCi/gm	0.1 μCi/gm
I-131	0.78	46.8	0.078
I-132	0.78	46.8	0.078
I-133	1.17	70.2	0.012
I-134	0.16	9.6	0.016
I-135	0.64	38.4	0.064

TABLE 15.6.3-6

IODINE SPIKE APPEARANCE RATES
(CURIES/SECOND)

I-131	I-132	I-133	I-134	I-135
1.7	9.2	3.7	4.4	3.5

TABLE 15.6.3-7

NOBLE GAS SPECIFIC ACTIVITIES IN THE REACTOR COOLANT BASED ON 1% FUEL DEFECTS

Gas	μCi/g
Xe-131m	2.20
Xe-133	270.00
Xe-135m	0.48
Xe-135	7.20
Xe-138	0.64
Kr-85m	2.00
Kr-85	7.30
Kr-87	1.30
Kr-88	3.60

TABLE 15.6.3-9

THYROID DOSE CONVERSION FACTORS
(rem/curie) (ref. 5)

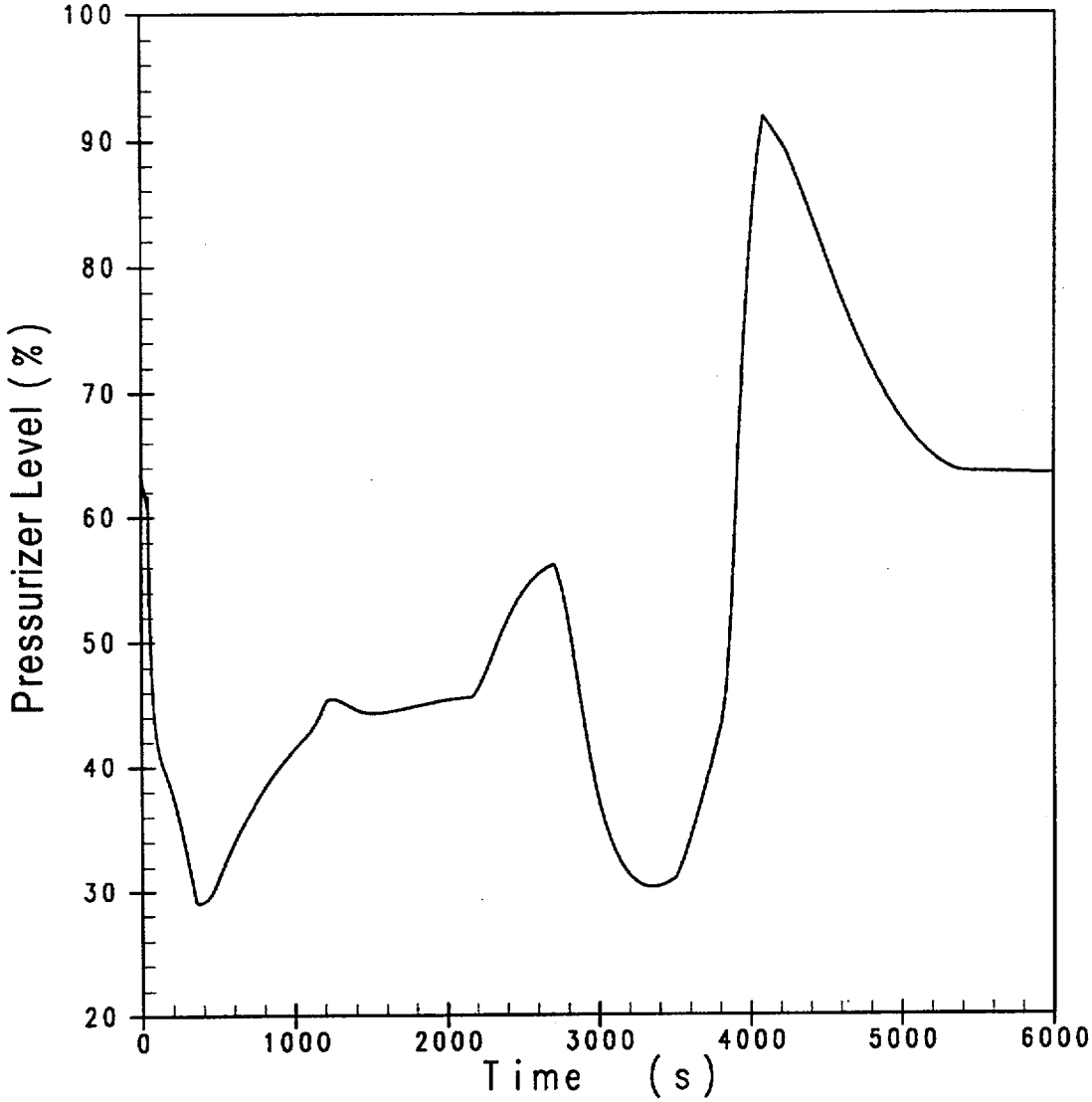
Nuclide	Conversion factor
I-131	1.07×10^6
I-132	6.29×10^3
I-133	1.81×10^5
I-134	1.07×10^3
I-135	3.14×10^4

TABLE 15.6.3-11

OFFSITE RADIATION DOSES

	Doses (rem)	
	Calculated Value	Allowable Guideline Value
1. Accident Initiated Iodine Spike		
Exclusion Area Boundary (0-2 h) Thyroid Dose	6.0	30
Low Population Zone (0-8 h) Thyroid Dose	3.0	30
2. Preaccident Iodine Spike		
Exclusion Area Boundary (0-2 h) Thyroid Dose	18.0	300
Low Population Zone (0-8 h) Thyroid Dose	8.0	300
3. Whole-Body Gamma and Beta-Skin Dose		
Exclusion Area Boundary (0-2 h) Whole-Body Gamma Dose	0.15	2.5
Beta-Skin Dose	0.20	not specified
Low Population Zone (0-8 h) Whole-Body Gamma Dose	0.05	2.5
Beta-Skin Dose	0.10	not specified

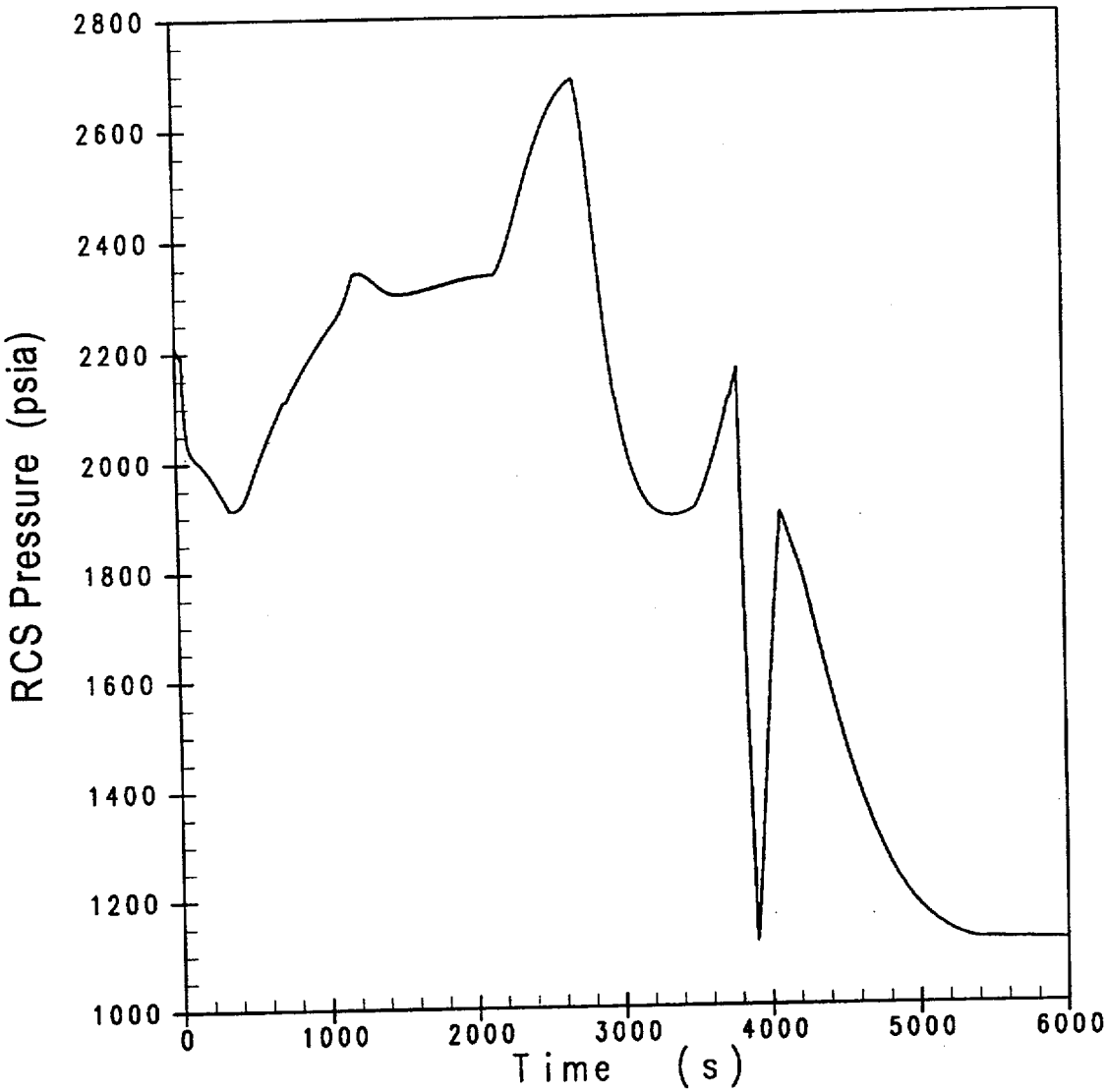
STEAM GENERATOR TUBE RUPTURE
PRESSURIZER LEVEL



REV 9 5/00
REV 3 12/92

VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2	PRESSURIZER LEVEL
	FIGURE 15.6.3-1

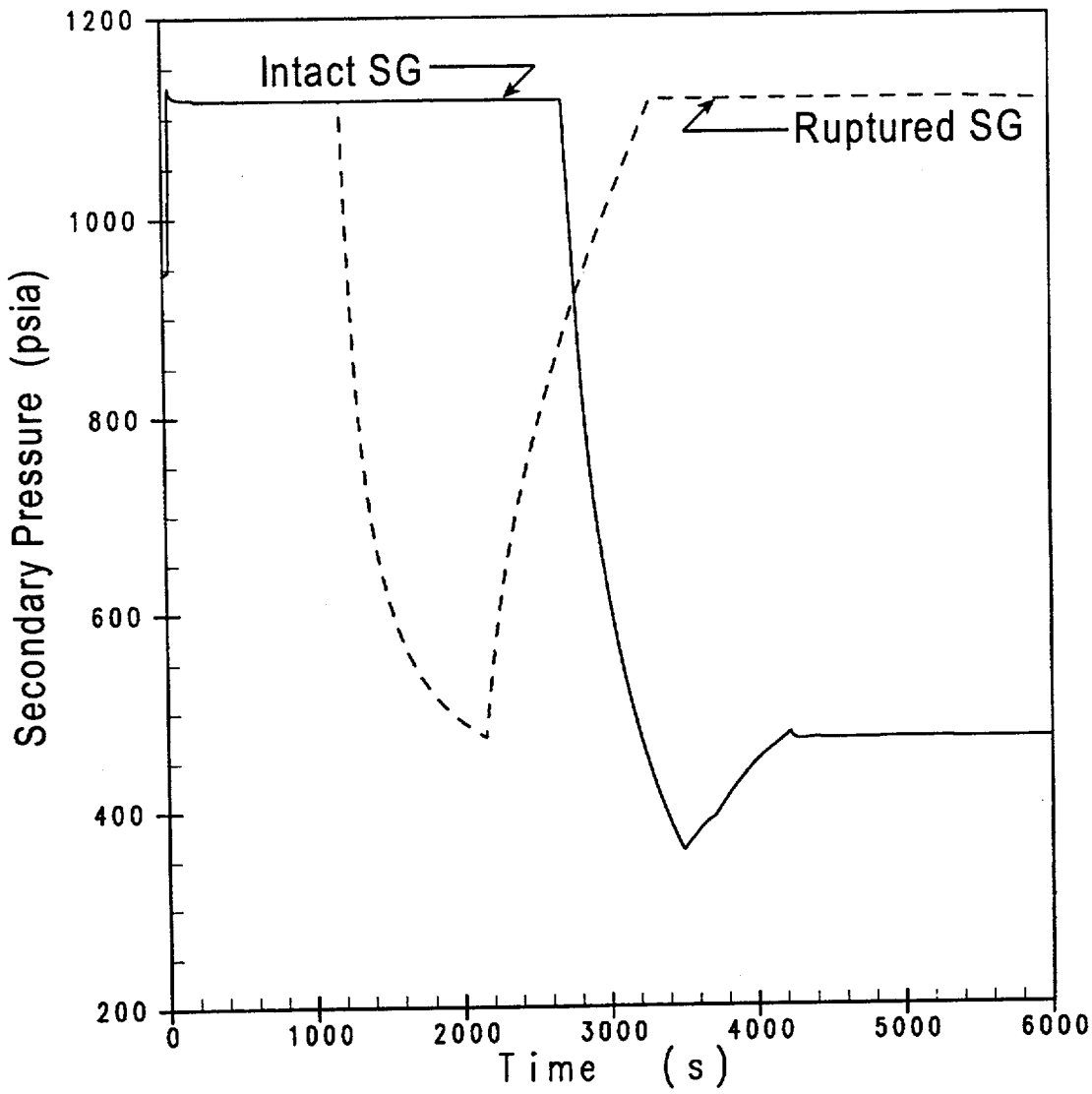
STEAM GENERATOR TUBE RUPTURE
RCS PRESSURE



REV 9 5/00
REV 3 12/92

VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2	RCS PRESSURE FIGURE 15.6.3-2

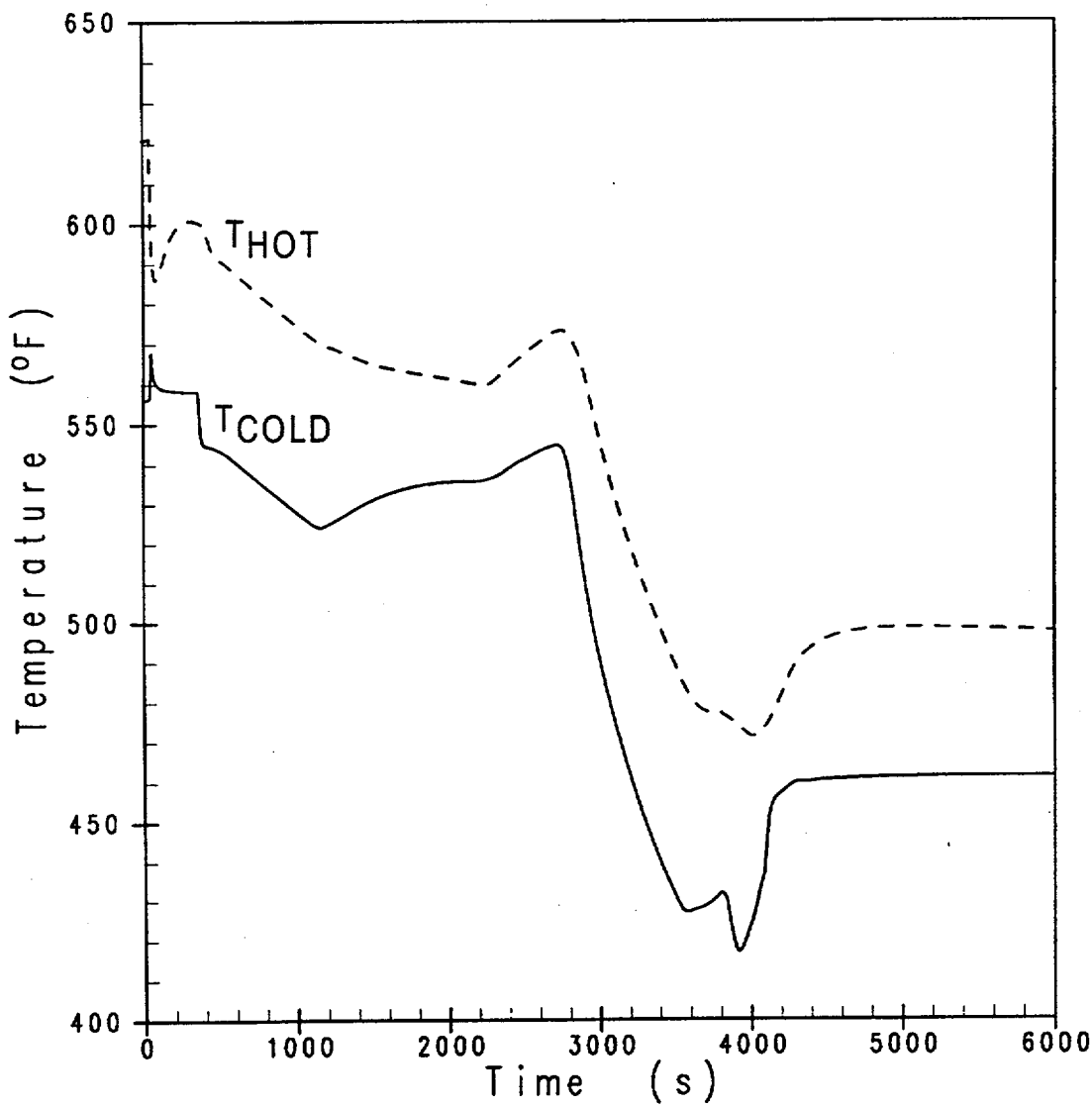
STEAM GENERATOR TUBE RUPTURE
SECONDARY PRESSURE



REV 9 5/00
REV 3 12/92

VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2	SECONDARY PRESSURE
	FIGURE 15.6.3-3

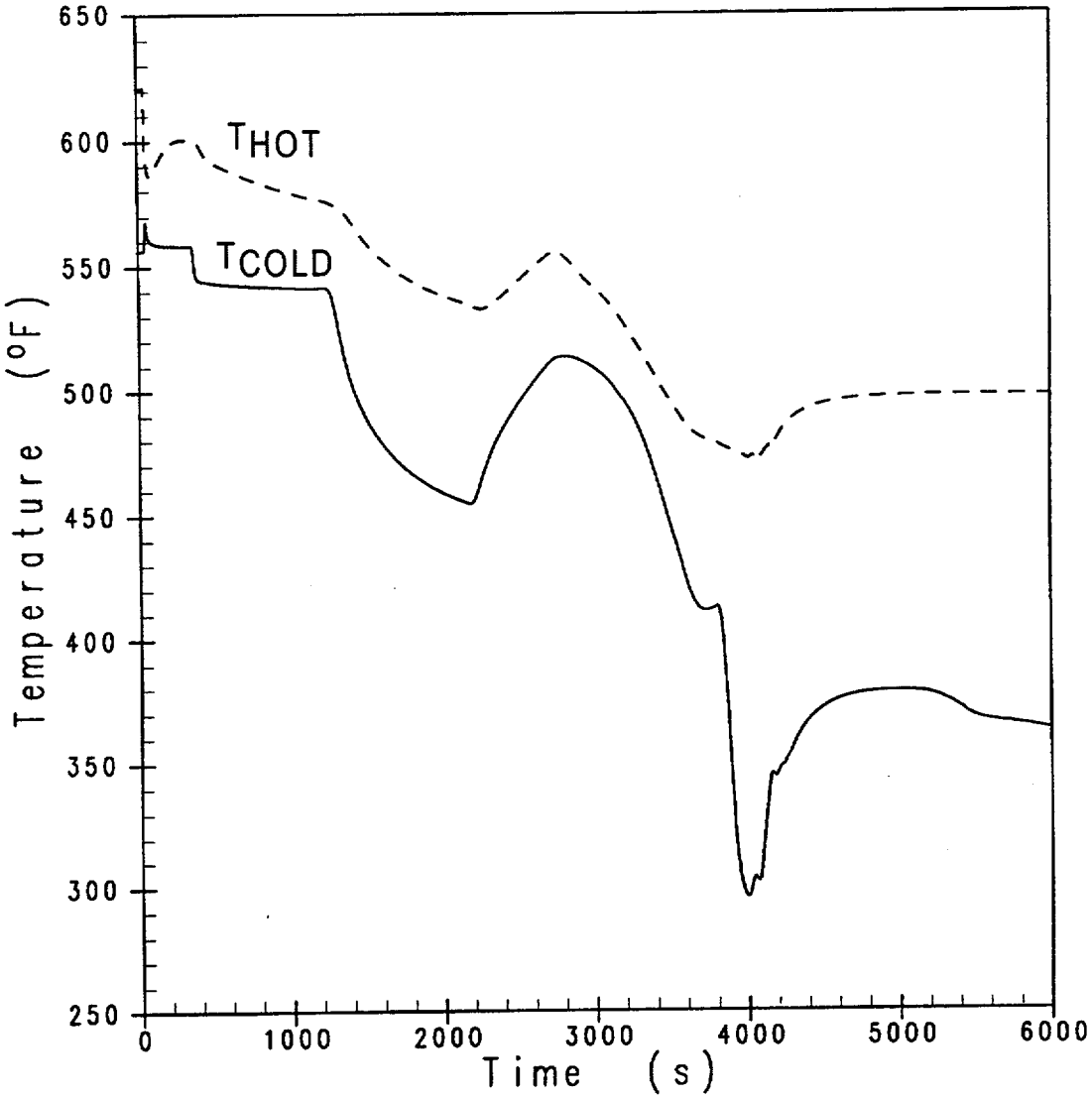
STEAM GENERATOR TUBE RUPTURE
INTACT LOOP HOT AND COLD LEG RCS TEMPERATURES



REV 9 5/00
REV 3 12/92

VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2	INTACT LOOP T_{HOT} AND T_{COLD}
	FIGURE 15.6.3-4

