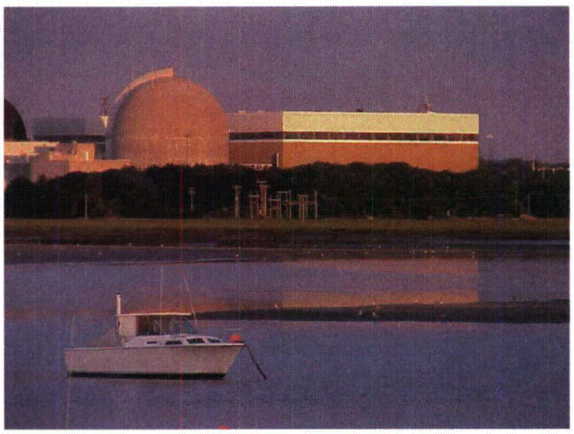
SEABROOK STATION
UPDATED FINAL SAFETY ANALYSIS REPORT

CHAPTER 10
STEAM AND POWER CONVERSION SYSTEM



10.1 SUMMARY DESCRIPTION

The steam and power conversion systems are those portions of the plant designed to transmit and convert the thermal energy produced in the reactor into electrical energy. The major components of the steam and power conversion system are located in the Turbine Building, and are not safety related. Design parameters of the steam and power conversion system components are presented in the applicable subsections which describe the major systems and components. The system components are shown on the general arrangement drawings in Section 1.2.

Steam at 1000 psia, 1191.8 H, 0.20 percent moisture is supplied from the outlet of four steam generators to drive a tandem-compound, six-flow exhaust, 1800-rpm turbine. Heat balances at the 100 percent rating and at the turbine design rating are shown in Figure 10.1-1 and Figure 10.1-2.

The turbine nameplate rating is 1,304,003 kW at 1.7 inches Hg absolute back pressure and zero percent makeup; the rating of the generator coupled to the turbine is 1,373,100 kVA at 75 psig H_2 pressure, 25,000 volts, 3 phase, 60 Hz and 0.94 power factor. The heat cycle results in a calculated gross T-G output of 1,305,600 kW at 100 percent load. Allowing for plant loads, the net plant output is approximately 1,254,600 kW.

During normal operation, main steam is taken ahead of the turbine stop valves to supply the single-stage reheaters and the turbine shaft sealing system. Crossover steam from the moisture separator-reheater outlets is supplied to drive two steam generator feedwater pump turbines. During startup or low load operation, main steam can be used to drive the steam generator feedwater pump turbines or the electric driven startup feed pump can be utilized.

Moisture separation with one stage reheat is provided between the high-pressure and low-pressure turbines for all steam entering the low-pressure turbines. Steam from the low-pressure turbines is condensed in a three-shell surface-type, two-pass condenser. Condensate is collected in condenser hotwells which are sized for a minimum of 3-minute storage capacity at full load. The condensate and Feedwater System returns feedwater to the steam generators through six stages of extraction feedwater heaters.

Circulating water (sea water) is pumped through the main condenser and returned to the ocean to dissipate the remaining unusable heat from the steam and power conversion system.

Disposal of heat from the Reactor Coolant System following sudden load rejections or trip of the unit is handled by the Steam Dump System. A detailed description of the Steam Dump System is contained in Subsection 10.4.4. The steam generators are also utilized as the heat sink for reactor heat decay, by absorbing heat from the Reactor Coolant System and producing steam. The steam may be bypassed around the main turbine and condensed in the condenser or vented to atmosphere if the condenser is not available.

The steam and power conversion system provides load-following capability at step load changes of plus 10 percent and minus 15 percent and ramp load changes of ±5 percent per minute over the load range of 15 to 100 percent reactor output. The system can accept a 50 percent load reduction without a reactor trip. The Steam Dump System is capable of bypassing nominally 40 percent of the full-load steam flow. This allows for shutdown from half-load or controlled reactor runback without atmospheric venting of steam through the steam generator relief valves. The Steam Dump System opens automatically to the extent necessary during rapid load reductions to remove excess heat from the Reactor Coolant System and closes as operating conditions stabilize at the new load.

Safety valves are provided on the main steam lines from steam generators, and the steam (shell) side of feedwater heaters and moisture separator-reheaters. Diaphragms are provided in the exhaust sections of the low pressure turbines for overpressure protection of the turbine exhaust sections and the main condensers.

The individual components of the steam and power conversion system are based on proven conventional design, acceptable for use in large central power generating stations. The turbine plant auxiliary equipment is selected to provide the optimum operating economy with maximum safety and reliability. All auxiliary equipment is specified for a design capability corresponding to the turbine design rating. Adequate design margins are included as required for wear and system surges to provide dependable service.

10.2 TURBINE GENERATOR

10.2.1 Design Bases

The Turbine Generator System is designed to receive the steam output of the nuclear steam supply system and convert its thermal energy into electrical energy. When operating at 100 percent of licensed power, the turbine generator has a nameplate output of 1,304,003 kW with throttle steam conditions of 975 psia, 1192 Btu/lb and a back pressure of 1.7" Hg abs.