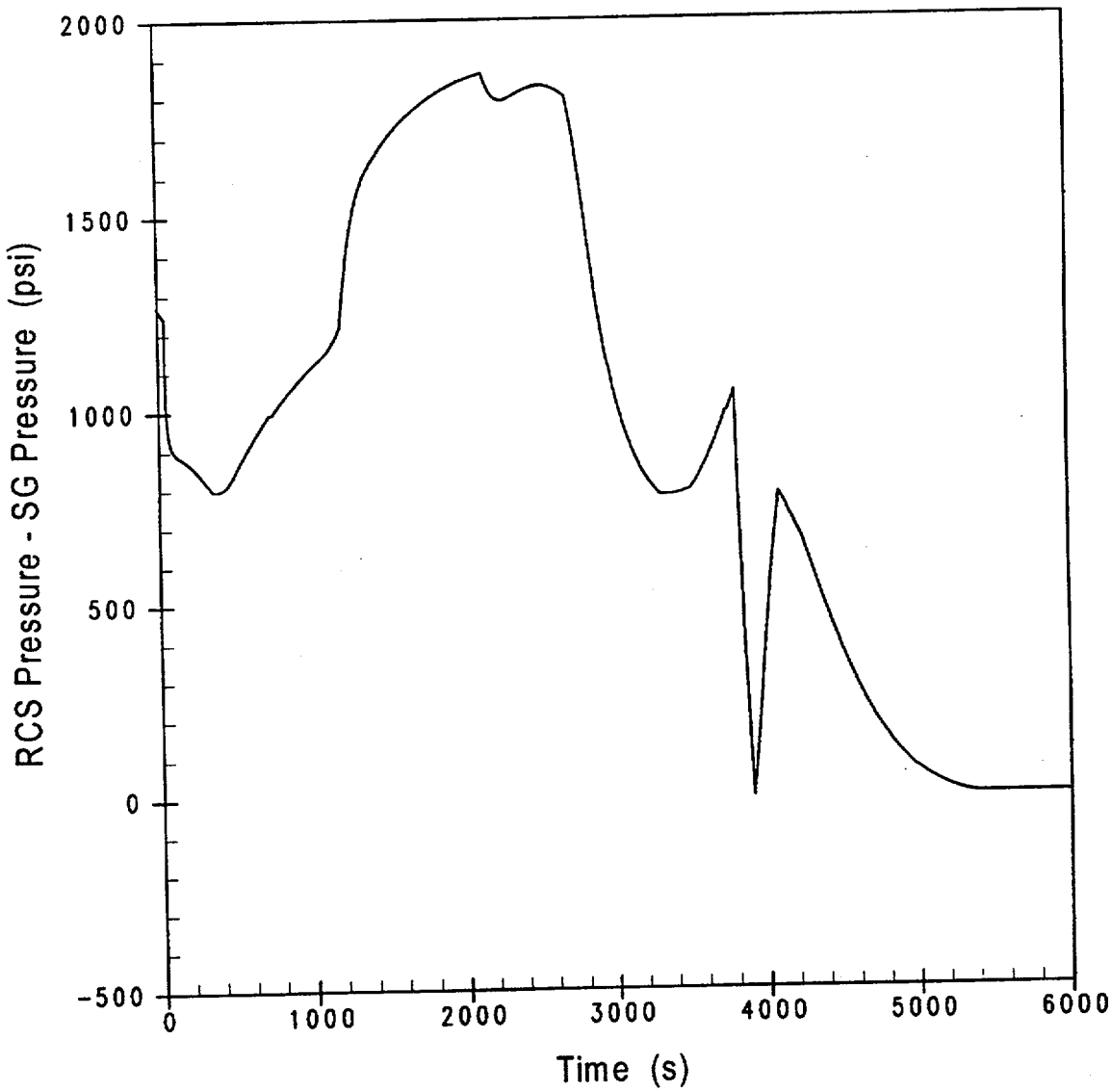
STEAM GENERATOR TUBE RUPTURE
RUPTURED LOOP HOT AND COLD LEG RCS TEMPERATURES



REV 9 5/00
REV 3 12/92

VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2	RUPTURED LOOP T_{HOT} AND T_{COLD}
	FIGURE 15.6.3-5

STEAM GENERATOR TUBE RUPTURE
DIFFERENTIAL PRESSURE BETWEEN RCS AND RUPTURED SG



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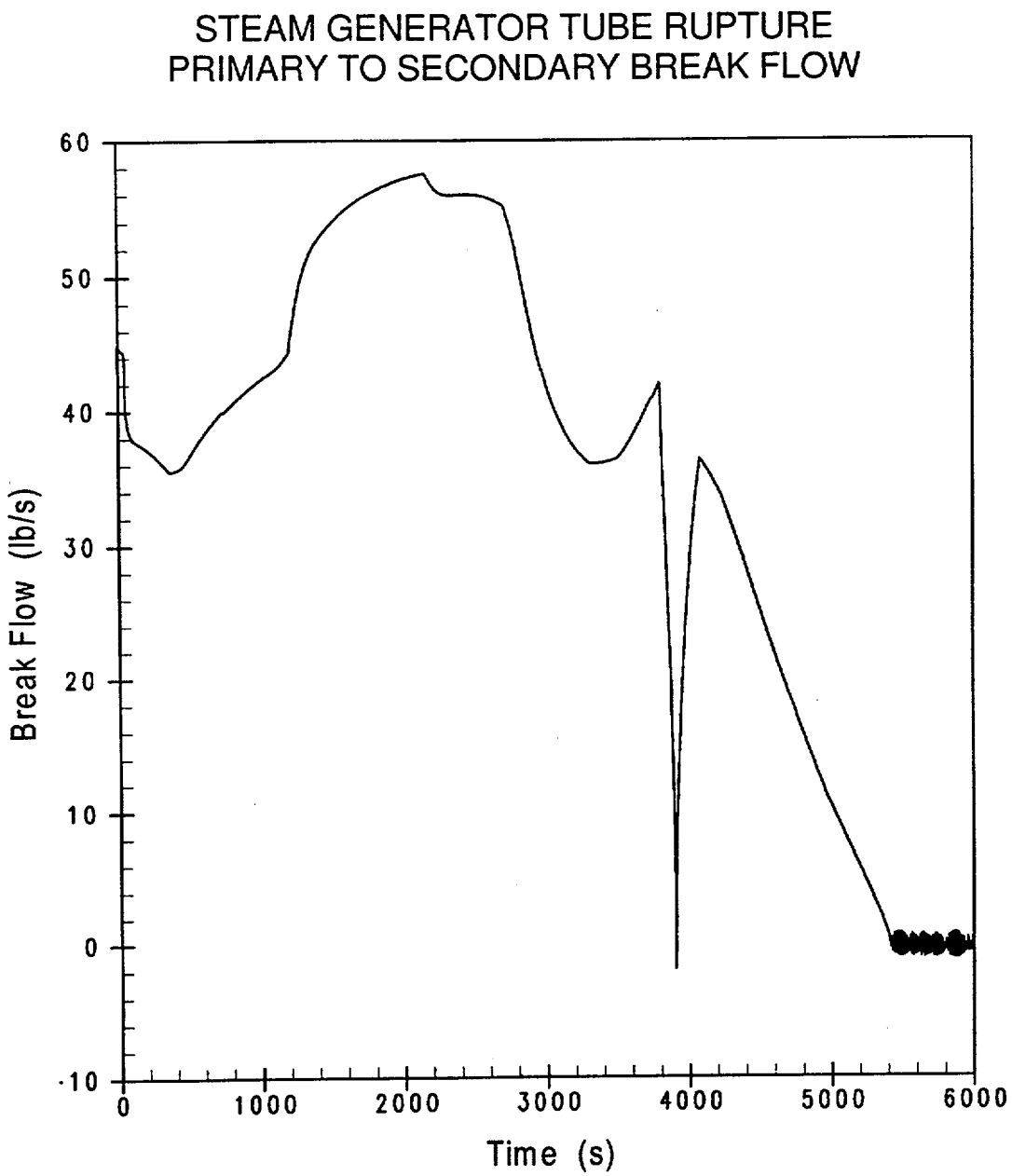
VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2	DIFFERENTIAL PRESSURE
	FIGURE 15.6.3-6



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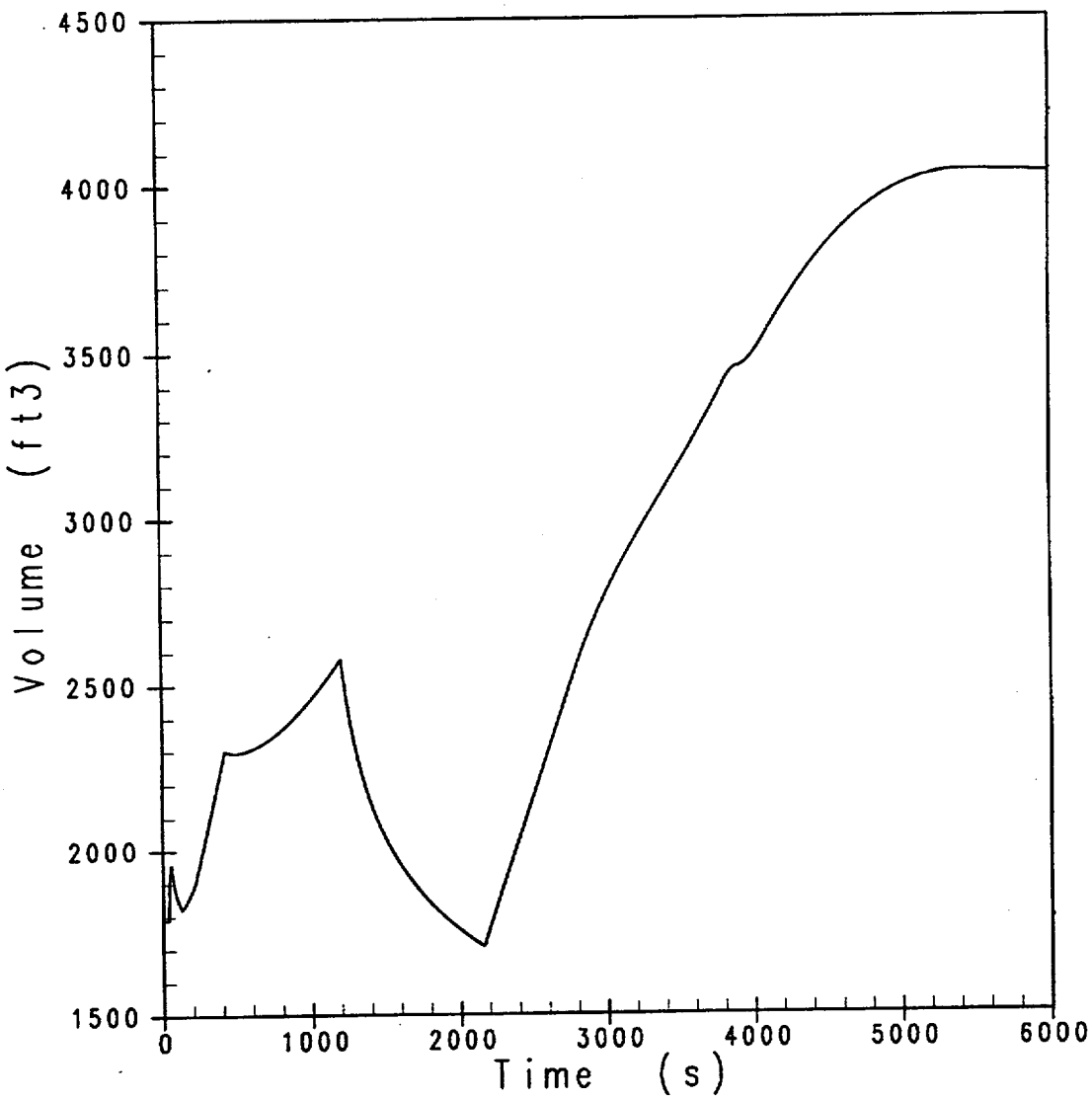
VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2	PRIMARY TO SECONDARY BREAK FLOW
	FIGURE 15.6.3-7

STEAM GENERATOR TUBE RUPTURE
RUPTURED SG WATER VOLUME



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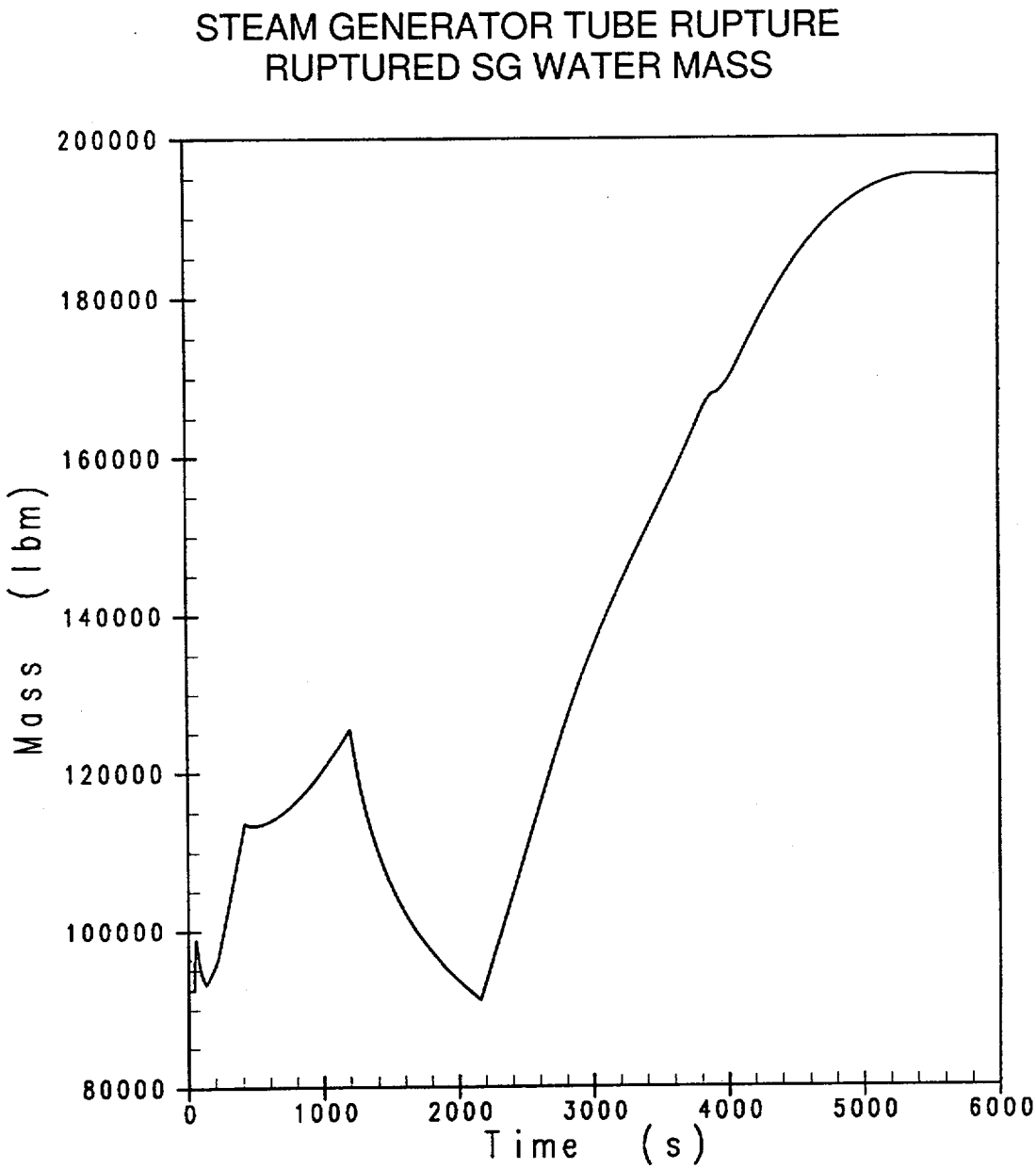
VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2	RUPTURED SG WATER VOLUME
	FIGURE 15.6.3-8



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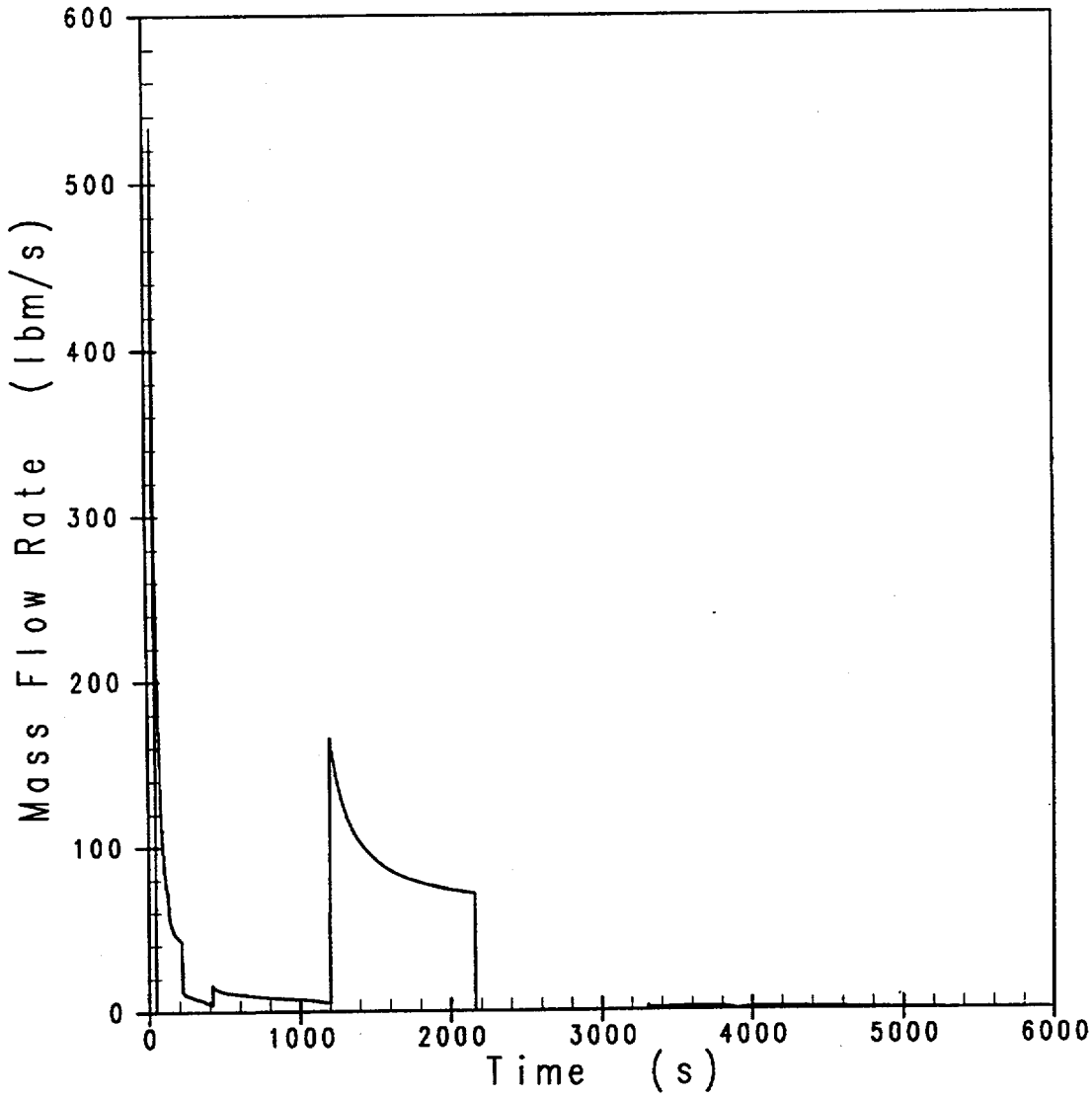
VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2	RUPTURED SG WATER MASS
	FIGURE 15.6.3-9

STEAM GENERATOR TUBE RUPTURE
RUPTURED SG ATMOSPHERIC MASS RELEASES



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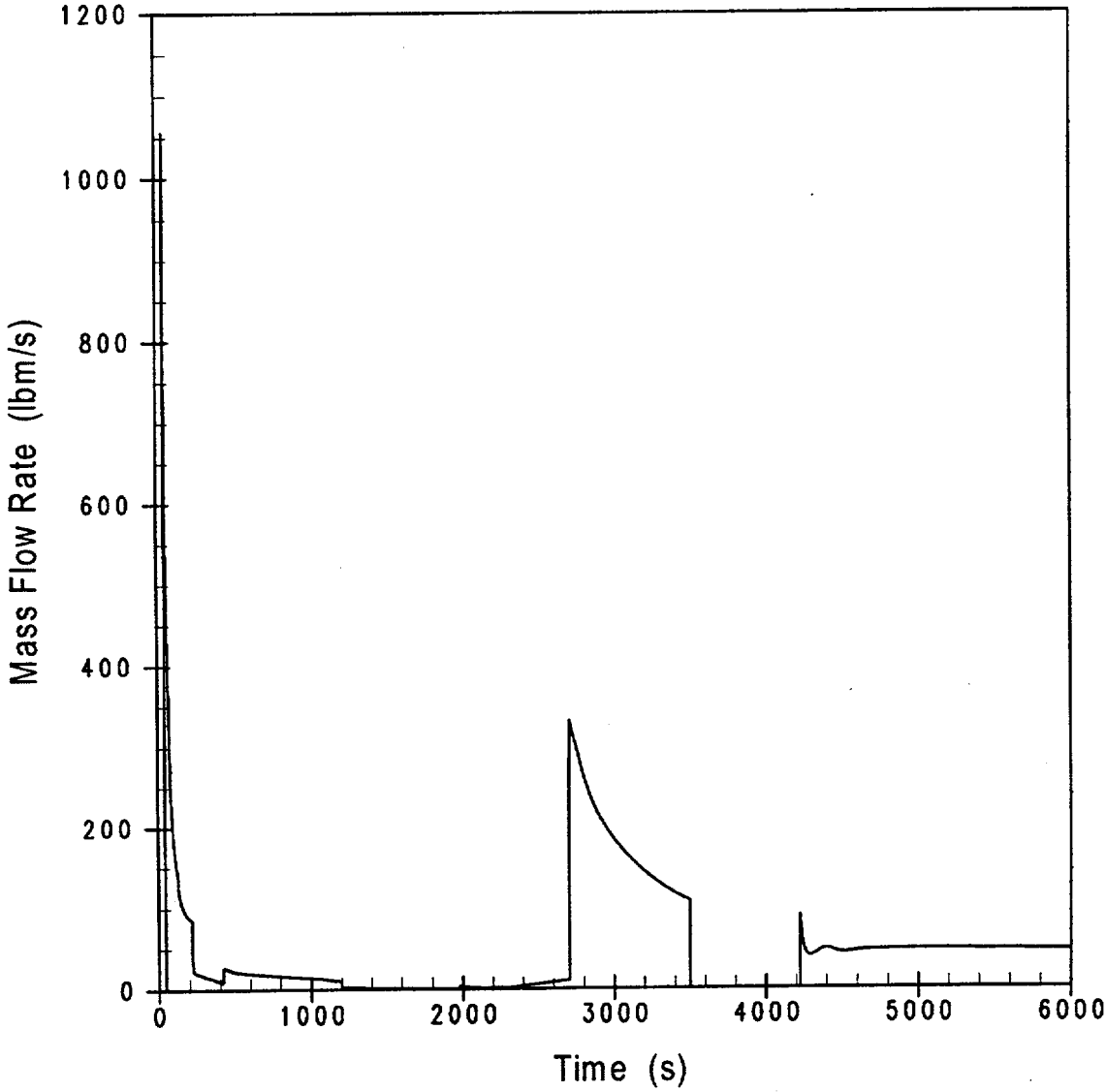
VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2	RUPTURED SG ATMOSPHERIC MASS RELEASES
	FIGURE 15.6.3-10

STEAM GENERATOR TUBE RUPTURE
INTACT SGS ATMOSPHERIC MASS RELEASES



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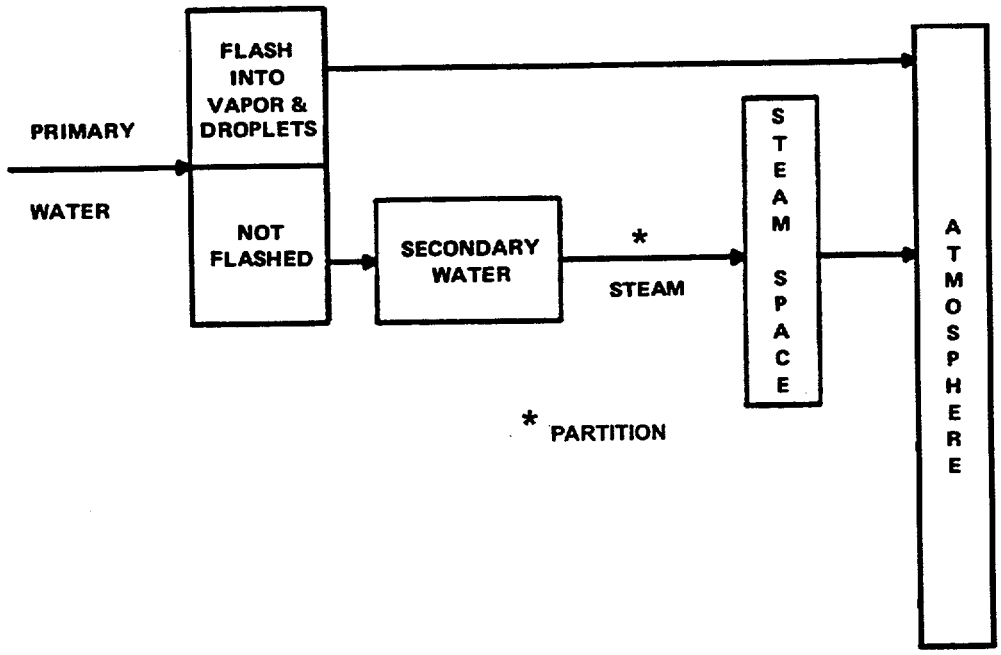
VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2	INTACT SG ATMOSPHERIC MASS RELEASES
	FIGURE 15.6.3-11



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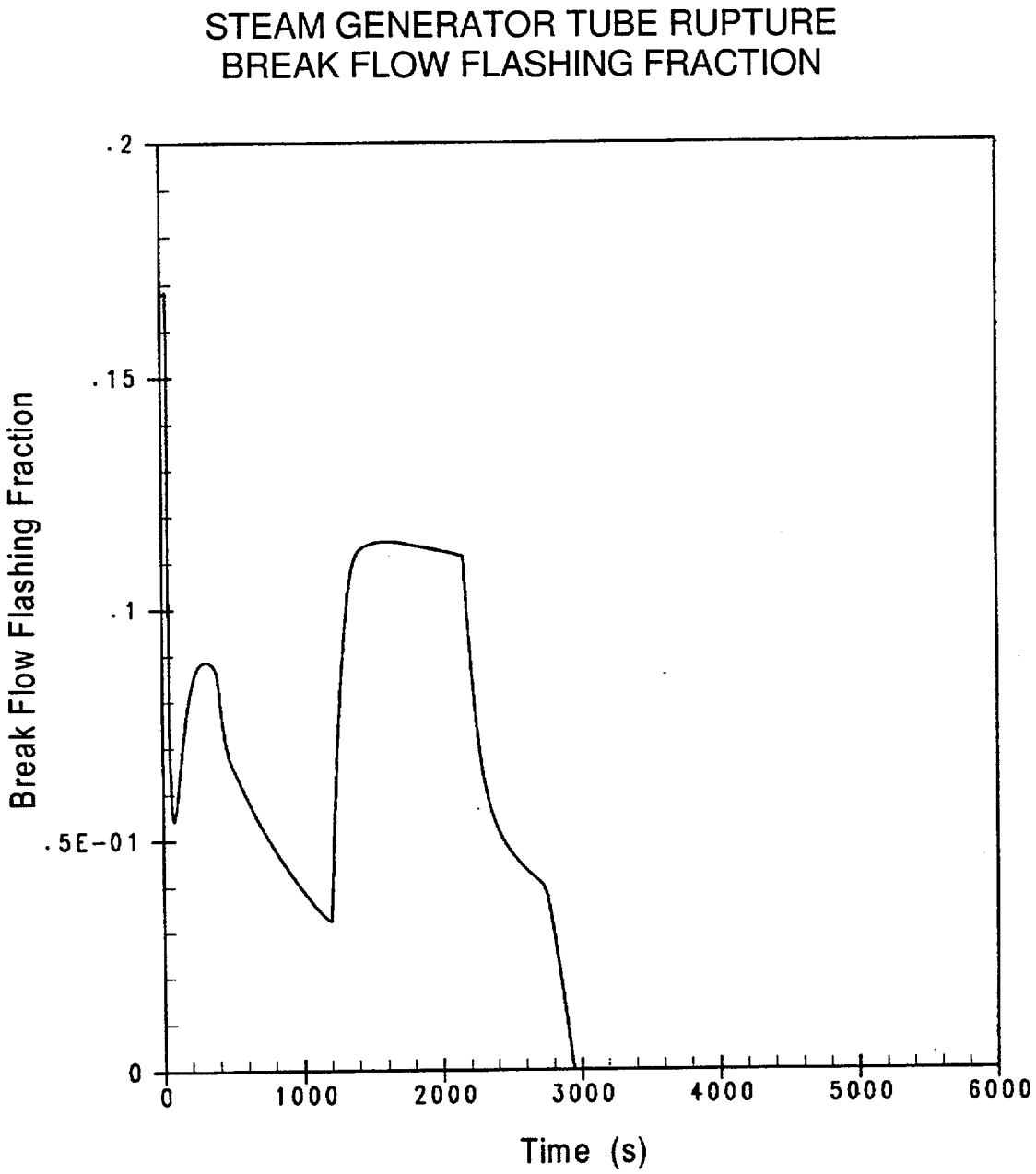
VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2	IODINE TRANSPORT MODEL
	FIGURE 15.6.3-12



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VOGTLE ELECTRIC GENERATING PLANT UNIT 1 AND UNIT 2	BREAK FLOW FLASHING FRACTION
	FIGURE 15.6.3-13

the beginning of reflood, the safety injection accumulator tanks rapidly discharge borated cooling water into the RCS, contributing to the filling of the reactor vessel downcomer. The downcomer water elevation head provides the driving force required for the reflooding of the reactor core. The safety injection pumps aid the filling of the downcomer and subsequently supply water to maintain a full downcomer and complete the reflooding process.

Continued operation of the ECCS pumps supplies water during long-term cooling. Core temperatures have been reduced to long-term, steady-state levels associated with dissipation of residual heat generation. After the water level of the refueling water storage tank reaches a minimum allowable value, coolant for long-term cooling of the core is obtained by semiautomatic switching to the cold leg recirculation mode of operation, in which spilled borated water is drawn from the containment emergency sumps by the residual heat removal pumps and returned to the RCS cold legs. The containment spray pumps are manually aligned to the containment emergency sumps and continue to operate to further reduce containment pressure.

Approximately 7.5 hours after initiation of the LOCA, the ECCS is realigned to supply water to the RCS hot legs to control the boric acid concentration in the reactor vessel. This time does not change due to the implementation of VANTAGE 5 or VANTAGE+ fuel.

15.6.5.2.2 Description of Small-Break LOCA Transient

As contrasted with the large break, the blowdown phase of the small break occurs over a longer time period. Thus, for the small-break LOCA there are only three characteristic stages; i.e., a gradual blowdown with a decrease in water level and a partial core uncovery, core recovery, and long-term recirculation.

Should a small break occur, depressurization of the reactor coolant system causes fluid to flow into the loops from the pressurizer resulting in a pressure and level decrease in the pressurizer. Reactor trip occurs when the low pressurizer pressure trip setpoint is reached. During the earlier part of the small-break transient, the effect of the break flow is not strong enough to overcome the flow maintained by the reactor coolant pumps through the core as they are coasting down following reactor trip. Upward flow through the core is maintained. However, the core flow is not sufficient to prevent a partial core uncovery. The ECCS is actuated when the appropriate setpoint is reached and provides sufficent core flow to recover the core.

Before the break occurs the plant is in an equilibrium condition; i.e., the heat generated in the core is being removed via the secondary system. During blowdown, heat from fission product decay, hot internals, and the vessel continues to be transferred to the reactor coolant system. The heat transfer between the reactor coolant system and the secondary system may be in either direction depending on the relative temperatures. In the case of continued heat addition to the secondary, secondary system pressure increases and steam relief via the atmospheric relief and/or safety valves may occur. Makeup to the secondary side is automatically provided by the auxiliary feedwater pumps. The safety injection signal isolates normal feedwater flow by closing the main feedwater isolation valves and initiates auxiliary feedwater flow by starting the auxiliary feedwater pumps. The secondary flow aids in the reduction of reactor coolant system pressure.

When the RCS depressurizes to approximately 615 psia, the cold leg accumulators begin to inject borated water into the reactor coolant loops. However, the vessel mixture level starts to increase to cover the fuel with ECCS pumped injection before the accumulator injection for most breaks. For all breaks, the accumulator injection provides enough water supply to bring the mixture level up to the upper plenum region where it is maintained. Due to the loss of offsite power assumption, the reactor coolant pumps are assumed to be tripped at the time of reactor trip during the accident and the effects of pump coastdown are included in the blowdown analyses.

15.6.5.3 Core and System Performance

15.6.5.3.1 Mathematical Model

The requirements of an acceptable ECCS evaluation model are presented in Appendix K of 10 CFR 50.[1]

15.6.5.3.1.1 Large-Break LOCA Evaluation Model. The analysis of a large-break LOCA transient is divided into three phases: blowdown, refill, and reflood. There are three distinct transients analyzed in each phase, including the thermal-hydraulic transient in the RCS, the pressure and temperature transient within the containment, and the fuel and clad temperature transient of the hottest fuel rod in the core. Based on these considerations, a system of interrelated computer codes has been developed for the analysis of the LOCA.

TABLE 15.6.5-2 (SHEET 1 OF 3)

LARGE-BREAK RESULTS

Calculation

Number of safety injection pumps operating: 1 CCP, 1 HHSI, 1 RHR
Steam generator tube plugging level (%): 10

Results	DECLG C_D=0.8	DECLG C_D=0.6	DECLG Maximum Safety Injection C_D=0.6	DECLG C_D=0.4
Peak clad temperature (°F)	1736	2046 (Unit 1) [a] 1996 (Unit 2) [a]	1830	1623
Peak clad location (ft)	8.75	8.75	6.25	7.0
Max local Zr/H_2O reaction (%)	3.70	5.62	2.88	1.97
Local Zr/H_2O location (ft)	5.5	8.75	7.0	7.0
Total Zr/H_2O reaction (%)	>1.0	>1.0	>1.0	>1.0
Hot rod burst time (s)	41.55	42.73	42.73	90.15
Hot rod burst location (ft)	5.5	6.0	6.0	7.0

Results (seconds)	DECLG C_D=0.8	DECLG C_D=0.6	DECLG Maximum Safety Injection C_D=0.6	DECLG C_D=0.4
Start	0.0	0.0	0.0	0.0
Reactor trip signal	0.436	0.442	0.442	0.454
Safety injection signal	1.50	1.72	1.72	2.14
Accumulator injection (CL)	12.10	14.80	14.80	19.80
End of bypass	25.513	30.839	30.839	37.622
End of blowdown	25.516	30.839	30.839	37.622
Pump injection	36.50	41.72	41.72	42.14
Bottom of core recovery	37.892	43.890	43.72	51.196
Accumulator empty	48.133	52.736	53.497	58.196
PCT time	150.75	192.79	55.70	90.15

TABLE 15.6.5-2 (SHEET 2 OF 3)

(Peak Clad Temperature Changes for Large Break LOCA Analysis)

	UNIT 1	Unit 2
Calculated PCT from ECCS Model analysis	1915°F	1915°F
PCT change due to evaluation for ±6°F T_{avg} uncertainty band	11°F	11°F
PCT change due to evaluation for containment purging	10°F	10°F
Model assessment - steam generator flow area application	10°F	10°F
Model assessment - structural metal heat modeling	-25°F	-25°F
Model assessment - LUCIFER code error correction	-6°F	-6°F
Model Assessment (1998) - Increased accumulator line resistances	43°F	43°F
LOCBART clad modeling change	-4°F	-4°F
Permanent radiation shield/TSP baskets	1°F	1°F
PCT change due to evaluation for VANTAGE 5 fuel transition cycle penalty (temporary)	50°F	0°F
ZIRLO™ cladding	5°F	5°F
Translation of fluid conditions from SATAN to LOCTA	15°F	15°F
PCT change when using 1.5x or 1.25x IFBA with ZIRLO™ cladding and initial fuel pressure of 100 psig. This applies to full power conditions at burnups less than 150 MWD/MTU.	21°F	21°F
Total resultant PCT[(b)]	2046°F	1996°F

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REV 6 4/97
REV 5 9/95
REV 4 4/94
REV 3 12/92
REV 1 3/91

TABLE 15.6.5-2 (SHEET 3 OF 3)

(Peak Clad Temperature Changes for Large Break LOCA Analysis)

a. PCT values contain assessments detailed in table 15.6.5-2 (Sheets 2 and 3).

b. Consistent with the 1997 10 CFR 50.46 ECCS Evaluation Model Annual Report which includes the Unit 1 Cycle 8 Reload Safety Evaluation, and is also consistent with the Unit 2 Cycle 7 Reload Safety Evaluation.

15.7.2 RADIOACTIVE LIQUID WASTE SYSTEM LEAK OR FAILURE

15.7.2.1 Identification of Causes

This is an infrequent fault, but the potential for release of significant amounts of radioactivity is present. The accident may be caused by an equipment malfunction or tank failure.

Liquid radwaste system leaks resulting in gaseous releases to the atmosphere and liquid releases to the ground water are bounded by the tank failure analyses presented in paragraphs 15.7.2.5 and 15.7.3.4, respectively. Accidental releases of alternate radwaste building liquid effluents to surface water are discussed in paragraph 2.4.13.2.

15.7.2.2 Sequence of Events and System Operation

The recycle holdup tank (RHT) is assumed to fail. This releases 100 percent of the tank capacity to the tank compartment.

15.7.2.3 Core and System Performance

This accident does not affect the core or the core system performance.

15.7.2.4 Barrier Performance

It is assumed that there are no barriers to the release of radioactivity from the auxiliary building.

15.7.2.5 Radiological Consequences

15.7.2.5.1 Method of Analysis

15.7.2.5.1.1 Physical Model. Reactor coolant shim bleed and some valve leakage is held in the RHT.

Table 15.7.3-1 provides an inventory and the concentrations of stored radioactivity in the tank. In the analyses, it is assumed that the liquid contents of the tank are released to the auxiliary building, and subsequently the airborne activity is released to the environment during the 2-h period immediately following the tank failure.

The RHT was selected because it contains the maximum total inventory.

15.7.2.5.1.2 Assumptions and Conditions. The major assumptions and parameters assumed in this analysis are listed below and in table 15.7.2-1.

A. The nuclide inventory of the failed tank is taken from table 15.7.3-1 and is based on 1-percent defective fuel.

B. The RHT failure is assumed to occur when the contents of the tank are at a maximum.

C. The doses are calculated as if the release were from the auxiliary building at ground level during the 2-h period immediately following the accident. No credit is taken for radioactive decay during holdup in the tank or in transit to the site boundary.

D. One-hundred percent of all noble gas activity in the tank is released while 1 percent of the iodine activity is released as airborne activity.

E. Credit is not taken for iodine removal by the nonsafety-grade auxiliary building heating, ventilation, and air-conditioning (HVAC) charcoal adsorber.

15.7.2.5.1.3 Mathematical Models Used in the Analysis.

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 program described in section 2.3; they are provided in table 15A-2.

C. The thyroid inhalation dose and total body immersion dose to a receptor at the exclusion area boundary or outer boundary of the low population zone were analyzed, using the models described in appendix 15A, sections 15A.2.4, and 15A.2.6, respectively.

15.7.4 FUEL HANDLING ACCIDENTS

The postulated fuel handling accident has been analyzed for three cases: case 1, a fuel handling accident outside the containment in the fuel handling building: case 2, a fuel handling accident inside the reactor containment building with the containment personnel airlock closed; and case 3, a fuel handling accident inside the containment with the personnel airlock door open.

15.7.4.1 Identification of Causes and Accident Description

The accident is defined as dropping of a spent fuel assembly onto another fuel assembly in the fuel storage area or refueling pool, resulting in the rupture of the cladding of all the fuel rods in the dropped assembly plus additional rods in the struck assembly, despite many administrative controls and physical limitations imposed on fuel handling operations. All refueling operations are conducted in accordance with prescribed procedures.

15.7.4.2 Sequence of Events and Systems Operations

The first step in fuel handling is the safe shutdown and cooldown of the reactor. After a radiation survey of the containment, the disassembly of the reactor vessel is started. After disassembly is complete, the first fuel handling is started. The first fuel transfer operation shall not begin until at least 100 h after shutdown.

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

15.7.4.3 Core and System Performance

The fuel handling accident in the containment building or the fuel building does not impact the integrity of the core or its system performance.

15.7.4.4 Barrier Performance

The barriers between the released activity and the environment are the containment building or the fuel building. Since these buildings are designed Seismic Category 1, it is safe to assume

that during the course of a fuel handling accident their integrity is maintained. Normally, release of radioactivity for a postulated accident in the fuel building is via the fuel building emergency filtration system. An open door in the fuel handling building pressure boundary could create another release path.

For a postulated accident in the containment building with the airlock closed, the release is limited to the minimal amount of radioactivity which could potentially be released prior to automatic containment isolation. For a postulated accident in the containment building with the airlock open the pathway for the release of activity is via the equipment building ventilation fan. During core alterations or movement of irradiated fuel assemblies within containment, the air lock door interlock mechanism may remain disabled, but the air lock must always be isolable by at lease one air lock door with a designated individual available to close the air lock door, or at least one air lock door must be closed. The requirements for containment penetration closure are sufficient to ensure fission product radioactivity release from containment due to a fuel handling accident during refueling is maintained to within the acceptance criteria of Standard Review Plan subsection 15.7.4 and General Design Criteria 19.