The turbine generator is intended for base load operation, but is capable of step load changes varying from 20 percent at one-quarter load to 67 percent at full-load, and ramp load changes up to 10 percent/minute increasing and 30 percent/minute decreasing. This is compatible with the nuclear steam supply operation limitations, which are plus 10 percent and minus 15 percent maximum step load changes and ±5 percent/minute ramp load changes, over the load range from 15 percent to 100 percent power.

The turbine generator is equipped with instrumentation to continuously monitor and alarm all significant variables, such as speed, load, pressures, temperatures, valve positions, thermal expansion movements, and shaft eccentricity and vibration.

The Turbine Generator System is a nonnuclear system, with associated piping designed in accordance with either manfacturer's standards or the Power Piping Code, ANSI B31.1. Pressure-containing vessels are designed in accordance with the ASME Boiler and Pressure Vessel Code, Section VIII, Division 1.

10.2.2 Description

The turbine generator and related auxiliary equipment, piping and pressure vessels are shown on the turbine building general arrangement drawings in Section 1.2. Turbine generator general auxiliaries and other equipment in the Turbine Building are cooled by the Secondary Component Cooling Water System (see Figure 10.4-14).

10.2.2.1 Turbine

The turbine is an 1800 rpm, tandem compound with six flow, low pressure stages and 43-inch last stage blades. The turbine consists of one high pressure double-flow section that exhausts into four single-stage reheat MSRs then to three low pressure double-flow sections.

Steam from the main steam header is admitted to the turbine through four angle-body control valves. The main stop valves are welded directly to the inlet nozzles of the control valves. The stop valve below seat chambers are connected by an equalizing line. An internal bypass valve is provided in one of the main stop valves to provide pre-warming steam to the stop and control valves bodies, as well as the turbine. The bypass valve also provides pressure equalization across the stop valve seats, which is required prior to opening the stop valves.

Steam leaving the high pressure turbine has a moisture content of approximately 14 percent. To improve turbine efficiency and reduce low-pressure turbine exhaust moisture, thus minimizing maintenance on the low-pressure blading, the steam exhausted from the high pressure turbine is passed through four moisture separator-reheaters in parallel.

Within the moisture separator-reheater, the moisture is removed from the wet exhaust steam, and the dried steam is reheated to approximately 530°F by condensing high pressure steam in a tube bundle. The reheated steam is admitted to the three low-pressure cylinders of the turbine, from which it exhausts into three individual shells of the main condenser.

The steam and water contained in the moisture separator-heater and cross-around piping would, on loss of load, tend to accelerate the turbine. To prevent this occurrence, combined stop and intercept valves are provided in the cross-around lines, at the inlet of the low pressure turbines. Relief valves which discharge to the condenser are provided on the moisture separator-reheaters to prevent overpressure in the cross-around system.

The turbine and generator bearings are lubricated by a conventional pressurized oil system. A main lubricating oil pump, driven by the turbine shaft, provides the bearing lubricating oil during normal operation. During startup or shutdown, bearing lubricating oil is supplied by AC motor-driven pumps. In addition, a DC motor-driven emergency oil pump provides bearing lubricating oil in case of loss of site power.

The main steam and intermediate reheat valves are opened by a 1600 psig hydraulic fluid system which is totally independent of the bearing oil system. These valves are closed by springs and steam forces upon depressurization of the hydraulic fluid system. The valve actuation system is such that loss of hydraulic fluid pressure for any reason leads to valve closing and consequent unit shutdown.

The turbine valve closure times are as follows:

Turbine main steam stop valve	0.15 seconds
Turbine main steam control valve	0.19 seconds
Intermediate stop valve	0.20 seconds
Intercept valve	0.17 seconds

The turbine valves will be tested for proper operation for overspeed protection at the frequency specified in the Technical Requirements Manual with any of the DEHC control workstations on the Main Control Board, Secondary Operator Workstation Console(s), or the Engineering Workstation in the Computer Room, as appropriate. These tests demonstrate:

a. full closure of the main stop valves, control valves and combined intermediate valves individually including capability for fast closure

b. proper operation of the Primary Overspeed and Emergency Overspeed functions

10.2.2.2 Generator

The generator is sized to accept the gross rated output of the turbine at rated steam conditions. It is a directly coupled, four pole, 1800 rpm, 60 Hz, 3 phase, 25,000 volt, hydrogen and water cooled unit, rated at 1373.1 MVA at 0.94 power factor and 75 psig hydrogen pressure, with a short circuit ratio of 0.52.

All the generator internal components, except the stator winding, are cooled by hydrogen which is contained within the generator frame and circulated in a closed loop by two fans mounted at the end of the rotor (see Figure 10.2-1). The heat absorbed by the gas is removed in two hydrogen coolers mounted directly on the generator frame by secondary component cooling water.

The bulk storage facility for the hydrogen gas will be located in the yard at coordinates 20,700 N and 78,800 E. In accordance with NFPA-50A requirements, this facility is located not less than 50 feet from any occupied building on the site.

All piping from the storage area to the respective buildings, i.e., Waste Processing Building for Reactor Coolant System use and Unit 1 Turbine Building for generator cooling, is routed underground and either enters the building above grade or is encased in a sleeve with a vent to atmosphere upon entering the building below grade (see Figure 10.2-1).

When filling the generator casing, the casing is first purged of air using an inert gas (carbon dioxide) to avoid an explosive hydrogen air mixture. Carbon dioxide is also used during the purging of the casing of hydrogen. During these operations, an analyzer is used to maintain a safe mixture. During normal operations a control panel monitors the generator's hydrogen system status with alarm points for hydrogen purity, high and low temperature.

The stator windings are cooled by de-ionized water circulating in a closed loop between the generator and a generator stator cooling water unit on the ground floor. The heat absorbed by the de-ionized water is removed in a heat exchanger by the secondary component cooling water. Failure of the Stator Cooling Water System initiates a unit power runback which reduces power to 22 percent.

10.2.2.3 Steam Extraction Connections

Turbine steam extraction connections are provided for six stages of feedwater heating. Steam is extracted from one stage of the high pressure turbine, from the high pressure turbine exhaust piping, and from four stages of the low pressure turbines. A combination of positively assisted check valves in the extraction steam lines and automatically controlled heater drain valves protects the turbine against water induction. Check valves are provided in extraction steam lines 3 through 6. There are no check valves in extraction steam lines 1 and 2, since these lines are located within the condenser neck. However, in all cases, the combination of valving and heater drain valve controls is such that no single equipment failure will result in water entering the turbine. The check valves in extraction steam lines 3 through 6 will also provide additional protection against turbine overspeed following a load reduction. The positive-assist action on the check valves is provided by spring-load air actuators. Periodic testing verifies that the operating pistons are free to move under the action of the spring when the air pressure is released.

10.2.2.4 Automatic Controls

The automatic control functions are programmed to protect the Reactor Coolant System with appropriate corrective actions, as explained in Chapter 7. The turbine is tripped every time the reactor is tripped. A reactor trip is initiated upon a turbine trip above approximately 45 percent of full power.

The turbine generator is controlled and protected by a Digital Electro-Hydraulic Control System (DEHC) that combines digital micro-processor based electronics and high-pressure hydraulic components to control the steam flow through the turbine.

The DEHC system utilizes a triple modular redundant (TMR) system architecture, consisting of three independent processors that employ Software Implemented Fault Tolerance (SIFT) technology. The system process sensors (with the exception of Main Steam Throttle Pressure), output actuators, I/O cards, and communication devices are also triplicated, to provide a fault tolerant design that supports high reliability and on-line maintenance. The single failure of any DEHC component will not result in a turbine trip, or loss of steam valve control that results in a loss of generation.

The DEHC Turbine Controller system consists of the following subsystem functions:

a. Speed Control Function

   The speed control function controls turbine speed over the entire speed range including to a speed high enough to test the protective overspeed devices. The speed control function uses magnetic speed sensors that provide input to TMR processors running the speed function software and protection algorithms. It is coordinated with load rejection functions and the trip system. The system is accurate enough to allow synchronization to the power grid. Should the main turbine speed increase for any reason, the speed control function reduces the main turbine speed to rated turbine speed by closing the control valves.

During turbine acceleration to rated speed, a wobbulator function is provided in the speed control function to prevent continuous constant speed operation near turbine bucket resonant frequencies. At rated speed, a speed error deadband feature is provided to mitigate the effect of frequency fluctuations on turbine load.

b. Load Control Function

The purpose of the load control function is to combine the load reference setpoint with inputs from the other control functions and calculate the resulting flow demand reference. The flow demand or control valve reference is derived through a series of minimum value gates, which selects the lowest reference to control the valves. The valve position limit is used as an upper limit on the original control valve reference as determined by speed and load.

Elements of Load Control include:

Load Reference and Rate – The load reference corresponds approximately to the desired load based on capability of the steam valves at rated main steam pressure. Load reference is a combination of both a load rate and load target. Thse have been scaled for interface to the operator, to approximate rated thermal power. The system has fixed and variable loading rates. Load reference increase is inhibited by the C16 permissive.