The emergency air lock is farther away from the control room air intake than the personnel airlock. Therefore, the release path from the personnel airlock remains bounding for control room dose. Similarly, potential release paths from the purge supply and exhaust ductwork are no closer than the personnel airlock release path. Offsite dose is not affected by the relative locations of the personnel and emergency airlocks, or the containment purge supply and exhaust ventilation.

The spent fuel pool and the refueling pool provide a minimum decontamination factor of 200 for elemental iodine.

15.7.4.5 Radiological Consequences

15.7.4.5.1 Method of Analysis

15.7.4.5.1.1 Physical Model. The possibility of a fuel handling accident is remote because of the many administrative controls and physical limitations imposed on the fuel handling operations. (Refer to subsection 9.1.4.) All refueling operations are conducted in accordance with prescribed procedures.

When transferring irradiated fuel from the core to the spent fuel pool for storage, the reactor cavity and refueling pool are filled with borated water at a boron concentration equal to or greater than that concentration specified for the spent fuel pool or that concentration specified for refueling, whichever is highest, which ensures subcritical conditions in the core even if all rod cluster control (RCC) assemblies are withdrawn. After the reactor head and RCC drive shafts are removed, fuel assemblies are lifted from the core, transferred vertically to the upender, lowered to a horizontal position on the transfer car and pulled through the transfer tube and canal, upended and transferred through the spent fuel pool transfer gate, then lowered into steel racks for storage in the spent fuel pool in a pattern which precludes any possibility of a criticality accident.

Fuel handling manipulators and hoists are designed so that the fuel cannot be raised above a position that provides an adequate water shield depth for radiation protection of operation personnel.

The containment, fuel building, refueling cavity, refueling pool, and spent fuel pool are designed to Seismic Category 1 requirements, which prevent the structures themselves from failing in the event of a safe shutdown earthquake. The spent fuel storage racks are also located to prevent any credible external missile from reaching the stored irradiated fuel. The fuel handling manipulators, cranes, trollies, bridges, and associated equipment above the water cavities through which the fuel assemblies move are designed to prevent this equipment from generating missiles and damaging the fuel. The construction of the fuel assemblies precludes damage to the fuel should portable or hand tools drop on an assembly.

A fuel handling accident could occur during the transfer of a fuel assembly from the core to its storage position in the spent fuel pool. The facility is designed so that heavy objects, such as the spent fuel shipping cask, cannot be carried over or tipped over onto the irradiated fuel stored in the spent fuel pool. Only one fuel assembly can be handled at a time. Movement of equipment handling the fuel is kept at low speeds, while exercising caution that the fuel assembly does not strike another object or structure during transfer from the core to its storage position. In the unlikely event that an assembly becomes stuck in the transfer tube, natural convection will maintain adequate cooling.

A. Containment Building Accident With Airlock Closed

At times during fuel handling operations, the containment is kept in an isolatable condition, with all penetrations to the outside atmosphere either closed or capable of being closed on an alarm signal from the effluent monitors or one of the two safety-related

redundant area radiation monitors, indicating that radioactivity is above the prescribed limits. (See section 11.5 and subsection 12.3.4.)

Hence, the equipment hatch and at least one of the two interlock doors on the personnel locks is kept closed. In addition to the area and effluent radiation monitors in the containment, portable monitors capable of sounding audible alarms are to be located in the fuel handling area. Should a fuel assembly be dropped and release activity above a prescribed level, the radiation monitors would sound an audible alarm, the containment would be isolated, and personnel would be evacuated. The purge and vent lines are automatically closed on a containment isolation signal, modes 1 through 4, thus minimizing the escape of any radioactivity. Hence, the radioactivity released due to the dropping of a fuel assembly in the containment building is contained, providing considerable holdup time allowing for radioactive decay of the released fission products.

B. Containment Building Accident With Airlock Open

At times during fuel handling operations, the containment airlock is open. The personnel hatch interlocks are disengaged, allowing both of the interlock doors to remain open and personnel to exit containment as quickly as possible. In addition to area and effluent radiation monitors in the containment, portable monitors capable of sounding audible alarms are to be located in the fuel handling area. Should a fuel assembly be dropped and release activity exceed a prescribed level, the radiation monitors would sound an audible alarm and personnel would be evacuated. Although there is no driving force, the radioactivity released due to the dropping of a fuel assembly is assumed to be released from the containment through the airlock area into the equipment building. The radioactivity is then released via the equipment building ventilation system to the outside atmosphere.

C. Fuel Building Accident

In the fuel building, a fuel assembly could be dropped in the transfer canal or in the spent fuel pool.

In addition to the area and effluent radiation monitors, portable radiation monitors capable of emitting audible alarms are located in this area during fuel handling operations. The doors in the fuel

building are normally kept closed to ensure controlled leakage characteristics in the spent fuel pool region during operations involving irradiated fuel. Doors on the pressure boundary, except for the railroad bay door, may be held open during operations involving irradiated fuel. The fuel handling building normal and post-accident ventilation system function is not lost due to opening of personnel doors. Administrative controls are in place to close the doors after a fuel handling accident to minimize any potential release to the environment. Additionally, if none of the fuel handling building exhaust filter units are in service, one should be placed in service to ensure flow past the radiation monitors, or the doors should be closed. Should a fuel assembly be dropped in the canal or in the pool and release radioactivity above a prescribed level, the radiation monitors would sound an alarm. (See section 11.5 and subsection 12.3.4.) On alarm signal, the fuel building ventilation is switched to the emergency mode and exhausts through the engineered safety features (ESF) emergency filtration system charcoal and high-efficiency particulate air filters to remove most of the halogens and particulates prior to discharging to the atmosphere via the plant vent.

A radiation monitor located in the fuel handling building ventilation exhaust duct sounds an alarm if the radioactivity in the vent discharge exceeds the prescribed level.

The probability of a fuel handling accident is very low because of the safety features, administrative controls, and design characteristics of the facility, as previously mentioned.

15.7.4.5.1.2 Assumptions and Conditions. The major assumptions and parameters assumed in the analysis are itemized in tables 15.7.4-1 and 15A-1.

In the evaluation of the fuel handling accident, the fission product release assumptions of Regulatory Guide 1.25 are followed. Table 15.7.4-2 provides a comparison of the design to the requirements of Regulatory Guide 1.25. The following assumptions, related to the release of fission product gases from the damaged fuel assembly, are used in the analyses except as identified in table 15.7.4-2:

A. The dropped fuel assembly is assumed to be the assembly containing the peak fission product inventory. All the fuel rods contained in the dropped assembly are assumed to be damaged. In addition, for the analyses of the

accident in the containment building the dropped assembly is assumed to damage 50 rods of an additional assembly.

B. The assembly fission product inventories are based on a radial peaking factor of 1.70.

C. The accident occurs 100 h after shutdown, which is the earliest time fuel handling operations can begin. Radioactive decay of the fission product inventories was taken into account during this time period.

D. Only that fraction of the fission products which migrates from the fuel matrix to the gap and plenum regions during normal operation is assumed to be available for immediate release to the water following clad damage.

E. The gap activity released to the fuel pool from the damaged fuel rods consists of 10 percent of the total noble gases and iodines, other than Kr-85, I-127, and I-129, which are 30 percent and I-131 which is 12 percent (NUREG CR-5009).

F. The pool decontamination factor is 1.0 for noble gases and organic iodine.

G. The effective pool decontamination factor is 200 for elemental iodine.

H. The iodine released from the fuel assembly is assumed to be composed of 99.75-percent inorganic and 0.25-percent organic species.

I. The activity which escapes from the pool is assumed to be available for release to the environment in a time period of 2 h.

J. No credit for decay or depletion during transit to the exclusion area boundary or the outer boundary of the low population zone is assumed.

K. No credit is taken for mixing or holdup in the fuel building atmosphere or equipment building.

L. For the case inside the reactor containment building, conservative credit is taken for mixing of the radioactivity released from the refueling pool with a minimum of the containment building free volume.

The mixing volume of 25 percent is assumed and is based on the normal airflow rate of four fan coolers.

M. The containment purge rate is 15,000 ft^3/min for the case with the containment airlock closed.

N. The containment purge is isolated within 10 s from the time the containment isolation signal is generated, and a 5-s signal generation time is assumed for the case with the airlock closed.

O. The control room emergency filtration system (CREFS) is initiated by RE-12116 and/or RE-12117 during a fuel handling accident in the fuel handling building.

P. For the case with the airlock open, the activity is assumed to be released from the containment to the outside atmosphere at a rate in which all the activity released from the damaged fuel assemblies would be released to the outside atmosphere in 2 hours if the containment airlock remains open. However, it is assumed that plant personnel close off the containment airlock release path at 15 min after the fuel handling accident. The radiological consequences of a fuel handling accident in containment have been evaluated assuming that the containment is open to the outside atmosphere. All airborne activity reaching the containment atmosphere is assumed to be exhausted to the environment within 2 hours of the accident. The calculated offsite and control room operator doses are within the acceptance criteria of Standard Review Plan 15.7.4 and General Design Criteria 19. Therefore, although the containment penetrations do not satisfy any of the NRC Policy Statement criteria, Technical Specifications provide containment closure capability to minimize potential offsite doses. Procedures provide administrative controls to ensure that the designated person available to close the personnel and/or emergency air lock doors does not have other duties that would preclude the ability to operate the door in a timely manner.

Q. The control room volume, normal and emergency mode flow rates, and emergency mode filter efficiencies for the control room ventilation system used to determine control room doses following a fuel handling accident with the containment airlock open are given in table 15A-1 of appendix 15A.

Based on the core inventories in table 15A-3 and items A through E above, the activity released from a fuel handling accident is presented in table 15.7.4-3.

15.7.4.5.1.3 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, section 15A.2.

B. The atmospheric dispersion factors are based on the onsite meteorological measurement described in section 2.3 and are provided in table 15A-2.

C. The thyroid inhalation and total-body immersion doses to a receptor located at the exclusion area boundary and outer boundary of the low population zone are analyzed using the models described in appendix 15A, subsections 15A.2.4 and 15A.2.5, respectively.

D. The thyroid inhalation, beta skin, at gamma body doses to personnel in the control room are analyzed using the models described in appendix 15A, and subsections 15A.3.3, 15A3.4, and 15A.3.5.

15.7.4.5.1.4 Identification of Leakage Pathways and Resultant Leakage Activity. For evaluating the radiological consequences due to the postulated fuel handling accident in the fuel building, the resultant activity is conservatively assumed to be released to the environment during the 0- to 2-h period immediately following the occurrence of the accident. This is a considerably higher release rate than that based on the actual ventilation rate. Therefore, the results of the analysis are based on the most conservative pathway available.

15.7.4.5.2 Identification of Uncertainties and Conservatisms in Analysis

The uncertainties and conservatisms in the assumptions used to evaluate the radiological consequences of a fuel handling accident result from assumptions made involving the amount of fission product gases available for release to the environment and the meteorology present at the site during the course of the accident. The most significant of these assumptions are:

A. It is assumed in the analysis that all the fuel rods in the dropped assembly are damaged. This is a highly conservative assumption, since in transferring fuel under strict fuel handling procedures, only under the worst possible circumstances could the dropping of a spent fuel assembly result in damage to all the fuel rods contained in the assembly.

B. The fission product gap inventory in a fuel assembly is dependent on the power rating of the assembly and the temperature of the fuel. The gap fractions from Regulatory Guide 1.25 and NUREG CR-5009 are conservatively assumed. Realistic calculations of gap fractions show less than 2 percent for all short lived isotopes.

C. Iodine removal from the released fission product gas takes place as the gas rises to the pool surface through the body of liquid in the spent fuel pool. The extent of elemental iodine removal is determined by mass transfer from the gas phase to the surrounding liquid and is controlled by the bubble diameter and contact time of the bubble in the solution. The values used in the analysis result in a release of elemental iodine approximately a factor of 3 greater than anticipated. The release of activity from the pool to the containment atmosphere is time dependent, and, consequently, there would be sufficient time for this activity to mix homogeneously in a significantly greater percent of the containment volume than assumed in the analysis.

D. Fuel handling building emergency filtration system charcoal filters are provided; however, no credit has been taken for their capability. This means a reduction in the iodine concentrations and, thus, a reduction in the thyroid doses at the exclusion area boundary and the outer boundary of the low population zone.

E. The containment purge exhaust system has charcoal adsorber units which filter any containment purge release. However, no credit has been taken for its capability (90-percent efficiency, minimum) since these units are not specifically designed to Seismic Category 1 criteria. It is expected that for any event which would produce a catastrophic failure of the charcoal adsorber unit to the extent that its filtering capability would be negated would also result in the purge exhaust fan becoming inoperable. Therefore, failure within the purge exhaust system would terminate any high-volume release from the containment. In fact, the purge exhaust fan is considerably more likely to be inoperable following any postulated event than the failure of a passive charcoal adsorber unit. Thus, although no credit in the analysis has been given for the normal purge exhaust filters, any release prior to containment isolation would be filtered, reducing the calculated releases by another factor of 10.

F. There is also conservatism in the time to first fuel transfer. Despite the fact that fuel could be transferred at 100 h, it is probable that fuel handling will begin sometime later.

G. The exhaust from the airlock area is via the equipment building ventilation fan (1526-B7-00200). The distance from the equipment building ventilation exhaust to the control room (CR) intake is 190 ft. If both the intake and exhaust fans fail (loss of power to nonsafety-grade equipment), activity may exfiltrate through the equipment building intake. The distance from the equipment building ventilation intake to the CR intake is 90 ft. For the case of the fuel handling accident inside containment with the airlock open, the activity is assumed to be released from the equipment building to the environment and from the environment to the CR intake. Furthermore, the CR dose analysis conservatively assumes that activity is released from the equipment building intake, which is closer to the CR than the equipment building exhaust.

H. The distance from the CR intake to the nearest point on the containment is 70 ft. The atmospheric dispersion factors (χ/Q) at the CR intake used to determine the CR doses following a LOCA are based on this distance of 70 ft. The distance of 90 ft from the equipment building intake to the CR intake is comparable to the distance of 70 ft used in the LOCA CR dose analysis. Thus, the same χ/Q as used in the LOCA CR dose analysis, which are listed in table 15A-2, are conservatively used for the CR dose analysis for a fuel handling accident inside containment with the airlock open.

I. The assumption that all radioactivity released due to the fuel handling accident is released from the containment and to the outside atmosphere in the initial 2 h following the accident if the containment remains open is conservative. There is no driving force to push this activity out of the containment. The bulk of the radioactivity would likely stay in the containment for much longer than 2 h, even if the containment remained open.

J. The meteorological conditions which may be present at the site during the course of the accident are uncertain. However, it is highly unlikely that meteorological conditions assumed will be present during the course of the accident for any extended period of time. Therefore, the radiological consequences evaluated, based on the meterological conditions assumed, are conservative.

15.7.4.5.2.1 Filter Loadings. The filtration systems which function to limit the consequences of a fuel handling accident in the fuel building are the fuel building emergency filtration system and the control room filtration system.

The activity loadings on the control room charcoal adsorbers as a function of time have been evaluated for the loss-of-coolant accident (LOCA), as described in subsection 15.6.5. Since these filters are capable of accommodating the design basis LOCA fission product iodine loadings, more than adequate design margin is available with respect to postulated fuel handling accident releases.

The activity loadings on the ESF filtration system charcoal adsorbers have been evaluated in accordance with Regulatory Guide 1.52, which limits the maximum loading to 2.5 mg iodine/g activated charcoal.

15.7.4.5.2.2 Doses to Receptor at the Exclusion Area Boundary and Low Population Zone Outer Boundary. The potential radiological consequences resulting from the occurrence of a postulated fuel handling accident occurring in the fuel building and in the reactor building have been conservatively analyzed, using assumptions and models described in previous sections. The total-body 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 2 h) at the low population zone outer boundary. The results are listed in table 15.7.4-4. The resultant doses are well within the guideline values of 10 CFR 100.

o_j = control room occupancy fraction during time interval j.

15A.3.4 CONTROL ROOM BETA SKIN DOSE CALCULATIONAL MODEL

The beta skin doses to a control room operator are calculated using the following equation:

$$D_{\beta\text{-}CR} = \sum_i DCF_{\beta i} \sum_j (IA)_{CRij} \times o_j$$

where:

$D_{\beta\text{-}CR}$ = beta skin dose in the control room (rem).

$DCF_{\beta i}$ = beta skin dose conversion factor for isotope i (rem-m_3/Ci-s).

IA_{CRij} = integrated activity concentration in the control room, $\frac{Ci\text{-}s}{jm^3}$ for the isotope i during time interval

o_j = control room occupancy fraction during time interval j.

15A.3.5 CONTROL ROOM GAMMA BODY DOSE CALCULATION

Due to the finite size of the control room, the gamma body doses to a control room operator will be substantially less than what they would be due to immersion in an infinite cloud of gamma emitters. The finite cloud gamma doses are calculated using Murphy's method which models the control room as a hemisphere.[3] The following equation is used:

$$D_{\gamma\beta\text{-}CR} = \frac{1}{GF} \sum_i DCF_{\gamma i} \sum_j (IA_{CRij}) \times o_j$$

where:

$D_{\gamma\beta\text{-}CR}$ = gamma body dose in the control room (rem).

GF = dose reduction due to control room geometry factor.

GF = $1173/V_1^{0.338}$ (dimensionless).

V_1 = volume of the control room (ft^3).

$DCF_{\gamma i}$ = gamma body dose conversion factor for isotope i (rem-m^3/Ci-s).

IA_{CRij} = integrated activity concentration in control room, $\frac{Ci\text{-}s}{m^3}$ for isotope i during time interval j.

o_j = control room occupancy fraction during time interval j.

15A.3.5.1 Model for Radiological Consequences Due to Radioactive Cloud External to the Control Room

This dose is calculated based on the semi-infinite cloud model which is modified using the protection factors described in subsection 7.5.4 of reference 4 to account for the control room walls.

15A.4 EFFECT OF EXTENDED FUEL BURNUP ON ACCIDENT DOSES

Table 15A-3 is based on discharge region average burnups of 36,000 MWd/Mtu. Calculations performed for the VANTAGE 5 fuel design[5] considered fuel enrichments of 4.5 weight percent and lead rod average burnups to 60,000 MWd/Mtu (discharge batch average burnups of 45,000 MWd/Mtu). The VANTAGE 5 analysis calculations showed the variations in core inventories to be insignificant. This finding is consistent with the current NRC position regarding extended fuel burnups[6,7]. Generic calculations performed for the VANTAGE+ fuel design[8] considered fuel enrichments up to 5.0 wt percent U-235, discharge batch average burnups of 54,000 MWd/Mtu, and lead rod burnups up to 75,000 MWd/Mtu. As was the case with the VANTAGE 5 analysis, the VANTAGE+ analysis also showed that variation in core inventories, as compared to values based on 36,000 MWd/Mtu, were insignificant. The NRC approved this conclusion with regard to operating and storing fuel, setting limits for the storage of fuel with enrichments up to a maximum of 5.0 wt percent U-235[9] in the Unit 1 and Unit 2 spent fuel racks. Thus, the use of table 15A-3, core inventories, in calculating the radiological consequences of accidents remains valid for LOPAR, VANTAGE 5, and VANTAGE+ fuel assembly designs to 60,000 MWd/Mtu lead rod average burnups.

REFERENCES

1. "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," United States Nuclear Regulatory Commission (USNRC) Regulatory Guide 1.145, August 1979.

2. "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50 Appendix I," USNRC Regulatory Guide 1.109, Rev 1, October 1977.

3. Murphy, K. G., and Campe, K. M., "Nuclear Power Plant Control Room Ventilation System Design for Meeting General Criterion 19," Paper presented at the 13th AEC Air Cleaning Conference.

4. "Meteorology and Atomic Energy 1968," D. H. Slade (ed.), USAEC Report, TID-24190, 1968.

5. Westinghouse letter 90GP*-G-0099, "VOGTLE 1 and 2 FINAL VANTAGE 5 RTSR and Licensing Submittal," dated October 2, 1990.

6. Federal Register/Vol. 53 No. 39/ Monday, February 29,1988/pages 6040-6043.

7. Baker, D. A., et al.,"Assessment of the Use of Extended Burnup Fuel in Light Water Power Reactors," NUREG/CR-5009, February 1988.

8. Davidson, S.L., and T. L. Ryan, "VANTAGE+ Fuel Assembly Reference Core Report," WCAP-12610-P-A (proprietary), April 1995.

9. C. K. McCoy to NRC, "Vogtle Electric Generating Plant Response to Request for Additional Information Revised Request to Revise Technical Specifications Credit for Boron and Enrichment Increase for Fuel Storage," LCV-0849-E, August 8, 1997.