Load Limit – A load limit function is provided that overrides the CV flow demand signal in a decreasing direction. Load limit may be either automatic, as in setbacks, or manual by the operator.

c. Flow Control Function

The flow control function translates the desired flow signals into valve stem position signals for the hydraulic servo actuators that position the four main steam stop and control valves and the six intermediate reheat intercept and stop valves. This includes corrective functions to compensate for the nonlinearity of the valves. In normal operation, the servo actuators of the main steam control valves gradually adjust the position of the control valve stem in response to variations of a position signal which depends on the flow signal from the load control function, while the main steam stop valves and the intercept and intermediate stop valves are fully open.

d. Valve Testing

The control system provides for the capability to demonstrate the closure of all turbine stop, control and combined intermediate valves at power or during shutdown one at a time (or in the case of the combined intermediate valves one set at a time). Valve testing demonstrates both normal closing speeds and fast closure near the end of travel. First stage pressure feedback is provided during on-line testing of control valves 1, 2, and 3 to open the control valves not being tested to compensate for the closing of the control valve being tested.

e. Overspeed Protection

The overspeed protection system consists of primary and emergency electronic overspeed protection systems. The primary electronic overspeed system is part of the normal speed control system. When functioning properly, the primary overspeed protection system prevents a turbine trip due to overspeed. The primary overspeed protection functions include normal speed control, Power Load Unbalance (PLU) and the Intercept Valve (IV) Trigger function. If these functions fail to limit overspeed, the primary overspeed protection system will provide a turbine trip at 1980 rpm (110% of rated speed).

PLU is a rate-sensitive function that is provided to initiate fast closing of the control valves and intercept valves under load rejection conditions that might lead to rapid rotor acceleration and consequent overspeed. The PLU action on valves occurs when turbine power exceeds generator load by approximately 40% and generator current is decreasing at a rapid rate. Intermediate steam pressure is used as a measure of turbine power and generator current is used as a measure of generator load. This rapid action is intended to limit shaft overspeed following a loss of generator load to a level below the setpoint of the emergency overspeed trip.

The IV Trigger is used to limit the effect of the volume of steam in the MSRs and associated steam piping in a load rejection. The IV trigger compares the error between the IV position reference and the actual position to the maximum allowable error and will fast close the IVs. After position error has returned to the normal range, the IVs are returned to normal control.

The Emergency Overspeed Trip function is totally redundant to and independent of the Primary Speed function. This function is realized by a separate protection module (located in the front standard DEHC cabinet) that consists of two sets of TMR processors independent of the primary speed function that receive input from three additional (i.e. not shared with the primary speed function) magnetic speed pickups. If the protection modules detect a 2/3 logic condition at the Emergency Overspeed setpoint of 1989 rpm (110.5% of rated speed), the system will generate a turbine trip signal.

The DEHC system will alarm upon a loss of any speed input signal and initiate a trip if the control system is incapable of adequately determining speed.

The main line of defense for turbine overspeed protection is the closure of the main steam valves. To further reduce the risk of turbine overspeed following a turbine trip, the sequential trip circuit that normally trips the generator breaker includes a main steam valve closed position interlock in series with a reverse power relay interlock. This ensures that all steam valves are closed and the generator motors for a set time delay prior to tripping the generator breaker. Refer to Section 8.2.1.3.e.3 for a more detailed description of generator breaker tripping.

f. Emergency Trip System

The emergency trip system is a high-pressure hydraulic fluid system that, when pressurized, permits opening of all turbine main steam and intermediate valves by the electro-hydraulic system, and, when depressurized, causes them to be rapidly closed by spring action. Upon reaching the overspeed trip set level(s), the depressurization of the emergency trip system is triggered by either or both of the following redundant overspeed protection functions:

1. Turbine Primary Overspeed trip set at 110 percent speed (1980 rpm).
2. Turbine Emergency Overspeed trip set at one-half percent higher than the Primary Overspeed trip (110.5% speed, or 1989 rpm).

Both the Primary and Emergency Overspeed trip systems each consist of three redundant magnetic speed pickups which monitor turbine speed and provide inputs to the respective control systems. The redundant overspeed trip systems initiate the trip outputs to the Turbine electrical trip devices (ETDs). There are six ETDs arranged in two sets of two-out-of-three hydraulic trip circuits. The trip circuit outputs de-energize the ETDs; when two-out-of-three of either set of the ETDs are de-energized, hydraulic fluid is dumped from the emergency trip system, resulting in a turbine trip. Test controls are provided to allow on-line testing of the individual (ETDs). During on-line testing of the ETDs, hydraulic fluid pressure is maintained since the design of the hydraulic trip system requires the operation of two-out-of-three ETDs to cause a turbine trip. During testing of an ETD, overspeed trip protection is maintained via the remaining ETDs not under test.

In addition to the redundant overspeed trip functions described above, each of the following devices or signals will actuate the Emergency Trip System resulting in a turbine shutdown:

<u>Process Related Trips</u>
Main Shaft Oil Pressure Low
Bearing Oil Pressure Low
Exhaust Vacuum Low
Exhaust Hood Temperature High
EHC Pressure Low
GSC Runback Failed
MSR Level High
Axial Position (Thrust Brg. Wear) Trip
ETS Pressure Low

<u>Speed Related Trips</u>
Primary Overspeed
Emergency Overspeed
Excessive Acceleration/Deceleration Rate
Speed Signal Faults

<u>External Trips</u>
Manual Trip (Control Room)
Manual Trip (Front Standard)
Turbine Vibration High
Generator Lockout Relay Actuated
Generator Breaker Trip (via control switch)
Train A SI/Reactor Trip/SG Level HIHI
Train B SI/Reactor Trip/SG Level HIHI

Direct Hardware Based Trips (Do not Rely on Software)
Manual Trip (Control Room)
Manual Trip (Front Standard)
ATWS Mitigation Trip

10.2.3 Turbine Disk Integrity

10.2.3.1 Materials Selection

Turbine wheels and rotors were made from electric furnace steel, vacuum poured, with carefully controlled quench and temper heat treatment to provide adequate fracture toughness. Samples from each forging were tested to ensure that detrimental elements such as sulfur and phosphorus are kept to low levels (0.015 max.) to ensure long-life fracture toughness for the environment in which the parts operate. Forgings were also subjected to a thorough ultra-sonic inspection to detect and identify any discontinuities present. All turbine wheel and rotor materials have the lowest fracture appearance transition temperatures (FATT) and highest Charpy V-notch (0°F) energies obtainable, on a consistent basis from water-quenched Ni-Cr-Mo-V material at the sizes and strength levels used. Since actual levels of FATT and Charpy V-notch energy vary, depending upon the size of the part and the location within the part, etc., these variations were taken into account in accepting specific forgings.

10.2.3.2 Fracture Toughness

The successful operation of a turbine generator depends on proper startup, loading, shutdown, and load-changing procedures to reduce thermal stresses, distortions, vibration and rotor shell differential expansion. These instructions are included in G.E. Company Starting and Loading Instructions.

Suitable material toughness was obtained through the use of materials described under Subsection 10.2.3.1, to produce a balance of adequate material strength and toughness, to ensure safety, while simultaneously providing high reliability, availability and efficiency during operation. Bore stress calculations include components due to centrifugal loads, interference fit and thermal gradients, where applicable. The ratio of material fracture toughness, K_{IC}, (as derived from materials tests on each wheel or rotor), to the maximum tangential stress is at least $2\sqrt{\text{inch}}$ for wheels and rotors at speeds from normal to 115 percent of rated speed, although the highest anticipated speed resulting from a loss of load is 110 percent. Adequate material fracture toughness needed to maintain this ratio is assured by destructive tests on material taken from the wheel or rotor, using correlation methods which are more conservative than that presented by J.A. Begley and W.A. Logsdon in Westinghouse Scientific Paper 71-1E7-MSLRF-P1.

10.2.3.3 High Temperature Properties

The stress-rupture properties of the high-pressure rotor materials and the methods of obtaining these properties are considered to be proprietary information by the turbine manufacturer.

10.2.3.4 Turbine Design

The turbine assembly is designed to withstand normal conditions and anticipated transients including those resulting in turbine trip without loss of structural integrity. A discussion of potential turbine missiles is given in Section 3.5. The design of the turbine assembly meets the following criteria:

a. Turbine shaft bearings are designed to retain their structural integrity under normal operating loads and anticipated transients, including those leading to turbine trips.

b. The multitude of natural critical frequencies of the turbine shaft assemblies existing between zero speed and 20 percent overspeed is controlled in the design and operation to avoid distress to the unit during operation.

c. The maximum tangential stress in wheels and rotors resulting from centrifugal forces, interferences fit and thermal gradients will not exceed 0.75 of the yield strength of the materials at 115 percent of rated speed.

d. The basis for turbine design overspeed is 5 percent above the normal setting of the emergency governor, which is 110 to 111 percent of rated speed. The design overspeed is 115 percent of rated speed.

e. The turbine discs are designed so that in-service inspections can be performed without removal of the discs from the shaft. High stress areas, such as keyways, can be ultrasonically inspected in regions under the wheel hubs as well as under the hub.

10.2.3.5 Pre-Service Inspection

The pre-service inspection program was as follows:

a. Wheel and rotor forgings were rough machined with minimum stock allowance prior to heat treatment, with the finish-machined wheel and rotor then subjected to 100 percent volumetric (ultrasonic), surface, and visual examinations, using General Electric acceptance criteria. These criteria are more restrictive than those specified for Class 1 components in the ASME Boiler and Pressure Code, Sections III and V, and include the requirement that subsurface sonic indications be removed, or evaluated to assure that they will not grow to a size which will compromise the integrity of the unit during the service life of the unit.

b. All finish-machined surfaces were subjected to a magnetic particle test, with no flaw indications permitted.

c. Each fully bucketed turbine rotor assembly was spin-tested at or above the maximum speed anticipated following a turbine trip from full load.

10.2.3.6 In-Service Inspection

a. The in-service inspection program for the turbine assembly includes the following:

Disassembly of the turbine at approximately 10-year intervals, with initial inspection after 6 years of service, during plant shutdown coinciding with the in-service inspection schedule required by the ASME Boiler and Pressure Vessel Code, Section XI, and complete inspection of all normally inaccessible parts, such as couplings, coupling bolts, turbine shaft, low pressure turbine buckets, low pressure wheels, and high pressure rotors.