TABLE 15A-5

DOSE CONVERSION FACTORS USED IN ACCIDENT ANALYSIS[a]

Nuclide	Total Body $\frac{rem\text{-}m^3}{Ci\text{-}s}$	Beta Skin $\frac{rem\text{-}m^3}{Ci\text{-}s}$	Thyroid (rem/Ci)[b]
I-131	NA	NA	1.49E+6
I-132	NA	NA	1.43E+4
I-133	NA	NA	2.69E+5
I-134	NA	NA	3.73E+3
I-135	NA	NA	5.60E+4
Kr-85m	3.71E-2	4.63E-2	NA
Kr-85	5.11E-4	4.25E-2	NA
Kr-87	1.88E-1	3.09E-1	NA
Kr-88	4.67E-1	7.52E-2	NA
Xe-131m	2.91E-3	1.51E-2	NA
Xe-133m	7.97E-3	3.15E-2	NA
Xe-133	9.33E-3	9.70E-3	NA
Xe-135m	9.91E-2	2.25E-2	NA
Xe-135	5.75E-2	5.90E-2	NA
Xe-138	2.80E-1	1.31E-1	NA

a. Dose conversion factors from International Commission on Radiological Protection (ICRP) 2 have been used in dose calculations performed subsequent to 1995.

b. Dose conversion factors from International Commission on Radiological Protection (ICRP) 30 have been used for SGTR calculations. See table 15.6.3-9.

TABLE 15A-6

REACTOR COOLANT IODINE CONCENTRATIONS FOR 1 μCi/g AND 60 μCi/g OF DOSE EQUIVALENT I-131[a]

Nuclide	Reactor Coolant Concentration (μCi/g) 1 μCi/g Dose Equivalent I-131	60 μCi/g Dose EquivalentI-131
I-131	0.76	45.6
I-132	0.76	45.6
I-133	1.14	68.4
I-134	0.195	11.7
I-135	0.63	37.8

a. SGTR calulations used values based on International Commission on Radiological Protection (ICRP) 30. See table 15.6.3-5.

TABLE 15A-7

IODINE APPEARANCE RATES IN THE REACTOR COOLANT (Ci/s)[c]

	Equilibrium Appearance Rates due to Fuel Defects[a]	Appearance Rates Due to an Accident-Initiated Iodine Spike[b]
I-131	3.4×10^{-3}	1.7
I-132	1.8×10^{-2}	9.0
I-133	7.2×10^{-3}	3.6
I-134	1.1×10^{-2}	5.5
I-135	6.8×10^{-3}	3.4

a. Based on RCS concentration of 1 μCi/g of dose equivalent I-131.

b. 500 x equilibrium appearance rate. These appearance rates are assumed to revert to the above equilibrium appearance rates once the reactor coolant iodine level increases to 100 μCi/g dose equivalent I-131.

c. SGTR calculations used values based on International Commission on Radiological Protection (ICRP) 30. See table 15.6.3-6.

TABLE 16.3-1 (SHEET 1 OF 3)

REACTOR TRIP SYSTEM INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME
1. Manual Reactor Trip	N/A
2. Power Range, Neutron Flux (N-0041, N-0042, N-0043, N-0044)	≤ 0.5 s[a]
3. Power Range, Neutron Flux, High Positive Rate (N-0041, N-0042, N-0043, N-0044)	N/A
5. Intermediate Range, Neutron Flux (N-0035, N-0036)	N/A
6. Source Range, Neutron Flux (N-0031, N-0032)	≤ 0.5 s[a]
7. Overtemperature ΔT (TE-0411, TE-0421, TE-0431, TE-0441)	[a] [c]
8. Overpower ΔT (TE-0411, TE-0421, TE-0431, TE-0441)	[a] [c]
9. Pressurizer Pressure--Low (PI-0455, PI-0456, PI-0457, PI-0458)	≤ 2 s
10. Pressurizer Pressure--High (PI-0455, PI-0456, PI-0457, PI-0458)	≤ 2 s
11. Pressurizer Water Level--High (LI-0459, LI-0460, LI-0461)	N/A
12. Reactor Coolant Flow--Low	
a. Single Loop (Above P-8)	≤ 1 s
b. Two Loops (Above P-7 and below P-8)	≤ 1 s

Loop 1	Loop 2	Loop 3	Loop 4
FI-0414	FI-0424	FI-0434	FI-0444
FI-0415	FI-0425	FI-0435	FI-0445
FI-0416	FI-0426	FI-0436	FI-0446

TABLE 16.3-1 (SHEET 2 OF 3)

FUNCTIONAL UNIT	RESPONSE TIME
13. Steam Generator Water Level--Low-Low [b]	≤ 2 s

Loop 1	Loop 2	Loop 3	Loop 4
LI-0519	LI-0529	LI-0539	LI-0549
LI-0518	LI-0528	LI-0538	LI-0548
LI-0517	LI-0527	LI-0537	LI-0547
LI-0551	LI-0552	LI-0553	LI-0554

14. Undervoltage - Reactor Coolant Pumps	≤ 1.5 s
15. Underfrequency - Reactor Coolant Pumps	≤ 0.6 s
16. Turbine Trip	
a. Low Fluid Oil Pressure (PI-6161, PI-6162, PI-6163)	N/A
b. Turbine Stop Valve Closure	N/A
17. Safety Injection Input from ESF	N/A
18. Reactor Trip System Interlocks	N/A
19. Reactor Trip Breakers	N/A
20. Automatic Trip and Interlock Logic	N/A

a. Neutron detectors are exempt from response time testing. Response time of the neutron flux signal portion of the channel shall be measured from detector output or input of first electronic component in channel. (This provision is not applicable to construction permits docketed after January 1, 1978. See Regulatory Guide 1.118, June 1978.)

b. See also Technical Specification 3.3.3.

c. FSAR analysis assumptions:

RTD response time:	≤ 4 s
Electronic delay time:	≤ 4 s (based upon a 2-s ΔT filter time constant and a 2-s trip time delay)
ΔT filter time constant:	0.0 s (not modeled explicitly; included as part of 4-s electronics delay)

TABLE 16.3-1 (SHEET 3 OF 3)

For above analysis values, following are required channel response times for an RTD time response of no greater than 4.4 s.

1. Overtemperature ΔT, T_{avg} input:	$\leq$ 3.6 s
2. Overtemperature ΔT, pressurizer pressure input:	$\leq$ 4 s
3. Overtemperature ΔT, nuclear flux input:	$\leq$ 4 s
4. Overpower ΔT, T_{avg} input:	$\leq$ 6.565 s
5. ΔT input:	$\leq$ 2.037 s

ΔT input response time includes effect of filter time constant. If response time verification is not performed with time constants on, the acceptance criterion is $\leq$ 1.6 s.

With the exception of the overpower ΔT, T_{avg} input, for RTD time constants longer than 4.4 s, the above channel response times can be reduced by the amount the RTD time constant exceeds 4.4 s.

TABLE 16.3-2 (SHEET 1 OF 6)

ENGINEERED SAFETY FEATURES RESPONSE ITEMS

INITIATION SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
1. Manual Initiation	
a. Safety Injection (ECCS)	N/A
Feedwater Isolation	N/A
Component Cooling Water	N/A
Containment Cooling Fans	N/A
Nuclear Service Cooling Water	N/A
Containment Ventilation Isolation	N/A
b. Containment Spray	N/A
c. Phase "A" Isolation	N/A
d. Auxiliary Feedwater	N/A
e. Steam Line Isolation	N/A
f. Control Room Ventilation Emergency Mode Actuation	N/A
g. Reactor Trip	N/A
h. Start Diesel Generators	N/A
2. Containment Pressure--High-1	
a. Safety Injection (ECCS)	≤ 39(1)/27(5)
b. Reactor Trip (from SI)	≤ 2
c. Feedwater Isolation	≤ 7
d. Phase "A" Isolation	≤ 2(6)

TABLE 16.3-2 (SHEET 2 OF 6)

INITIATION SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
e. Containment Ventilation Isolation	≤ 1.5[6]
f. Auxiliary Feedwater	≤ 60
g. Nuclear Service and Component Cooling Water	≤ 100[1]/88.5[2]
h. Containment Cooling Fans	≤ 48[1]/36.5[2]
i. Control Room Ventilation Emergency Mode Actuation	69.3[12]/99.3[13]
j. Start Diesel Generators	≤ 13.5[7]
3. Pressurizer Pressure--Low	
a. Safety Injection (ECCS)	≤ 39[1]/27[5]
b. Reactor Trip (from SI)	≤ 2
c. Feedwater Isolation	≤ 7
d. Phase "A" Isolation	≤ 2[6]
e. Containment Ventilation Isolation	≤ 1.5[6]
f. Auxiliary Feedwater	≤ 60
g. Nuclear Service and Component Cooling Water	≤ 100[1]/88.5[2]
h. Containment Cooling Fans	≤ 48[1]/36.5[2]
i. Control Room Ventilation Emergency Mode Actuation	69.3[12]/99.3[13]
j. Start Diesel Generators	≤ 13.5[7]
4. Steam Line Pressure--Low	
a. Safety Injection (ECCS)	≤ 39[1]/27[5]

TABLE 16.3-2 (SHEET 3 OF 6)

INITIATION SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
b. Reactor Trip (from SI)	≤ 2
c. Feedwater Isolation	≤ 7
d. Phase "A" Isolation	≤ 2[6]
e. Containment Ventilation Isolation	≤ 1.5[6]
f. Auxiliary Feedwater	≤ 60
g. Nuclear Service and Component Cooling Water	≤ 100[1]/88.5[2]
h. Containment Cooling Fans	≤ 48[1]/36.5[2]
i. Control Room Ventilation Emergency Mode Actuation	69.3[12]/99.3[13]
j. Start Diesel Generators	≤ 13.5[7]
k. Steam Line Isolation	≤ 10[3]
5. Containment Pressure--High-3	
Containment Spray	≤ 65[2]/94[1]
6. Containment Pressure--High-2	
Steam Line Isolation	≤ 10[3]
7. Steam Line Pressure - Negative Rate--High	
Steam Line Isolation	≤ 10[3]
8. Steam Generator Water Level--High-High	
a. Turbine Trip	≤ 2.5
b. Feedwater Isolation	≤ 7
9. Steam Generator Water Level--Low-Low	
a. Motor-Driven Auxiliary Feedwater Pumps	≤ 60

TABLE 16.3-2 (SHEET 5 OF 6)

INITIATION SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
16. Fuel Handling Building Exhaust Duct Radiation	
a. Fuel Handling Building Post Accident Ventilation Actuation	N/A

TABLE NOTATIONS

(1) Time to full ECCS flow. Signal sensing, diesel generator starting, and sequencer loading delays included.

(2) Diesel generator starting delay not included. Offsite power available.

(3) Electrohydraulic valves.

(4) Deleted.

(5) Time to full ECCS flow. Diesel generator starting delay not included.

(6) Does not include valve closure time.

(7) Signal sensing, diesel generator starting and diesel generator breaker delay included.

(8) Does not include valve closure time and relates to post-accident radiation sources as specified in FSAR subsection 15.7.4.

(9) The response time shall include the time delay associated with the loss of voltage relays plus the delay associated with operation of the respective SF sequencer output relays.

(10) The response time shall include the time delay associated with the undervoltage relays plus the delay associated with operation of the respective SF sequencer output relays.

TABLE 16.3-4 (SHEET 4 OF 16)

Valve Number	Function	Valve Closure Time(s)
Containment Ventilation Isolation (continued)		
HV-2626B	Containment mini-purge supply and equalizing	≤ 5
HV-2627B	Containment mini-purge supply and equalizing	≤ 5
HV-2628A	Containment pre-access purge exhaust and equalizing	≤ 10
HV-2629A	Containment pre-access purge exhaust and equalizing	≤ 10
HV-2628B	Containment mini-purge exhaust and equalizing	≤ 5
HV-2629B	Containment mini-purge exhaust and equalizing	≤ 5
HV-2624A	Post-accident air exhaust	N/A
HV-2624B	Post-accident air exhaust	N/A
3. Safety Injection		
HV-8811B[d]	RHR emergency sump suction	N/A
HV-8811A[d]	RHR emergency sump suction	N/A
HV-2134[e]	NSCW supply to reactor cavity coolers	≤ 40
HV-2138[e]	NSCW return from reactor cavity coolers	≤ 40
HV-2135[e]	NSCW supply to reactor cavity coolers	≤ 40
HV-2139[e]	NSCW return from reactor cavity coolers	≤ 40
HV-8105	Normal charging line	≤ 17[j]
HV-1809[e]	NSCW supply to containment coolers	N/A
HV-1807[e]	NSCW supply to containment coolers	N/A
HV-1806[e]	NSCW supply to containment coolers	N/A

TABLE 16.3-4 (SHEET 9 OF 16)

Valve Number	Function	Valve Closure Time(s)
Remote Manual	(continued)	
HV-2790A[i]	Containment H_2 monitor suction	N/A
HV-2790B[i]	Containment H_2 monitor suction	N/A
HV-2793A[i]	Containment H_2 monitor discharge	N/A
HV-2792B[i]	Containment H_2 monitor suction	N/A
HV-2791B[i]	Containment H_2 monitor suction	N/A
HV-2792A[i]	Containment H_2 monitor suction	N/A
HV-2793B[i]	Containment H_2 monitor discharge	N/A
HV-5194[b][k]	Auxiliary feedwater	N/A
HV-5197[b][k]	Auxiliary feedwater	N/A
HV-5195[b][k]	Auxiliary feedwater	N/A
HV-5196[b][k]	Auxiliary feedwater	N/A
HV-9556A[k]	Steam generator secondary side sample	N/A
HV-9556B[k]	Steam generator secondary side sample	N/A
HV-9555A[k]	Steam generator secondary side sample	N/A
HV-9555B[k]	Steam generator secondary side sample	N/A
HV-9554A[k]	Steam generator secondary side sample	N/A
HV-9554B[k]	Steam generator secondary side sample	N/A
HV-9553A[k]	Steam generator secondary side sample	N/A
HV-9553B[k]	Steam generator secondary side sample	N/A
HV-3009[h]	Main steam to auxiliary feedwater pump driver	N/A

TABLE 16.3-4 (SHEET 10 OF 16)

Valve Number	Function	Valve Closure Time(s)
Remote Manual	(continued)	
HV-3019[h]	Main steam to auxiliary feedwater pump driver	N/A
6. Manual		
1213-U6-050[c]	Purification water supply to refueling cavity	N/A
1213-U6-051[c]	Purification water supply to refueling cavity	N/A
1418-U4-005[c]	Demineralized water supply	N/A
2401-U4-211[c]	Breathing air supply	N/A
1411-U4-676[c]	Chemical addition	N/A
1411-U4-677[c]	Chemical addition	N/A
1411-U4-678[c]	Chemical addition	N/A
1411-U4-679[c]	Chemical addition	N/A
1204-U4-159[c]	Accumulator sample line	N/A
1204-U4-161[c]	Accumulator sample line	N/A
1204-U4-160[c]	Accumulator sample line	N/A
1204-U4-162[c]	Accumulator sample line	N/A
1202-U4-001	NSCW supply to containment fire protection	N/A
1202-U4-002	NSCW supply to containment fire protection	N/A
1508-U4-012[c]	Post-accident air exhaust	N/A

TABLE 16.3-5 (SHEET 5 OF 36)

Protective Device Number	Powered Equipment
1/2BBC07-2 1/2BBC07-1	CTB COOLING UNIT A7007 (1/2HV-2585A)
1/2BBC08-2 1/2BBC08-1	CTB COOLING UNIT A7008 (1/2HV-2585B)
1/2ABC13-2 1/2ABC13-1	CTB NORM PURGE EXH ISO (1/2HV-2628A)
1/2ABC15-2 1/2ABC15-1	CHARGE TO REAC COOL SYSTEM ISO (1/2HV-8146)
1/2BBC15-2 1/2BBC15-1	CHARGE TO REAC COOL SYSTEM ISO (1/2HV-8147)

Breaker coordination is provided between 1/2AB0505 and the following:

1/2ABC20	ISOLATION TRANSFORMER 2ABC20X, 1ABC20RX (2-1807-Y3-RX11) (1-1807-Y3-RX25)
1/2ABC23	ISOLATION TRANSFORMER 2ABC23X, 1ABC23RX (2-1808-T3-103) (1-1808-T3-112)

Breaker coordination is provided between 1/2BB0705 and the following:

1/2BBC20	ISOLATION TRANSFORMER 2BBC20X, 1BBC20RX (2-1807-Y3-RX12) (1-1807-Y3-RX26)

TABLE 16.3-5 (SHEET 6 OF 36)

Protective Device Number	Powered Equipment
1/2BBC23	ISOLATION TRANSFORMER 1/2BBC23X (1/2-1808-T3-104)

Breaker coordination is provided between 1/2AB0514 and the following:

1/2ABF13	ISOLATION TRANSFORMER 1/2ABF13X (1/2-1808-T3-105)

Breaker coordination is provided between 1/2BB0714 and the following:

1/2BBF13	ISOLATION TRANSFORMER 2BBF13X, 1BBF13RX (2-1808-T3-106) (1-1808-T3-115)
1/2ABC19-2 1/2ABC19-1	ACCUMULATOR LOOP #3 (1/2HV-8808C)
1/2BBC19-2 1/2BBC19-1	ACCUMULATOR LOOP #2 (1/2HV-8808B)
1/2ABC32-2 1/2ABC32-1	THERMAL BARRIER CW RCP 003 (1/2HV-19055)
1/2ABC33-2 1/2ABC33-1	THERMAL BARRIER CW RCP 004 (1/2HV-19057)
1/2NBE08-2 1/2NBE08-1	RCP OIL LIFT PUMP MOTOR #1 (1/2-1201-P6-001-M02)

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Protective Device Number	Powered Equipment
Backup protection is provided by 1ABA29 for Unit 1 and 2ABA02 for Unit 2 for the following:	
1AYA103	FHB PA FILTER EXH MOTOR HTR (A-1542-N7-001-H02
1/2AYA105	CB ESF CHILLER WTR PMP MOTOR HTR (1/2-1592-P7-001-H01)
1/2AYA110	CB NORMAL A/C UNIT FAN MOTOR HTR (1/2-1539-A7-005-H01)
1/2AYA112	CB CR CHILLER RM VENT FAN MOTOR HTR (1/2-1531-B7-002-H01)
1/2AYA118	AUX CCW PMP MOTOR HTR (1/2-1217-P4-001-H01)
1/2AYA125	CCW PMP MOTOR HTR (1/2-1203-P4-005-H01)
1/2AYA127	CCW PMP MOTOR HTR (1/2-1203-P4-001-H01)
1/2AYA129	CCW PMP MOTOR HTR (1/2-1203-P4-003-H01)

TABLE 16.3-5 (SHEET 33 OF 36)

Protective Device Number	Powered Equipment
1/2AYA131	NSCW PMP MOTOR HTR (1/2-1202-P4-001-H01)
1/2AYA133	NSCW PMP MOTOR HTR (1/2-1202-P4-003-H01)
1/2AYA135	NSCW PMP MOTOR HTR (1/2-1202-P4-005-H01)

Backup protection is provided by 1BBA29 for Unit 1 and 2BBA02 for Unit 2 for the following:

1/2BYA105	ESF CHILLED WTR PMP MOTOR HTR(1/2-1592-P7-002-H01)
1/2BYA110	CB AUX RELAY RM A/C MTR HTR (1/2-1539-A7-002-H01)
1BYA112	CB ELEC EQUIP RM ESF A/C MOTOR HTR (1-1539-A7-006-H01)
1/2BYA114	CB CR CHILLER RM VENT FAN MOTOR HTR (1/2-1531-B7-004-H01)
1/2BYA119	NSCW PMP MOTOR HTR (1/2-1202-P4-002-H01)
1/2BYA121	NSCW PUMP MOTOR HTR (1/2-1202-P4-004-H01)

TABLE 16.3-5 (SHEET 36 OF 36)

Protective Device Number	Powered Equipment
1/2BYF115	AUX FW PP HSE FAN MTR HTR (1/2-1593-B7-002-H01)
1/2BYF117	DIESEL TUNNEL 1T4B VENT FAN MTR HTR (1/2-1540-B7-002-H01)
1/2NYR09 CB3 (LOCATED IN FLUX MAPPING CONSOLE 1/2-1612-Q5-MDF)	INCORE RACK NO. 2 DEHUMIDIFIER
1/2NYS09 CB1 (LOCATED IN FLUX MAPPING CONSOLE 1/2-1612-Q5-MDF)	INCORE RACK NO. 2 DEHUMIDIFIER
CB2 (LOCATED IN FLUX MAPPING CONSOLE (1/2-1612-Q5-MDF)	INCORE FLUX MAPPING CONSOLE CONTROL POWER

Primary and backup protection is provided by fuses for the following:

PERSONNEL AIR LOCK NORMAL LIGHTING

PERSONNEL AIR LOCK ESSENTIAL LIGHTING

PERSONNEL AIR LOCK EMERGENCY LIGHTING

ESCAPE AIR LOCK NORMAL LIGHTING

ESCAPE AIR LOCK ESSENTIAL LIGHTING

ESCAPE AIR LOCK EMERGENCY LIGHTING

TABLE 16.3-6 (SHEET 1 OF 2)

AREA TEMPERATURE MONITORING

UNIT 1 - MINIMUM NORMAL HVAC COMPONENTS

BUILDING	EQUIPMENT DESCRIPTION	TAG NUMBER
AUX	(1 of 2) Supply Fans AND (2 of 3) Continuous Exhaust Fans	1-1551-A7-001, -002 1-1553-N7-001, -002, -003
CONTROL	(1 of 1) Wing Area Normal AC Unit AND	1-1533-A7-001
	(1 of 1) Wing Area Normal Exh and Return Fan	1-1533-B7-001
FHB[a]	(1 of 2) Supply Fans AND (1 of 2) Normal Exhaust Filtration Unit	A-1541-A7-001, -002 A-1541-N7-001, -002
EQPT	(0 of 2) Fans	1-1526-B7-001, -002

AREA TEMPERATURE LIMITATIONS

BUILDING	ROOM NO.	TEMPERATURE LIMIT (°F)
AUX	212	100
AUX	A047	100
AUX	B017	100
AUX	B023	100
AUX	C083	100
AUX	C113	100
AUX	D053	100
AUX	D067	100
AUX	D072	100
AUX	D073	100
AUX	D119	100
AUX	D121	100
CONTROL	A055	100
CONTROL	B065	100
CONTROL	B074	100
CONTROL	B078	100
EQUIPMENT	117	120
EQUIPMENT	125	120
FHB	B008	104

TABLE 16.3-6 (SHEET 2 OF 2)

AREA TEMPERATURE MONITORING

UNIT 2 - MINIMUM NORMAL HVAC COMPONENTS

BUILDING	EQUIPMENT DESCRIPTION	TAG NUMBER
AUX	(1 of 2) Supply Fans AND (2 of 3) Continuous Exhaust Fans	2-1551-A7-001, -002 2-1553-N7-001, -002, -003
CONTROL	(1 of 1) Wing Area Normal AC Unit AND (1 of 1) Wing Area Normal Exh and Return Fan	2-1533-A7-001 2-1533-B7-001
FHB[a]	(1 of 2) Supply Fans AND (1 of 2) Normal Exhaust Filtration Unit	A-1541-A7-001, -002 A-1541-N7-001, -002
EQPT	(0 of 2) Fans	2-1526-B7-001, -002

AREA TEMPERATURE LIMITATIONS

BUILDING	ROOM NO.	TEMPERATURE LIMIT (°F)
AUX	221	100
AUX	A082	100
AUX	B113	100
AUX	C010	100
AUX	D106	100
AUX	D107	100
AUX	D113	100
AUX	D123	100
CONTROL	A002	100
CONTROL	A010	100
CONTROL	B002	100
CONTROL	B010	100
CONTROL	B019	100
EQUIPMENT	117	120
EQUIPMENT	125	120
FHB	B006	104

a. Whenever less than the minimum normal HVAC components for the fuel handling building are inoperable, the area temperatures must be verified for applicable rooms of both Units 1 and 2.