This inspection will consist of visual, surface, and volumetric examinations, as indicated below:

1. A thorough volumetric examination will be conducted of all low pressure wheels and high pressure rotors, including areas immediately adjacent to keyways and bores. This examination is predicated on the development of suitable remote inspection equipment.
2. Visual examination of all accessible surfaces or rotors and wheels
3. Visual and surface examination of all low pressure buckets
4. 100 percent surface examination of couplings and coupling bolts.

b. The in-service testing and inspection of main steam and reheat valves is identified in the Technical Requirements Manual.

c. The extraction steam check valves described in Subsection 10.2.2.3 will be subject to monthly testing. The monthly test will check the operation of the power assist actuator and control components and for valve disc integrity.

10.2.4 Evaluation

The turbine-generator and its auxiliary systems are non-Nuclear Safety Class, and there are no safety-related systems, or portions of systems, located close to the turbine-generator. The turbine overspeed control system equipment is located in the front standard on the operating floor. The turbine is tripped by low hydraulic fluid pressure, so a hydraulic line break caused by high or moderate energy pipe failure would have the same result. A separate Emergency Overspeed trip function module including separate speed sensors provides additional protection against turbine overspeed, and acts independently of the Primary Overspeed trip function.

There is normally no radioactivity in the secondary system. In the event of a steam-generator tube leak, however, radioactivity can be present in the secondary system. The amount of radioactivity in the secondary system under this condition is a function of the level of activity the Primary Coolant System and the amount of tube leakage. See Chapter 11 for an estimate of the activity level in the secondary steam system due to tube leakage.

The steam-generator blowdown processing system is described in Subsection 10.4.8.

Radiation levels in the Turbine Building are discussed in Chapters 11 and 12.

10.3 MAIN STEAM SUPPLY SYSTEM

10.3.1 Design Bases

a. The Main Steam Supply System is designed to:

1. Conduct steam from the steam generators to the main turbine, feed pump turbine drives, emergency feed pump turbine drive, reheaters, turbine gland sealing system, Steam Dump System and the Auxiliary Steam System

2. Control steam generator pressure during startup and shutdown and while the condenser is not available

3. Provide over-pressure protection for the steam generators

4. Isolate the Containment from the Main Steam Supply System and provide for main steam line warmup

5. Provide a means to dissipate the heat generated in the Nuclear Steam Supply System, during all modes of normal operation including a turbine trip at full load. Heat removal is discussed in Section 10.4.

b. Safety-related portions of the system will function, as required, during all operating conditions, including adverse environmental occurrences, accidents, and loss of offsite power, in the event of a malfunction or failure of an active component.

c. Pipe cracks or breaks, including pipe whip, in high and moderate energy piping will not preclude essential functions of safety-related portions of the system.

d. The system design provides for functional testing of safety-related components and in-service inspection of safety-related portions of the system.

e. The system has the capability to detect, control and isolate system leakage and preclude accidental release to the environment.

f. The portions of the piping from the steam generators to the five degree restraints downstream of the main steam isolation valve outside of the Containment meet seismic Category I criteria, and are designed in accordance with the ASME Code Section III, Division I (1971), class 2. Piping downstream of the five degree restraints is designed in accordance with the Power Piping Code ANSI B31.1-1967.

g. Other design considerations are discussed in the following sections:

Protection Against Natural Phenomena	Section 3.1
Environmental and Missile Effects	Section 3.1
Quality Group and Seismic Design Classification	Section 3.2
Wind and Tornados	Section 3.3
Flooding	Section 3.4
Missile Protection	Section 3.5
Pipe Rupture	Section 3.6
Seismic Design	Section 3.7
Environmental Design	Section 3.11

10.3.2 Description

10.3.2.1 Piping

The Main Steam System and related piping and instrumentation diagrams are shown in Figure 10.3-1, Figure 10.3-2, Figure 10.3-3, Figure 10.3-4, Figure 10.3-5, Figure 10.3-6, Figure 10.3-7, Figure 10.3-8 and Figure 10.3-9. The extraction steam subsystem is shown on Figure 10.3-10 and Figure 10.3-11. For flow, pressure and enthalpy values at various points in the Main Steam System, see Figure 10.1-1, Heat Balance.

Four 30-inch main steam lines carry steam from each of the four steam generators to the 48-inch main steam header. Each line includes a flow restrictor which is an integral part of the steam generator nozzle, a power-operated atmospheric relief valve, five safety valves and one isolation valve. Two 6-inch lines carry steam from two of the four 30-inch lines to the trip and throttle valve of the emergency feed pump turbine drive.

Four 30-inch turbine supply lines carry steam from the 48-inch main steam header to the turbine stop valve. Two 24-inch turbine supply lines carry steam from the 48-inch main steam header to the Auxiliary Steam System, main feed pump turbine drives, reheaters and steam dump valves.

The entire Main Steam Supply System is designed for 1200 psia at 600°F. The design of the portions of piping with safety valves considers the dynamic forces from the opening of the safety valves. Safety valves settings are shown in Figure 10.3-2. The spacing of the safety valves on the main steam lines is in compliance with Reg. Guide 1.67 and referenced Code Case 1569.

The main steam lines are provided with sufficient drainage for startup and normal operation. Low point drains are valved and returned to the condenser.

All components except the flow restrictors and portions of the 30" lines are located outside of the Containment.

10.3.2.2 Flow Restrictor

The primary function of the flow restrictor is to limit the flow from a steam generator in the event the main steam pipe ruptures downstream of the restrictor (see Section 5.4). Also, the flow restrictors are the primary elements for the steam flow input signal to the Feedwater Control System.

10.3.2.3 Penetrations

Because of the high temperatures involved, and the restraint imposed by the Containment, a separate penetration is provided for each main steam line. The penetrations are manufactured in accordance with the ASME Code, Section III, Class 2 and MC requirements. For a complete description of penetrations see Subsection 3.8.2.

10.3.2.4 Atmospheric Relief Valve

A power-operated atmospheric relief valve (ARV) is provided in the 30-inch line from each steam generator. These valves provide for controlled removal of reactor decay heat during reactor cooldown, plant startup, and after a turbine trip, when the condenser and/or the turbine bypass system are not available.

The atmospheric relief valves are located adjacent to the main steam safety valves described in Subsection 10.3.2.6. The safety valves will operate without plant operator action for an indefinite period, and will maintain the main steam pressure between 1185 psig and 1225 psig during the hot standby condition. When available, the atmospheric relief valves can be used to reduce main steam pressure for both hot and cold shutdown conditions.

Each ARV automatically regulates its respective steam generator outlet header pressure to approximately 1140 psia. The valves are capable of automatic operation over the steam pressure range of 1300 psia to 125 psia, when the Residual Heat Removal (RHR) System is put into operation. Manual operation of the valves' controllers will allow atmospheric relief down to atmospheric pressure. The nominal capacity of each valve is 531,000 lbs/hr at 1140 psia inlet pressure, for a total combined capacity of 10 percent of the maximum steam flow. The maximum capacity of each valve does not exceed 970,000 lbs/hr at 1200 psia inlet pressure, to limit heat release if a valve inadvertently opens.

Operation of the ARV can be either automatic pressure control or manual position control from the main control board. The valve can also be operated from the remote shutdown panel and locally. After a seismic event, the valves can be manually controlled from the control room or the remote shutdown panel. The backup high pressure gas supply is discussed in Subsection 9.3.1.1.

The valve has a stroke time of less than or equal to 70 seconds, and fails in the closed position. "Full open" and "full closed" position indication lights are located in the control room, and on the remote shutdown panel. Each valve is an 8"x10" ANSI Class 900# globe type valve.

A stop valve upstream of each ARV is provided for isolation. Each valve discharges to the atmosphere through a restrictor (silencer) sized and supplied by the ARV manufacturer. The ARVs were designed, fabricated and inspected in accordance with the ASME Boiler and Pressure Vessel Code, Section III, Division 1, Class 2. The silencer is classified NNS.

10.3.2.5 Emergency Feed Pump Turbine Supply

Two pneumatically operated valves are installed in each of the two branch connections from steam generators E-11A and E-11B. Valves MS-V393 and MS-V394 are containment isolation and EFW steam supply isolation valves. These valves are provided with seismically designed air supplies to ensure closure from the control room in accordance with GDC-57 criteria. These branch valves are redundant, and either branch connection will satisfy the emergency feed pump turbine drive (EFPTD) steam requirements. The branch connections feed a common header that contains a pneumatically operated steam supply isolation valve (MS-V395) located upstream and adjacent to the trip and throttle valve of the EFPTD. For actuation of the steam supply isolation valves and a full description of the Emergency Feed System, see Section 6.8.

10.3.2.6 Safety Valves

Five spring-loaded, self-actuated safety valves on each 30-inch steam generator outlet line provide overpressure protection for the secondary side of the steam generators, and consequently for the main steam piping. The total capacity of the twenty valves is approximately 110 percent of full-load steam flow at a pressure not exceeding 110 percent of the steam generator shell side design pressure. The maximum capacity of any single valve does not exceed 970,000 lbs/hr at an inlet pressure equal to the steam generator shell design pressure, to limit heat release if a valve inadvertently sticks open.

The valves are set at 1185 to 1225 psig with capacities ranging from 893,160 to 922,950 lbs/hr, respectively. These safety valves are designed, fabricated and inspected in accordance with the ASME Boiler and Pressure Vessel Code, Section III, Division 1, Class 2. These valves may be utilized for safe shutdown of the plant and are classified "active."