17.2 OPERATIONS QUALITY ASSURANCE PROGRAM

Southern Nuclear Operating Company (SNC) has established a total quality program to maintain standards of quality covering the engineering, design, procurement, manufacturing, construction, testing, maintenance, repair, refueling, and modifications of applicable structures, components, and systems for VEGP. As the exclusive operating licensee, SNC assumes full responsibility for the quality program and will take appropriate action to guarantee that VEGP is designed, constructed, and operated with sound engineering principles and safe operating practices.

The responsibility for establishing the operations quality assurance program policies, goals, and objectives, as summarized below, is vested in the vice president-project (Vogtle).

A. All participants in the quality assurance program, including suppliers and contractors, shall meet appropriate sections of 10 CFR 50, Appendix B criteria.

B. The program described herein shall be documented by written procedures or instructions, appropriate to the circumstances.

C. The quality assurance program shall provide control over activities affecting the quality of structures, systems, and components consistent with their importance to safety.

D. Activities affecting quality shall be accomplished under controlled conditions which include the use of appropriate equipment, suitable environmental conditions, and assurance that prerequisites for the given activity have been satisfied.

E. The quality program shall take into account the need for special controls, processes, test equipment, tools, and skills to attain the required quality.

F. Verification of conformance to established requirements shall be accomplished by individuals or groups who do not have direct responsibility for performing the work being verified.

G. The quality program shall provide for indoctrination and training of personnel performing activities affecting quality, as necessary, to ensure that suitable proficiency is achieved and maintained.

H. Management shall regularly review the status, adequacy, and implementation of the quality program through direct oversight and the quality assurance auditing process.

This section describes the operations quality assurance program (OQAP) for VEGP and SNC. It defines responsibility and authority and prescribes measures for the control and accomplishment of activities affecting the quality of safety-related structures, systems, and components of VEGP. The OQAP requirements herein are implemented in approved procedures, which provide requirements for the performance of safety-related activities. The OQAP is implemented by SNC and executed by various responsible parties to ensure the plant's safe and reliable operation and to satisfy the quality assurance requirements of Appendix B of 10 CFR 50, as delineated in Regulatory Guide 1.33, Revision 2, Quality Assurance Program Requirements (Operation). American National Standards Institute N18.7-1976, Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants, was used for guidance in developing the OQAP.

The organizational structure of SNC is described in section 13.1. The following subsection amplifies upon section 13.1 with regard to establishment and execution of the OQAP.

17.2.1 ORGANIZATION

The principal elements implementing the OQAP are as follows:

- Corporate organization.
- Nuclear support organization.
- Plant organization.
- Safety audit and engineering review (SAER).

Figure 17.2.1-1 shows the relationship of the above elements involved in the OQAP.

17.2.1.1 Corporate Organization

The vice president-project (Vogtle) is the final management authority responsible for the development, implementation, changes to, and review of the OQAP. These responsibilities are accomplished through the organization shown in figure 17.2.1-1 and are discussed in the following paragraphs.

The vice president-project (Vogtle) is responsible to the president/CEO through the executive vice president for the effective implementation of the quality assurance program for operation of the VEGP. The nuclear plant general manager reports to the vice president-project (Vogtle) and is responsible for the day-to-day operation of the VEGP.

The vice president-project (Vogtle) regularly assesses the scope, adequacy, and compliance of the OQAP to 10 CFR 50, Appendix B, through a combination of direct oversight, frequent meetings with the safety audit and engineering review manager (SAERM), review of safety review board activities, and assessment of quality assurance audits.

An annual assessment of the effectiveness of the quality assurance auditing program is performed and results are reported to the vice president-project (Vogtle).

Refer to chapter 13 for more details of the nuclear operations corporate organization.

17.2.1.1.1 Safety Review Board

The safety review board shall be appointed by the vice president-project (Vogtle) for the purpose of advising him in matters related to nuclear power plant safety. Board members shall have competence in various fields of engineering technology and be familiar with nuclear safety, environmental, and regulatory requirements. The safety review board has access to the advice and services of technical specialists within and outside expertise as necessary. Board members' responsibilities and authorities are provided in the following discussion.

17.2.1.1.1.1 Function. The SRB shall function to provide independent review and audit of designated activities in the areas of:

A. Nuclear power plant operations,
B. Nuclear engineering,
C. Chemistry and radiochemistry,
D. Metallurgy,

E. Instrumentation and control,
F. Radiological safety,
G. Mechanical and electrical engineering, and
H. Quality assurance practices.

The SRB shall report to and advise the vice president-project on those areas of responsibility specified in paragraphs 17.2.1.1.1.7 and 17.2.1.1.1.8.

17.2.1.1.1.2 Composition. The SRB shall be composed of a minimum of five persons who, as a group, provide the expertise to review and audit the operation of a nuclear power plant. The chairman and other members shall be appointed by the vice president-project (Vogtle) or other such person as he/she may designate. The composition of the SRB shall meet the requirements of ANSI N18.7-1976. No more than a minority of the SRB will be members of the onsite operating organization.

The chairman and each member of the SRB, as a minimum, have a bachelor's degree in engineering or related sciences and 3 years of professional level experience in their field of specialty.

17.2.1.1.1.3 Alternates. All alternate members shall be appointed in writing by the SRB chairman to serve on a temporary basis; however, no more than a minority of alternates shall participate as voting members in SRB activities at any one time.

17.2.1.1.1.4 Consultants. Consultants shall be utilized as determined by the SRB chairman to provide expert advice to the SRB.

17.2.1.1.1.5 Meeting Frequency. The SRB shall meet at least once per 6 months.

17.2.1.1.1.6 Quorum. The quorum of the SRB necessary for the performance of the SRB review functions described in paragraph 17.2.1.1.1.7 and audit functions described in paragraphs 17.2.1.1.1.8 and 17.2.18.1 shall consist of the chairman or his/her designated alternate and at least a majority of the SRB members including alternates. No more than a minority of the quorum shall have line responsibility for operation of the plant.

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REV 4 4/94
REV 1 3/91

17.2.1.1.1.7 Review. The SRB shall be responsible for the review of:

A. The safety evaluations for:

1. Changes to procedures, equipment, or systems; and
2. Tests or experiments completed under the provision of 10 CFR 50.59, to verify that such actions did not constitute an unreviewed safety question;

B. Proposed changes to procedures, equipment, or systems which involve an unreviewed safety question as defined in 10 CFR 50.59;

C. Proposed tests or experiments which involve an unreviewed safety question as defined in 10 CFR 50.59;

D. Proposed changes to Technical Specifications or the Operating License;

E. Violations of codes, regulations, orders, technical specifications, license requirements, or of internal procedures or instructions having nuclear safety significance;

F. Significant operating abnormalities or deviations from normal and expected performance of plant equipment that affect nuclear safety;

G. All reportable events;

H. All recognized indications of an unanticipated deficiency in some aspect of design or operation of structures, systems, or components that could affect nuclear safety; and

I. Reports and meetings of the PRB.

17.2.1.1.1.8 Audits. The specific requirements for audits of plant activities which shall be performed under the cognizance of the SRB are discussed in paragraph 17.2.18.1.

17.2.1.1.1.9 Records. Records of SRB activities shall be prepared, approved, and distributed as indicated below:

A. Minutes of each SRB meeting shall be prepared, approved, and forwarded to the vice president-project (Vogtle) within 14 days following each meeting;

B. Reports of reviews encompassed by paragraph 17.2.1.1.1.7 shall be prepared, approved, and forwarded to the vice president-project (Vogtle) within 14 days following completion of the review; and,

C. Audit reports encompassed by paragraph 17.2.1.1.1.8 shall be forwarded to the vice president-project (Vogtle) and to the management positions responsible for the areas audited within 30 days after completion of the audit by the auditing organization.

17.2.1.1.2 Nuclear Support

The nuclear support general manager (Vogtle) reports to the vice president-project (Vogtle) and is responsible for providing support services for the operation of VEGP in accordance with the OQAP.

17.2.1.2 Plant Organization

The plant staff (as described in this chapter and in chapter 13) will perform safety-related activities in accordance with written, approved procedures. Quality assurance requirements will be included in detailed plant procedures. The managers, superintendents, and supervisors in the VEGP organization will be responsible for implementation of the OQAP for activities under their purview. The nuclear plant general manager will regularly assess the workload of all departments involved in the OQAP to ensure that a sufficient number of personnel are available for complete and efficient implementation.

Refer to chapter 13 for more details of plant staff organization and responsibilities. Specific discussion of PRB organization and responsibilities follows.

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17.2.1.2.1 Plant Review Board (PRB)

The Plant Review Board (PRB) serves as a review and advisory group to the NPGM. The PRB reviews plant administration, maintenance, and operations related to nuclear safety and environmental aspects; ensures these activities are consistent with company policy, approved procedures, and operating license provisions; and makes recommendations to the NPGM on matters brought before it.

The PRB is composed of department superintendents or managers, or supervisory personnel reporting to department superintendents or managers, from the departments listed in paragraph 17.2.1.2.1.2 below. The qualification levels for the PRB members are at least equivalent to those described for supervisors as listed in table 13.1.3-1. For supervisors not listed in table 13.1.3-1, the individual shall have a minimum of 5 years of power plant experience, of which 4 years may be substituted by a bachelor's degree in engineering or related science and of which 1 year is nuclear experience.

Collectively, these members possess the type and degree of expertise required to review unplanned events and proposed changes to systems, procedures, tests, and experiments that affect nuclear safety. The chairman, his/her alternate, and other members of the PRB are designated by the NPGM. The minimum quorum of the PRB consists of the PRB chairman or his/her designated alternate and four members (including not more than two voting alternates). The PRB meets at least once per calendar month and as convened by the PRB chairman or his/her designated alternate.

Further detail of the policies for the PRB is provided in the following discussion. Plant procedures implement these policies for the PRB and also contain other necessary detail regarding the activities and duties of the PRB.

17.2.1.2.1.1 Function. The PRB shall function to advise the NPGM on all matters related to nuclear safety.

17.2.1.2.1.2 Composition. The PRB shall be composed of at least five but no more than nine department superintendents or managers, or supervisory personnel reporting directly to department superintendents or managers, including representation from the functional areas listed below:

- Operations.
- Maintenance.

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- Health physics and chemistry.
- Engineering.

A senior health physicist is acceptable for the health physics and chemistry PRB representative. The chairman, his/her alternate and other members and their alternates of the PRB shall be designated by the NPGM.

17.2.1.2.1.3 Alternates. No more than two alternates shall participate as voting members in PRB activities at any one time.

17.2.1.2.1.4 Meeting Frequency. The PRB shall meet at least once per calendar month and as convened by the PRB chairman or his/her designated alternate.

17.2.1.2.1.5 Quorum. The quorum of the PRB necessary for the performance of the PRB responsibility and authority provisions of these requirements (paragraph 17.2.1.2.1) shall consist of the chairman or his/her designated alternate and four members including alternates.

17.2.1.2.1.6 Responsibilities.

17.2.1.2.1.6.1 The PRB shall be responsible for:

A. Review of:

1. Procedures which establish plant-wide administrative controls to implement the QA program or technical specifications surveillance program,

2. Procedures for changing plant operating modes,

3. Emergency and abnormal operating procedures,

4. Procedures for effluent releases of radiological consequences, and

5. Fuel handling procedures.

B. Review of:

1. The following programs and changes thereto,

a. Primary coolant sources outside containment

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b. In-plant radiation monitoring
c. Secondary water chemistry
d. Post-accident sampling
e. Post-accident monitoring instrumentation
f. Radioactive effluent controls

2. Proposed procedures and changes to procedures which involve an unreviewed safety question as per 10 CFR 50.59.

C. Review of all proposed tests and experiments that affect nuclear safety;

D. Review of all proposed changes to the Technical Specifications;

E. Review of all proposed changes or modifications to plant systems or equipment that affect nuclear safety, including proposed changes to chapter 16.3;

F. Investigation of all violations of the technical specifications, including the preparation and forwarding of reports covering evaluation and recommendations to prevent recurrence, to the vice president-project (Vogtle) and to the SRB.

G. Review of all reportable events;

H. Review of plant operations to detect potential hazards to nuclear safety;

I. Performance of special reviews, investigations, or analyses and reports thereon as requested by the NPGM or the SRB;

J. Review of the Security Plan and implementing procedures and submittal of recommended changes to the NPGM and the SRB;

K. Review of the Emergency Plan and implementing procedures and submittal of recommended changes to the NPGM and the SRB;

L. Review of any accidental, unplanned, or uncontrolled radioactive release including the preparation of reports covering evaluation, recommendations, and disposition of the corrective action to prevent recurrence and the forwarding of these reports to the vice president-project (Vogtle) and to the SRB;

M. Review of changes to the PCP, the ODCM, and the radwaste treatment systems; and

N. Review of the fire protection program and implementing procedures and submittal of recommended changes to the NPGM.

17.2.1.2.1.6.2 The PRB shall:

A. Recommend in writing to the NPGM approval or disapproval of items considered under paragraph 17.2.1.2.1.6.1.A through E prior to their implementation;

B. Render determinations in writing with regard to whether or not each item considered under paragraph 17.2.1.2.1.6.1.A through F constitutes an unreviewed safety question; and

C. Provide written notification within 24 hours to the vice president-project (Vogtle) and the SRB of disagreement between the PRB and the NPGM; however, the NPGM shall have responsibility for resolution of such disagreements pursuant to Technical Specification 5.1.1.

17.2.1.2.1.7 Records. The PRB shall maintain written minutes of each PRB meeting that, at a minimum, document the results of all PRB activities performed under the responsibility provisions of these requirements. Copies shall be provided to the vice president-project (Vogtle) and the SRB.

17.2.1.3 Safety Audit and Engineering Review

SAER under the direction of the SAERM has been established to provide a comprehensive independent audit program to verify that safety-related activities are performed and controlled in compliance with the OQAP. The audit program will be conducted by the safety audit and engineering review supervisor (SAERS) and his/her staff stationed at the VEGP and the SAER staff located in the general office. The onsite independent safety engineering group (ISEG), as described in subsection 13.4.3 and paragraph 17.2.1.3.4, also reports to the SAERM and is responsible for independent technical assessments of plant operating activities.

SAER is independent of the nuclear support and VEGP plant staffs. The SAERM reports to the vice president-project (Vogtle). (See figure 17.2.1-1.)

The qualifications for SAER personnel meet the requirements of Regulatory Guide 1.146, Qualification of Quality Assurance Program Audit Personnel for Nuclear Power Plants, as described in section 1.9.

The minimum qualification for the SAERM is to hold an engineering or related science degree and have a minimum of 5 years experience in the areas of engineering, field construction, or plant operation. Two of these 5 years must involve working under a nuclear quality assurance program.

The size of the SAER organization is based on meeting the commitment for the required audits and on experience gained at the Hatch and Farley Nuclear Plants. The approximate size of the site SAER organization is eight technical people for two unit operation.

The corporate office SAER staff provides technical and administrative support to the onsite SAER in the performance of its assigned responsibilities.

17.2.1.3.1 Safety Audit and Engineering Review Manager

The SAERM, located at the corporate office, is responsible to the vice president-project (Vogtle) for assuring implementation of the OQAP and for managing activities of the SAER organization. The SAERM can effectively assure conformance to the quality assurance requirements and is sufficiently independent from cost and schedule concerns when opposed to safety considerations. The SAERM has a staff at the corporate office and a SAERS and his/her staff located at the site to conduct quality assurance activities. The SAERM has been given authority by the vice president-project (Vogtle) to stop, in a timely manner, work on the VEGP site which is not being performed in accordance with the provisions of the OQAP. The SAERM keeps management informed on the effectiveness and implementation of the OQAP through direct interface and distribution of audit reports and other documents.

Specific duties and responsibilities of the SAERM are as follows:

A. Manages the SAER organization.

B. Develops the SAER portions of OQAP to conform to approved policies.

C. Maintains close liaison with the vice president-project (Vogtle) to ensure that program policies and procedures are being implemented and enforced.

D. Keeps the vice president-project (Vogtle) informed orally on routine matters, but reports significant items immediately, both orally and in writing. In addition, major audit reports, including summaries of audits by major contractors, are forwarded to the vice president-project (Vogtle).

E. Informs management of quality assurance activities through distribution of audit reports and other correspondence.

F. Reviews and concurs with quality assurance programs of SNC, the architect/engineer Southern Company Services (SCS), and contractors (Gaithersburg Regional Office of Bechtel Power Corporation (Bechtel), Westinghouse, and plant contractors) including changes thereto prior to the start of work governed by such programs.

G. Audits or verifies accomplishment of delegated audits of the quality assurance programs of SCS, SNC, Bechtel, Westinghouse, and suppliers of materials, equipment, or services, to ensure compliance with approved quality assurance programs and procedures.

H. Assures that site activities conform to quality assurance program requirements and procedures.

I. Assures that audit and monitoring requirements are carried out on project quality-related functions.

J. Ensures the proper implementation of the supplier quality surveillance functions/activities and changes thereto.

K. Ensures that review, audit, reporting, and record-keeping requirements of the SRB are accomplished in accordance with license commitments.

L. Ensures the availability of adequate technical support to the SRB.

17.2.1.3.2 Safety Audit and Engineering Review Supervisor

Located at the plant site and reporting to the SAERM, the SAERS directs the onsite SAER group.

The SAERS will assist in the review of administrative procedures for comment as requested by the SAERM. The SAERS will develop SAER procedures to be approved by the SAERM and utilized in the performance of his activities. The SAERS will serve as a nonvoting member of the PRB, which is discussed in subsection 17.2.15. The SAERS has the responsibility and authority to identify problems, recommend solutions, and verify corrective actions. The SAERS will also provide input for an onsite operations QA indoctrination program as described in subsection 17.2.2. Copies of all site SAER audits and verifications will be sent to the SAERM.

17.2.1.3.3 Safety Audit and Engineering Review (Onsite)

The SAER staff will selectively audit those quality-related activities that are within the scope of the OQAP, as described in subsection 17.2.2, to verify compliance with the requirements of the OQAP. These consist of activities performed onsite and those performed offsite in support of VEGP when directed. The results of all such audits will be reported to the organization audited and the vice president-project (Vogtle).

SAER personnel have written authority to stop work on a system, structure, or component that affects nuclear safety if the work is not in accordance with provisions of the OQAP. Disputes arising from differences of opinion between SAER personnel and other department personnel will be resolved by the lowest level of management possible. If necessary, the vice president-project (Vogtle) will make the final disposition. The SAERM shall regularly assess the SAER workload to ensure a sufficient number of personnel are available for complete and efficient implementation of their quality assurance responsibilities. Specific duties and responsibilities of the SAER onsite group (headed by the SAERS) are as follows:

A. Prepares annual schedule and performs planned audits of organizations and activities (and contractors).

B. Provides the SAERM with information on site activities on a routine basis.

C. Maintains open-items list of SAER onsite group audit results; follows up until resolved and closed out.

D. SAER participates in the development process for plant procedures by evaluating procedure adequacy and implementation in the SAER audit program and through its non-voting membership in the Plant Review Board.

E. To assure that controls are in place to effectively maintain plant procedures, the SAER staff will perform a biennial audit of plant procedures and the procedural development and maintenance program utilizing a representative sampling process. The audit will ensure the acceptability of the procedures and verify that the procedure review and revision program is being implemented effectively. The root cause of significant deficiencies will be determined and corrected.

Safety audit engineering review personnel have access to meetings where quality matters are discussed. The SAERM shall designate the types of meetings his staff or representative(s) from his staff will routinely attend, including day-to-day work planning meetings and staff meetings.

17.2.1.3.4 Independent Safety Engineering Group (Onsite)

The Independent Safety Engineering Group (ISEG) shall function to examine plant operating characteristics, NRC issuances, industry advisories, Licensee Event Reports, and other sources of plant design and operating experience information, which may indicate areas for improving plant safety. The ISEG shall review and make detailed recommendations for revised procedures, equipment modifications, maintenance activities, operations activities, or other means of improving plant safety to the vice president-project (Vogtle) and appropriate plant management.

The authority of the ISEG is such that:

A. The ISEG function shall be independent of the functions of the Plant Review Board and the Safety Review Board.

B. The ISEG shall not be responsible for signoff functions such that it becomes involved in the operating organization.

The ISEG shall be composed of at least five, dedicated, full-time positions. Each shall have either:

1. A bachelor's degree in engineering or related science and at least 2 years professional level experience in his/her field, at least 1 year of which shall be in the nuclear field, or

2. At least 5 years of nuclear experience and hold or have held a Senior Reactor Operator's license, or

3. At least 10 years of professional level experience in his/her field, at least 5 years of which shall be in the nuclear field.

   At least three of these positions shall meet the requirements of item 1.

The ISEG shall be responsible for maintaining surveillance of plant activities to provide independent verification (not responsible for signoff function) that these activities are performed correctly and that human errors are reduced as much as practical.

Results of activities performed by ISEG shall be prepared, maintained, and forwarded each calendar month to the vice president-project (Vogtle).

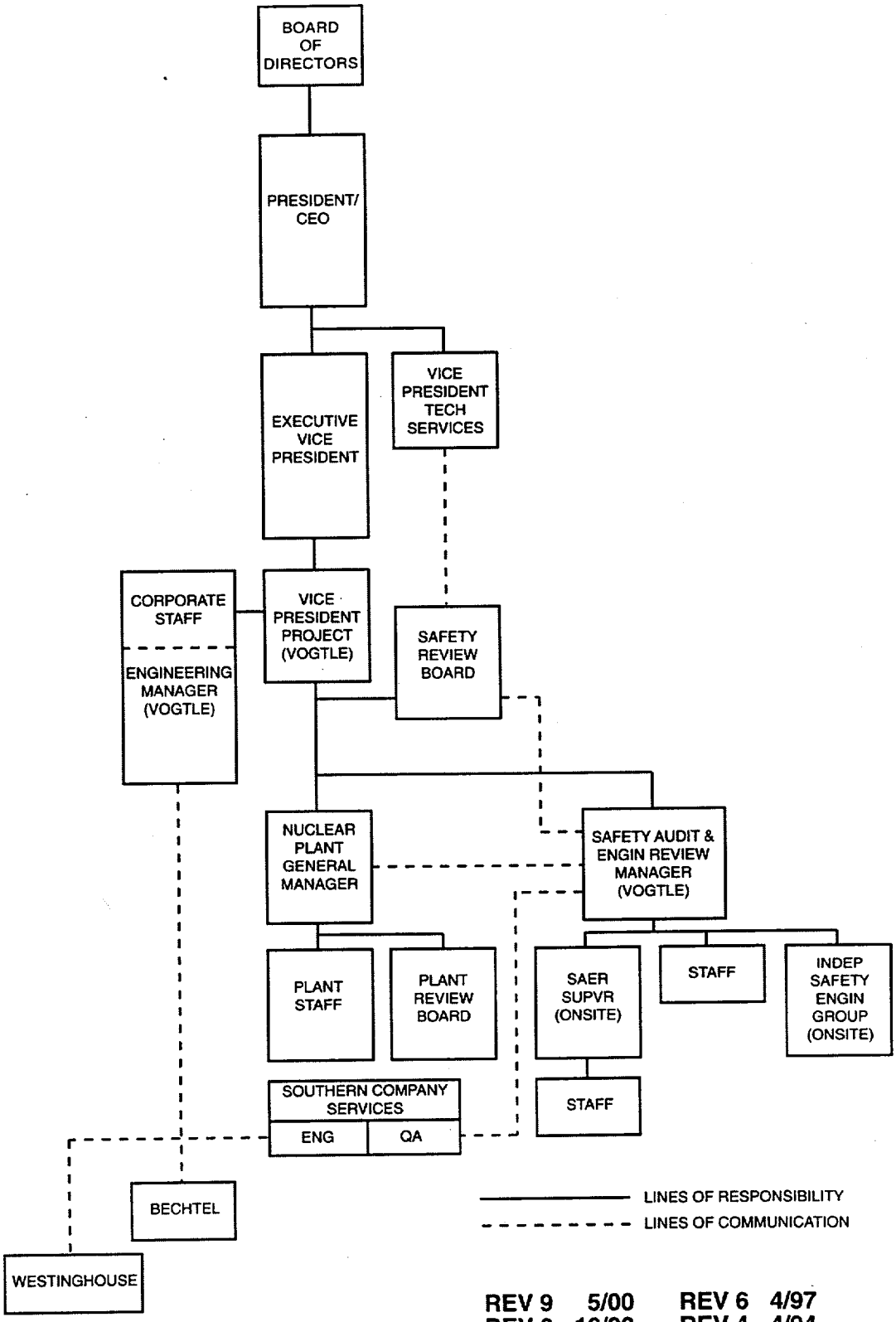
17.2.1.4 Engineering

The Nuclear Support (Vogtle) has overall responsibility for assuring the availability of and providing or securing adequate engineering and technical support for the VEGP. The engineering manager and the licensing support manager (Vogtle) serve as the principal interface between Nuclear Support (Vogtle) and SCS or others for coordination and direction of engineering and technical support provided to the VEGP.

SCS is the primary architect/engineer for VEGP during the operations phase. In addition to providing design engineering/modification support as requested, SCS provides quality assurance support, including audits, supplier qualification and surveillances, and engineering procedure reviews. SCS also administers for VEGP the contract for engineering services provided by Bechtel and selected engineering services provided by Westinghouse. Activities within the SCS work scope are governed by the Plant Vogtle Operational Support Policy and Procedures Manual and the Support Services Quality Assurance Policy Manual. These activities, as well as architect/engineering activities performed by others under contract to SCS or contracted directly by SNC, will meet the requirements of the OQAP. The SAERM performs or causes to be performed audits of these functions.

Bechtel is under contract to SCS to provide architect/engineering services for structures, systems, and components assigned to them by SNC or SCS. The work scope includes plant design, development of purchase recommendations, and support of the SCS supplier surveillance functions by providing procurement surveillance services for selected items. All engineering work is performed in accordance with Engineering Department Procedures (EDPs) and instructions (EDPIs) which are consistent with the Plant Vogtle Operational Support Policy and Procedures Manual (PVOSPPM).

The GPC System Protection and Control Department is responsible for the development and implementation of the protection policy for the GPC transmission system, and distribution substations. SCS Engineering is responsible for the development and implementation of the protection policy for the generating plants. This includes the development of protective relay applications and settings for 13,800-V, 4160-V, and 480-V safety-related systems. These safety-related activities are governed by appropriate QA program procedures. The SAERM performs or delegates audit of these safety-related activities.



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REV 8	10/98	REV 4	4/94
REV 7	9/97	REV 1	3/91

VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

SNC ORGANIZATIONAL CHART FOR QUALITY ASSURANCE DURING PLANT OPERATIONS

FIGURE 17.2.1-1

17.2.2 OPERATIONS QUALITY ASSURANCE PROGRAM

The operations quality assurance program (OQAP) requirements and controls are established to comply with the requirements of 10 CFR 50, Appendix B, and 50.54a, as described in the Final Safety Analysis Report.

The policies, goals, and objectives of the OQAP are established in section 17.2. The items to be covered under the OQAP are designated as "Q" items and are listed for general guidance in table 3.2.2-1. This list was prepared and kept current by the architect/engineer during plant design, construction, and startup, and is now maintained by Nuclear Support. The following guidance is considered when specifying quality requirements:

- The importance of malfunction or failure of the item to plant safety or reliability.
- The design and fabrication complexity or uniqueness of the item.
- The need for special controls and surveillance over process and equipment.
- The degree to which functional compliance can be demonstrated by inspection or test.
- The quality history and degree of standardization of the item.

Safety-related activities will be carried out under suitably controlled conditions, including use of written, approved procedures and appropriate equipment, proper environmental conditions, and assurance that prerequisites have been satisfied. Quality-affecting procedures required for the administration of the OQAP are reviewed for consistency with program commitments and corporate policies and are properly documented and controlled. These policies are documented and approved by the nuclear plant general manager. Quality policies and procedures are mandatory requirements which must be implemented and enforced. Documented procedures control

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the actions of the various organizations involved, such as SAER, quality control, engineering, and operations, including actions to determine the extent of quality assurance control to be applied. These activities include but are not limited to certain aspects of the following as they apply to safety-related structures, equipment, and systems:

- Administrative controls and reporting requirements.
- Health physics and radiation protection.
- Environmental monitoring and environmental technical specifications.
- Plant chemistry.
- Radioactive waste control.
- Reactor and plant operations.
- Materials control (procurement, storage, and issue).
- Quality control (nondestructive examination, inspection, etc.).
- Surveillance program.
- Test equipment calibration and control.
- Security Plan and procedures.
- Emergency Plan and procedures.
- Design changes and plant modification control.
- Departmental training programs.
- Maintenance programs.
- Fuel handling and special nuclear material control.
- Inservice inspection program.
- Procedure control and review.
- Records management and document control.
- Fire protection program (subsection 9.5.1).

- Corrective action programs.
- Computer code programs.

Measures shall be established to ensure that major contractors performing safety-related activities at or in association with VEGP have an adequate quality assurance/quality control program commensurate with the scope of their activities. Applicable sections of American National Standards Institute N18.7-1976 will be required of contractors and suppliers to the degree necessary to ensure compliance with the OQAP.

It is the responsibility of SNC and Southern Company Services, Inc., to comply with the OQAP for plant-associated activities, such as procurement, qualification of suppliers, new plant design, and the necessary redesign or evaluation of any of the safety-related systems, structures, or components.

As delegated by management, the nuclear plant general manager shall have the overall site responsibility for developing and implementing the OQAP with the exception of controls that are assigned to SAER. Safety audit and engineering review conducts independent audits of quality activities, as necessary, to assess the scope, status, implementation, and effectiveness of the OQAP. These audits will be conducted in accordance with predetermined schedules, with audit results documented in reports and a followup system utilized to ensure corrective action is taken when necessary and to verify implementation. The safety audit and engineering review manager (Vogtle) transmits the results of the audit findings to the vice president-project (Vogtle).

Measures shall be provided for indoctrination and training of personnel performing safety-related activities to ensure that high levels of quality are attained in job performance which complies with SNC quality policies and procedures. (See section 13.2 for training description.)

Safety audit and engineering review personnel, in accordance with SAER procedures, will receive formal and on-the-job training in quality assurance policies, requirements, procedures, techniques, plant operation, maintenance, nonconformance control, and auditing. These procedures also prescribe methods for documenting the accomplishment of training.

Records of training and qualifications of SAER personnel will be maintained by the SAERM and will document compliance with appropriate requirements. Personnel of departments other than

SAER, in addition to the training described in section 13.2, will receive indoctrination which includes quality assurance policies and procedures. In addition to this indoctrination, plant operators, engineering personnel, maintenance personnel, quality control personnel, and health physics personnel will receive instruction in their areas of the OQAP. The plant training and emergency preparedness manager will schedule, evaluate, and maintain records of training and indoctrination of plant personnel. Records of training programs shall be maintained as discussed in section 13.2.

Changes that reduce commitments to the OQAP will be submitted in writing to the Nuclear Regulatory Commission (NRC) for approval prior to implementation.

Changes to the OQAP description that do not reduce the commitments will be submitted to the NRC at least every 18 months, consistent with the current FSAR revision cycle. The commitments of this subsection are audited according to the quality assurance audit program, which is described in subsection 17.2.18.

17.2.4 PROCUREMENT DOCUMENT CONTROL

The quality requirements for the control of documents prepared by SNC or its designated agents, for safety-related components, materials, and services are consistent with the provisions of Regulatory Guide 1.123, Quality Assurance Requirements for Control of Procurement of Items and Services for Nuclear Power Plants, as described in section 1.9. The vice president-project (Vogtle) is responsible for procurement planning of VEGP.

To satisfy the quality assurance requirements for procurement document control, the following specific requirements shall be implemented. Measures shall be established to ensure:

A. Procedures are established delineating the sequence of actions to be accomplished in the preparation, review, approval, and control of procurement documents. The nuclear plant general manager is responsible for the preparation, review, approval, and control of procurement documents.

B. A review and concurrence of the adequacy of quality assurance requirements stated in procurement documents is performed by qualified personnel knowledgeable in the quality assurance requirements. Plant procedures will define training requirements for ensuring knowledge of quality assurance requirements. This review is to determine that all quality assurance requirements are correctly stated, can be inspected and controlled, and provide adequate acceptance and rejection criteria and that the procurement document has been prepared in accordance with quality assurance program requirements.

C. The person performing the review will be an individual independent of the procurement document originator. Documented evidence of the review and approval of procurement documents is provided and available for verification.

D. Procurement documents identify those 10 CFR 50, Appendix B, requirements that must be complied with and described in the supplier's quality assurance program. This quality assurance program or portions thereof shall be reviewed and concurred with by the nuclear plant general manager and qualified personnel knowledgeable in quality assurance prior to implementation of activities affected by the program.

E. Procurement documents contain or reference applicable design basis technical requirements including regulatory requirements, component and material identification, drawings, specifications, codes and industrial standards, including their revision status, test and inspection requirements, and special process instructions, for activities such as fabrication, cleaning, erecting, packaging, handling, shipping, storing, and inspecting.

F. Procurement documents contain applicable requirements which identify the documentation to be prepared, maintained, submitted, and made available to the nuclear plant general manager for review and/or approval, such as drawings, specifications, procedures, inspection and fabrication plans, inspection and test records, personnel and procedure qualifications, materials, and chemical and physical test results.

G. Procurement documents contain the requirements for the retention, control, and maintenance of vendor records.

H. Procurement documents contain the procuring agency's right of access to vendor's facilities and records for source inspection and audit.

I. Changes and/or revisions to procurement documents are subject to the same review and approval requirements as the original documents.

J. Procurement documents for spare or replacement parts of safety-related structures, systems, and components are reviewed for adequacy by qualified personnel knowledgeable in quality assurance requirements. The review is to determine similarity, compatibility, and inclusion of the quality assurance requirements and acceptance criteria of the original item.

Assurance that the requirements for procurement document control are implemented will be provided by the audit program described in subsection 17.2.18.

17.2.5 INSTRUCTIONS, PROCEDURES, AND DRAWINGS

Activities affecting quality shall be prescribed by and accomplished in accordance with documented instructions, procedures, or drawings. The nuclear plant general manager is responsible for ensuring that procedures are prepared and followed and that personnel are trained on the procedures. They shall contain appropriate quantitative (such as dimensions, tolerances, and operating limits) or qualitative (such as workmanship and samples) acceptance criteria for determining that important activities have been satisfactorily accomplished. The requirements dealing with instructions, procedures, or drawings are consistent with the provisions of Regulatory Guide 1.30, Quality Assurance Requirements for the Installation, Inspection, and Testing of Instrumentation and Electric Equipment; Regulatory Guide 1.33, Quality Assurance Program Requirements (Operation); and Regulatory Guide 1.116, Quality Assurance Requirements for Installation, Inspection, and Testing of Mechanical Equipment and Services, as described in section 1.9.

17.2.5.1 Plant Procedures, Instruction, and Drawings

The VEGP procedure manuals will contain procedures which shall be used for periodic tests and calibrations, special processes, modifications, maintenance, and other plant activities on safety-related systems, equipment, or structures. Certain procedures important to safety shall be reviewed by the plant review board before being signed by a member of plant management. Inspection plans, tests, calibrations, special processes, maintenance, modifications, repair procedures, and changes thereto will be reviewed and concurred with by a person knowledgeable in quality assurance requirements. Plant procedures will define training requirements for ensuring knowledge of quality assurance requirements.

Written administrative procedures will ensure proper control of instructions such as temporary procedure changes and standing orders. These instructions will be of limited scope and will be issued for limited time periods to ensure proper requirements, included in the procedures, are not bypassed or neglected.

Design documents (active drawings, closed design change packages, ABNs) will reflect the properly reviewed and approved configuration of the plant. Changes will be as a result of design changes (subsection 17.2.3) or "as-builts" which will be controlled by written procedures. The design organization will implement design and drawing control on design documents through written procedures.

Specific requirements for VEGP procedures are provided in the following paragraphs.

17.2.5.1.1 Required Procedures

Written procedures shall be established, implemented, and maintained covering the activities referenced below:

A. The applicable procedures recommended in Appendix A of Regulatory Guide 1.33, Revision 2, February 1978;

B. The emergency operating procedures required to implement the requirements of NUREG-0737 and Supplement 1 to NUREG-0737 as stated in Generic Letter No. 82-33;

C. Security Plan implementation;

D. Emergency Plan implementation;

E. Process Control Program implementation;

F. Quality assurance for effluent and environmental monitoring

G. Fire Protection Program implementation;

H. Technical Specifications Improvement Program implementation; and

I. All programs specified in Technical Specification 5.5.

17.2.5.1.2 Procedure Approval

Each procedure of paragraph 17.2.5.1.1 above, and changes thereto, shall be reviewed as set forth in administrative procedures and approved by either the nuclear plant general manager (NPGM) or the department head of the responsible department prior to implementation with the exception of the following which shall be approved by the NPGM:

1. Procedures which establish plant-wide administrative controls (which implement the quality assurance program and the Technical Specification surveillance program),

2. Unit operating procedures (UOPs)

3. Emergency operating procedures (EOPs)

4. Abnormal operating procedures (AOPs)

5. Procedures for implementing the Security Plan, Emergency Plan, and the Fire Protection Program, and

6. Fuel handling procedures.

PRB responsibilities for procedures are delineated in paragraph 17.2.1.2.1.

17.2.5.1.3 Temporary Changes to Procedures

Temporary changes to procedures of paragraph 17.2.5.1.1 may be made provided:

A. The intent of the original procedure is not altered;

B. The change is approved by two members of the plant management staff, at least one of whom holds a senior operator license; and

C. The change is documented, reviewed in accordance with paragraph 17.2.5.1.2, and approved by the NPGM or department head of the responsible department within 14 days of implementation.

17.2.5.2 Other Procedures

Each contractor performing quality-related activities shall have properly reviewed and approved quality assurance programs or procedures or function under the control of the operations quality assurance program. These programs or procedures shall be reviewed to ensure that they meet the applicable quality assurance requirements contained herein.

Confirmation that these instructions and procedures are properly implemented is accomplished through activities performed by the SAER organization. See subsection 17.2.18 for details.

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17.2.6 DOCUMENT CONTROL

Herein are contained the requirements for the control of documents that prescribe activities affecting quality. The documents which are to be controlled include:

A. Manufacturing, design, construction, and installation drawings for safety-related structures, systems, or components.

B. Approved procurement documents for safety-related structures, systems, or components.

C. Plant procedures which implement requirements of the operations quality assurance program.

D. Maintenance, modification, and operating procedures which implement requirements of the operations quality assurance program.

E. Final Safety Analysis Report, Emergency Plan, Security Plan, environmental reports, Technical Specifications, and Operating License Stage Environmental Report.

F. Inspection and test procedures for safety-related structures, systems, or components.

G. Design documents (e.g., calculations, specifications, and analyses) including documents related to computer codes.

H. Instructions and procedures for such activities as fabrication and installation.

I. As-built documents.

J. Quality assurance and quality control manuals.

K. Deficiency reports.

The requirements for document control are consistent with the position of Regulatory Guide 1.33, Quality Assurance Program Requirements (Operation), as described in section 1.9. Provisions are included which require that:

A. Documents and changes thereto are reviewed for adequacy and approved for release by authorized personnel in accordance with written procedures. These procedures identify those positions responsible for reviewing, approving, and issuing documents and revisions thereto. Measures are also established to

review documents prior to release to ensure that the quality assurance requirements are sufficiently, clearly, and accurately stated. The individuals responsible for the quality assurance requirement review shall be an individual other than the person who generated the document. Individuals authorized to perform the quality assurance requirement review shall be knowledgeable of the quality assurance requirements. The approving authority is responsible for implementing review of the document for quality assurance requirements.

B. Documents and changes thereto are promptly distributed to ensure availability prior to commencement of work. "As-built" documents are prepared in a timely manner to accurately reflect the actual plant design.

C. Changes to documents are reviewed and approved by the same organization that performed the original review and approval unless another qualified organization is designated.

D. Master status lists identifying the current revision of documents are periodically updated and utilized to preclude the use of superseded documents. Procedures are established for control of document revision. These procedures will specify the types of documents to be controlled and the periodic updating and distribution of controlled documents.

E. Obsolete or superseded documents are destroyed or identified to prevent their inadvertent use.

The plant support assistant general manager is responsible for ensuring that plant personnel are provided with current documents. The responsible supervisors are responsible for ensuring that the latest revision of appropriate documents is being used to perform safety-related activities. Documents controlled by offsite organizations will receive control comparable to that of site documents.

Safety audit and engineering review will audit activities to determine proper implementation of the quality requirements contained herein. The quality assurance audit function is described in subsection 17.2.18.

17.2.7 CONTROL OF PURCHASED MATERIAL, EQUIPMENT, AND SERVICES

Spare and replacement parts of safety-related structures, systems, and components are subject to quality assurance program controls in place when the procurement is made and to technical requirements that comply with Final Safety Analysis Report requirements. Expendable or consumable items will be procured in accordance with the provision described herein. The requirements for the control of purchased materials, equipment, and services are consistent with the positions of Regulatory Guide 1.38, Quality Assurance Requirements for Packing, Shipping, Receiving, Storage, and Handling of Items for Water-Cooled Nuclear Power Plants, and Regulatory Guide 1.123, Quality Assurance Requirements for Control of Procurement of Items and Services for Nuclear Power Plants, as described in section 1.9.

17.2.7.1 Supplier Qualification (Vogtle)

The vice president-project (Vogtle) is responsible for ensuring qualification of suppliers of safety-related equipment, materials, or services, including spare and replacement parts, for the VEGP. These activities are in accordance with the operations quality assurance program. SNC shall review the effectiveness of quality control of approved suppliers or subsuppliers at intervals designed to maintain confidence in the suppliers' programs. The review intervals shall be determined according to the complexity of the product or services received from these suppliers. These reviews will also include periodically evaluating the suppliers' certificates of conformance by audits, independent inspections, or tests to ensure they are valid. Prior to accepting a prospective supplier as a qualified GPC nuclear supplier, qualified personnel will evaluate the prospective supplier's capability to provide services or products of acceptable quality based on one or more of the following:

A. The supplier's capability to comply with the elements of 10 CFR 50, Appendix B, that are applicable to the type of material, equipment, or services being procured.

B. A review of previous records and performance of suppliers who have provided similar articles of the type being procured.

C. A survey of the supplier's facilities and quality assurance program to determine the capability to supply a product which meets the design, manufacturing,

and quality assurance requirements. If a survey is performed, the results will be documented and filed. The method used and information analyzed to qualify a supplier will also be documented. Where surveys or audits cannot be conducted, acceptable alternatives will be provided.

Plant procedures will require that procurement documents specify:

A. Documentation that identifies the purchased item and the specific procurement requirements (e.g., codes, standards, and specifications) met by the item.

B. Documentation identifying any procurement requirements that have not been met.

C. A description of those nonconformances from the procurement requirements dispositioned "use as is" or "repair."

17.2.7.2 Evaluation Process

The evaluation of suppliers, including spare and replacement parts for safety-related systems and surveillance during fabrication, inspection, testing, and shipment, shall be performed in accordance with approved procedures. These procedures shall include the following:

A. Instructions that specify the characteristics or processes to be witnessed, inspected or verified, and accepted; the method of surveillance and the extent of documentation required; and those responsible for implementing these instructions and releasing documentation.

B. Audits and surveillance which ensure that the supplier complies with the quality requirements. Surveillance shall be performed on those items where verification of procurement requirements cannot be determined upon receipt.

17.2.7.3 Receiving Inspection

Receipt inspections at VEGP are performed in accordance with written procedures to ensure that:

A. Materials, equipment, or components are verified by a receipt inspector qualified to ANSI N45.2.6 to be identifiable and traceable to associated documentation.

17.2.12 CONTROL OF MEASURING AND TEST EQUIPMENT

Measures for the control of measuring and test equipment are consistent with the position of Regulatory Guide 1.33, Quality Assurance Program Requirements (Operation), as described in section 1.9.

Provisions for the control of applicable measuring and test equipment require that:

A. Procedures shall be established which describe the calibration technique, calibration frequency, maintenance and control of measuring and test instruments, tools, gauges, fixtures, reference standards, transfer standards, and nondestructive test equipment to be used in the measurement and inspection of safety-related components, systems, and structures.

B. Measurement and test equipment shall be uniquely identified and have traceability to calibration test data.

C. Measuring and test instruments shall be calibrated and maintained at specific intervals, based on the required accuracy, purpose, degree of usage, stability characteristics, and other conditions affecting the measurement.

D. Measuring and test equipment shall be calibrated on or before the designated due date or removed from service until corrective action can be taken.

E. When measuring and test equipment is found to be out of calibration, an investigation shall be conducted and documented to determine the validity of previous inspections performed and the acceptability of those items previously inspected since the previous calibration. Procedures are established to ensure that measurements made with measuring and test equipment found to be out of calibration will be evaluated to determine the need for reinspection.

F. Calibrating instruments shall have known, valid relationships to a nationally recognized standard or natural physical constant. If no national standard or natural physical constant exists, the basis of calibration will be documented and approved by the responsible department team leader or higher management.

G. Records will be maintained which indicate the complete status of all items under the calibration system and reflect the last and future calibration dates or the last calibration date and frequency, if applicable.

H. Calibration facilities used for calibrating sensitive and close tolerance measuring and test equipment shall provide an environment that is sufficiently controlled to allow the measuring device to be evaluated and calibrated to its required accuracy.

Reference standards will have an uncertainty (error) requirement of no more than one-fourth of the tolerance of the measuring and test equipment being calibrated; a greater uncertainty will be acceptable provided the equipment and standards not meeting this requirement are documented and approved by the responsible department team leader or higher management. Comparison standards used in calibration of reference standards will have a tolerance equal to or smaller than the reference standard and will be traceable to the standards housed in the National Institute of Standards and Technology. The calibration of reference standards using comparison standards will be supported by certification reports and data sheets.

The appropriate department team leader or higher management is responsible to the nuclear plant general manager for the development, approval, and implementation of procedures which provide for calibrations and for maintenance and control of measuring and test equipment within their scope or responsibility. The administrative procedures that control the calibration program are reviewed by the plant review board and approved by the nuclear plant general manager. Quality control inspection personnel are responsible for verifying the control of measuring and test equipment during inspections of work activities and ensuring that it is in compliance with plant procedures.

Effectiveness of the program is ensured through periodic audits performed under the quality assurance audit system that is described in subsection 17.2.18.

17.2.16 CORRECTIVE ACTION

Approved plant procedures shall be written to ensure that conditions adverse to quality, failures, malfunctions, deficiencies, deviations, defective materials and equipment, and nonconformances on safety-related systems are promptly identified, documented, and corrected. The program described herein requires the identification and correction of conditions adverse to quality. These requirements are consistent with the provisions of Regulatory Guide 1.33, Quality Assurance Program Requirements (Operation), as described in section 1.9.

Corrective action is initiated and tracked following the determination of conditions adverse to quality. Failures, malfunctions, deficiencies, deviations, defective materials and equipment, and nonconformance on safety-related systems are documented by appropriate procedures.

Various processes are used to identify problems requiring corrective action. Items requiring prompt evaluation and correction are handled with a condition report. Condition reports are also used for handling nonconforming items as discussed in subsection 17.2.15 and for other conditions adverse to quality, such as operational deficiencies, unusual events or conditions, and failures to meet Technical Specifications requirements. Other processes for identifying problems include SAER audits and Nuclear Regulatory Commission (NRC) inspections.

The shift supervisor is responsible for necessary immediate corrective action and reporting when condition reports identify conditions significantly compromising plant safety.

When significant conditions adverse to quality exist, the Nuclear Safety and Compliance (NSAC) supervisor is responsible for ensuring the investigation of the problem to determine the root cause and for ensuring and tracking the necessary corrective action to prevent recurrence.

The NSAC supervisor is responsible for analyzing condition reports, SAER audit findings, and NRC violations and deviations for adverse trends. Any identified trends are reported to the appropriate department for corrective action and to the nuclear plant general manager and SAER supervisor. Based on this input, as well as inputs from other processes, the nuclear plant general manager is responsible for making sure that proper corrective action is being taken.

Reports of significant conditions adverse to quality are reviewed by the plant review board, and its recommendations regarding corrective action are forwarded to appropriate management personnel. Verification of proper implementation of corrective action program requirements is achieved by the quality assurance audit program, as described in subsection 17.2.18.

17.2.17 QUALITY ASSURANCE RECORDS

Plant procedures require the proper identification of documents which furnish objective evidence of activities affecting the quality of the VEGP. Written procedures which are consistent with applicable codes, standards, and procurement documents will control the storage and retention of records. The minimum records to be controlled and retained are described in paragraph 17.2.17.1, Record Retention; additional records are retained in accordance with approved procedures. Typical records include plant history, operating logs, reportable occurrences, results of reviews, inspections, tests, audits, material analyses, qualification of personnel, procedures, and equipment; and other documentation, such as drawings, specifications, procurement documents, calibration procedures and reports, nonconformance reports, and corrective action reports.

The plant administration manager is responsible for the receipt, storage, retrievability, issue, and control of specified plant documents. Personnel and organizations submitting records and documents to the documentation staff for storage shall be responsible for providing adequate information with that record or document to allow its systematic storage and retrieval. Inspection records submitted to the documentation staff for storage will be reviewed by the responsible organization to ensure that the requirements in subsection 17.2.10 are included.

Record storage facilities have been constructed, located, and secured to prevent destruction of applicable records by fire, flooding, theft, and deterioration by environmental conditions. Activities associated with quality assurance records are consistent with the position of Regulatory Guide 1.88, Collection, Storage, and Maintenance of Nuclear Power Plant Quality Assurance Records, and Regulatory Guide 1.33, Quality Assurance Program Requirements (Operation), as described in section 1.9.

Safety audit and engineering review will audit activities to ensure the records and retention requirements described herein are implemented. (See subsection 17.2.18.)

Specific requirements for quality assurance records are provided in the following paragraphs of this subsection.

17.2.17.1 Record Retention

17.2.17.1.1 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.

17.2.17.1.2 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 paragraph 17.2.5.1.1;

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.

17.2.17.1.3 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;

17.2.18 AUDITS

Safety audit and engineering review (SAER) is responsible for the program of planned and scheduled audits devised to verify the effectiveness of and compliance with all aspects of the operations quality assurance program (OQAP). The objectives of the quality assurance audit system are to:

A. Ensure that the OQAP has been developed in accordance with applicable requirements.

B. Verify by audit that the OQAP has been implemented.

C. Assess the effectiveness of the OQAP.

D. Identify program deficiencies.

E. Verify resolution of deficiencies and nonconformances.

F. Ensure compliance with Final Safety Analysis Report commitments.

The quality assurance audit program requirements are consistent with the provisions of Regulatory Guide 1.33, Quality Assurance Program Requirements (Operation); Regulatory Guide 1.144, Auditing of Quality Assurance Programs for Nuclear Power Plants; and Regulatory Guide 1.146, Qualification of Quality Assurance Program Audit Personnel for Nuclear Power Plants, as described in section 1.9. To satisfy the requirements for the audit system, the following measures shall be established and implemented:

A. The performance of audits is in accordance with written procedures or checklists and is conducted by appropriately trained and qualified personnel not having direct responsibilities in the areas being audited.

B. Audit results are documented, including the formal identification of problems found and the recommended corrective actions where appropriate, and are issued through preparation of audit finding reports.

C. Responsible management takes the necessary action to correct the deficiencies revealed by the audit.

D. Deficient areas are monitored until corrections have been accomplished.

E. Audits include an objective evaluation of quality-related practices, procedures, and

instructions; the effectiveness of implementation; and the conformance with policy directives.

F. The audits include the evaluation of work areas, activities, and processes and the review of documents and records.

G. The following types of audits shall be performed:

1. Audits to provide a comprehensive independent verification and evaluation of quality-related procedures and activities to ensure that they are adequate and effectively comply with the OQAP requirements. These audits are performed under the direction of the SAERM.

2. External audits performed or delegated by the SAERM on principal contractors and suppliers performing activities on safety-related structures, systems, and components. These audits include verification and evaluation of their quality assurance program, procedures, and activities to ensure that they are adequate and effectively comply with all aspects of the OQAP and procedure requirements.

H. Audits are regularly scheduled on the basis of status and safety importance of the activities being performed; they are initiated early enough to ensure effective quality. Audit schedule changes reflecting more frequent audits are required by one or more of the following conditions:

1. When significant changes are made in functional areas of the OQAP, such as significant reorganization or procedure revisions.

2. When there is evidence that the performance or reliability of safety-related items is in jeopardy due to deficiencies or nonconformances in the OQAP.

3. When a systematic, independent assessment of OQAP effectiveness is necessary.

4. When it is necessary to verify implementation of required corrective actions.

The SAER audit personnel will receive formal and on-the-job training prior to the initiation of the OQAP as outlined in this chapter. The SAER organization under the safety audit and engineering review manager will implement the quality assurance audit system. Paragraph 17.2.1.3 describes the SAER organization.

Plant management is responsible for making available plant records, reports, log books, and other quality assurance documents requested by SAER auditors.

I. Analysis of audits performed and reports which indicate quality trends and the effectiveness of the OQAP will be provided to management.

17.2.18.1 Performance of Audits

During the operating life of the plant, the onsite activities to be audited will include but not limited to the categories stated below. Additionally, certain aspects of these audit categories are performed under SRB cognizance within the specified time interval, as noted, with:

1. A maximum allowable extension not to exceed 25 percent of the inspection audit interval.

2. A total maximum combined interval time for any 3 consecutive inspection or audit intervals not to exceed 3.25 times the specified inspection or audit interval.

3. Audits shall be performed at least once per 24 months unless otherwise noted.

These audits performed under the cognizance of the SRB shall encompass:

A. The conformance of reactor and plant operation to provisions contained within the technical specifications and applicable license conditions, at least once per 12 months;

B. The performance, training, and qualifications of the entire plant staff, at least once per 12 months;

C. The results of actions taken to correct deficiencies occurring in plant equipment, structures, systems, or method of operation that affect nuclear safety, at least once per 6 months;

D. The performance of activities required by the OQAP to meet the criteria of Appendix B, 10 CFR Part 50, at least once per 24 months;

E. The fire protection equipment and program implementation at least once per 12 months, utilizing either a qualified offsite licensee fire protection engineer or an outside independent fire protection consultant. The fire protection programmatic controls including the implementing procedures, at least once per 24 months by qualified licensee QA personnel. An outside independent fire protection consultant shall be used at least every third year;

F. The radiological environmental monitoring program and the results thereof at least once per 12 months;

G. The ODCM and implementing procedures, at least once per 24 months;

H. The PCP and implementing procedures for processing and packaging of radioactive wastes, at least once per 24 months;

I. Effluent and environmental monitoring, at least once per 12 months;

J. The emergency plan and implementing procedures, at least once per 12 months (as defined in the Emergency Plan);

K. The security plan and implementing procedures, at least once per 12 months (as defined in the Physical Security Plan);

L. Fitness for Duty, per the requirements of 10 CFR 26.80(a), (extension does not apply);

M. Special nuclear material control, per the requirements of 10 CFR 70.58(c)(2), (extension does not apply);

N. Health physics and radiation protection, per the requirements of 10 CFR 20.1101(c), (extension does not apply);

O. Industrial Safety, per the requirements of 29 CFR 1910.269, (extension does not apply);

P. Administrative controls and reporting requirements;

Q. Fuel handling;

R. Plant chemistry;

S. Material control (procurement, storage, and issue);

T. Quality control (nondestructive examination, inspection, training, qualification/certification of quality control personnel, etc.);

U. Surveillance program;

V. Test equipment calibration and control;

W. Balance of plant instrument calibration;

X. Computer codes;

Y. Design changes and plant modification control;

Z. Maintenance programs;

AA. Inservice inspection program.

operations facility, and personnel on duty in various parts of the plant, as described below.

A. Communication to/from the control room is provided using sound-powered telephone systems. The equipment is available for maintenance (six jacks/box), refueling (two jacks/box), and shutdown purposes (two jacks/box).

B. The communication system within the plant is a public address system (one page line and party line common to all zones and four party lines). This system is for roving operator communications and is provided with the capability for the merger of four zones: Unit 1, Unit 2, administration building, and outside areas. This system provides remote merging in:

1. Unit 1 control room.

2. Unit 2 control room.

3. Central alarm station.

4. Secondary alarm station.

5. Captain's office in the PESB.

This also functions as an emergency alarm (supplied with emergency power from an uninterruptible power supply system). This system is actuated by a multi-tone generator applicable for discriminable tones. Capability for initation and selection of the tone for the emergency alarm is provided in the control room.

C. Plant offsite communication system uses the public telephone systems.

D. Other communication facilities are:

1. Microwave communications at 57 MHz.

2. Walkie-talkie with external antenna for radio contact between the control room, technical support center, emergency operations facility, and field personnel.

3. Security intercom linking control room, security subsystem, and security monitoring station.

4. Federal Telephone System (FTS) for the emergency notification system (ENS) and the health physics network.

Communications are further discussed in subsection 9.5.2.

18.1.2.4 Controls

Controls are defined as devices actuated by the control room personnel (manual mode) or by the systems (automatic mode) to influence system functioning.

A. The control devices used on the instrument panels are designed to minimize the size of the panels, while still allowing convenient and reliable operation. Insofar as is practicable, uniform types of control devices are used throughout the control room.

B. Control devices of redundant trains are designed for separation by physical barriers where there is an operational requirement for redundant trains to be close to each other.

C. Control devices for safety systems are designed to satisfy Seismic Category 1 requirements.

D. Control switches selected for the auxiliary board and panels are similar to those selected for the QMCBs (Electroswitch series 20 or equivalent).

E. Control switch development is designed to comply with the following:

1. Left-right (right throw) or bottom-top (up-throw)	Start, close (circuit breaker), open (valve), on, raise, and other positive or increasing functions and associated indicating lights.
2. Right-left (left throw) or top-bottom (down-throw).	Stop, trip (circuit breaker), close (valve), off, lower, and other negative or decreasing functions and associated indicating lights.

F. Where possible, automatic controls for parallel valves are designed to be kept in parallel (left to right). The series valves are designed to be kept in series

18.3.4.3.2 Engineering Consensus

The development and selection of corrections to HEDs was accomplished by group meetings of the detailed CRDR team. Alternate approaches, benefits, and costs were discussed, and an engineering consensus was arrived at on the recommended approach to correct each HED. The diverse backgrounds of team members were intended to provide input from all disciplines on the resolution of HEDs. The procedures in use called for final recording of the board recommendations. Brief summaries of the resolutions and supporting comments were reported in the Detailed CRDR Report of June 10, 1986.

18.3.4.3.3 Cost-Benefit Analysis

The engineering consensus approach considered costs vs. benefits but did not perform a detailed, documented cost-benefit analysis. The emphasis was on correction of the HEDs to enhance operator performance. The correction of safety significant HEDs was a commitment without regard to cost. In all cases the team members sought to develop the most practical solution balancing constructability, schedule, and cost to achieve the objective of enhanced operator performance.

Some example cost considerations were:

A. Labeling and procedure changes were always implemented to achieve compliance with NUREG 0700. These were typically less than $10,000-projects.

B. Panel rearrangements generally less than $100,000 were implemented.

C. A major control room layout modification of $300,000 was implemented.

D. Major control panel replacements which would cost several million dollars were not recommended. HEDs involving such changes for exact compliance with NUREG 0700 guidelines were addressed with alternate solutions.

18.3.5 COORDINATION WITH OTHER ACTIVITIES

The CRDR process was coordinated with other post-TMI activities in several ways. These activities included the following:

- Upgrading Emergency Operating Procedures.
- Detailed control room design review.

- Post-accident monitoring system (section 7.5).
- Safety parameter display system (subsection 9.5.10).
- Upgrading emergency response facilities.

The mechanism for coordination of control room improvements with other programs was the Emergency Operating Procedures validation program.

The coordination was achieved by: (1) having several members of the detailed CRDR team serve as members of the EOP observation teams during the EOP validation exercises, (2) the task analysis was a common basis for both the detailed CRDR and EOP programs, and (3) the findings or discrepancies identified during the validation exercises related to man and machine interface were processed via the detailed CRDR HED assessment process.

In addition, control room modifications that resulted from the Emergency Operating Procedures validation program were addressed in the detailed CRDR verification plan to ensure design improvements provided the necessary correction.

18.3.5.1 Emergency Operating Procedures

The task analysis portion of the detailed CRDR used the Westinghouse ERGs as plant-specific EOPs as its starting point. Thus, the task of upgrading emergency procedures is inherently integrated into the detailed CRDR. The simulator validation of the task analysis was combined with the VEGP EOP validation program.

18.3.5.2 Safety Parameter Display System (SPDS)

GPC used the SPDS during the EOP validation exercise, and evaluated it to NUREG 0700 Section 6.7 (process computer) guidelines. Some HEDs identified during the EOP validation exercise or the computer survey and judged to be significant by the review team were resolved by incorporating certain features into the SPDS and associated displays. In addition, the detailed CRDR team leader was chairman of a control room computer task force and participated in developing displays and board arrangements. This served to incorporate human engineering requirements into the design of the SPDS and further integrate them into the detailed CRDR process. (See the Emergency Plan, section H.4.5.)

18.3.5.3 Regulatory Guide 1.97

The design of Regulatory Guide 1.97 instrumentation was essentially complete at the time of this review. This instrumentation was evaluated in the detailed CRDR survey and task analysis.

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