10.3.2.7 Main Steam Isolation Valves

One main steam isolation valve (MSIV) with a bypass valve is provided on each steam generator main steam line. The bypass valve is used at startup prior to opening the MSIV for pressure equalization and warming of the main steam piping system. During normal operation, the bypass valves are locked closed with the breakers locked open. The MSIV provides positive shut-off of steam flow in either direction during emergency as well as normal operation. The MSIV is a gate valve actuated by a hydraulic/pneumatic actuator. Hydraulic fluid is pumped into the valve actuator to open the valve against a pressurized pneumatic system. The valve is closed by pneumatic pressure when the hydraulic fluid pressure is relieved. The actuator is a stored nitrogen unit with a hydraulic cylinder coupled directly to a precharged nitrogen accumulator which stores the closing energy. The precharged high pressure nitrogen is stored in an integral, essentially spherical accumulator which is designed as a pressure vessel meeting the requirements of ASME VIII, Div. 1. A dual hydraulic control system is provided to ensure reliability. The MSIV is also capable of being tested online by partial closure of the valve.

For the MSIV (and FWIV) the precharged nitrogen accumulators are integral to the actuator assembly, and are designed to seismic Category I requirements. Separate instrument air accumulators are not required. Compressed air is supplied to air motors to operate the hydraulic pumps to open the MSIVs, but this is not a safety function. Motor-driven pumps perform a similar function for the FWIVs. Pressure switches are provided to alarm on low accumulator gas pressure. The accumulator is designed to minimize gas leakage. If gas pressure does decrease, it is a maintenance function to recharge the accumulator using nitrogen supply.

The MSIV is designed, fabricated and inspected in accordance with the ASME Code Section III, Class 2, and is classified "active." The safety-related electrical components of the actuator are Class 1E.

10.3.2.8 Turbine Stop Valves

A description of these valves in included in Section 10.2.

10.3.3 Safety Evaluation

The main steam cycle provides for heat removal from the NSSS. After a load reduction or turbine trip, this function is provided for by the Steam Dump System, which discharges into the condenser (see Subsection 10.4.4). When the condenser is not available, the atmospheric relief valves permit control of steam generator pressure.

The main steam safety valves are sized to prevent overpressure from load rejection at full power operation.

For a discussion of the increase in heat removal from the Reactor Coolant System, due to inadvertent opening of a main steam dump, relief, or safety valve, see Subsection 15.1.4.

In the event of a main steam line break, the MSIVs will close within five seconds from receipt of signal. The loss of main condenser vacuum has no effect upon the operation or operability of the main steam isolation valves.

Upon a loss of condenser vacuum the turbine stop valves close, the turbine is tripped and the turbine steam bypass valves are prevented from opening.

Reactor coolant system heat removal is accomplished by the steam generator atmospheric steam reliefs or the steam generator safety valves. In this event, the main steam isolation valves would remain open, which is an acceptable condition. For a discussion of increase in heat removal from the Reactor Coolant System due to a rupture of a main steam line, see Subsection 15.1.5.

Safety-related components of the Main Steam System are tested to insure operability under seismic and adverse environmental conditions. Redundant active components are provided to insure the system's safety functions in the event of a component failure.

Pipe restraints and the ability of components to operate under adverse environmental conditions provide protection from pipe breaks or cracks. The main steam and feed water pipe chases are designed to withstand a single-ended rupture of a main steam line.

The capability for testing safety-related components (Subsection 10.3.4) and provision for in-service inspection of the safety-related portion of the system (Section 6.6 and Subsection 3.9.6) are included.

Detection of radioactive leakage is facilitated by the area radiation monitoring system (Section 12.3), the process radiation monitoring system (Section 11.5), the primary coolant leakage detection system (Subsection 5.2.5), and the sampling systems (Subsection 9.3.2).

10.3.4 Inspection and Testing Requirements

An initial hydrostatic test of the main steam system piping is performed to the applicable construction codes. During preoperational testing the operation of the main steam safety valves and main steam isolation valves is demonstrated. Refer to Chapter 14 for further discussion.

Under normal plant operations, a program of in-service inspection of the welds in the safety class portion of the Main Steam System is performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code. Operability of main steam isolation and safety valves is verified in accordance with Technical Specification requirements.

A plant test program performed during initial pre-operational testing verified the ability to achieve plant cooldown using the manual-operated atmospheric relief valves.

10.3.5 Secondary Water Chemistry

10.3.5.1 Description

Secondary side water chemistry is established and maintained within the steam generator supplier's specification by:

a. Using a deaerating condenser that removes oxygen from the condensate

b. Chemical addition

c. Continuous blowdown of the steam generators

d. Cleaning the condenser and the Condensate and Feedwater Systems before startup, or during condenser circulating water inleakage, using the Condensate Polishing System.

When the steam is condensed, undissolved gases are released. These gases, including oxygen, are taken out by the condenser evacuation system. The oxygen content in the condensate leaving the condenser is controlled by procedure to minimize erosion-corrosion in the condensate and feedwater systems.

Additional oxygen control is accomplished by the addition of hydrazine at the condensate pump's discharge header or steam generator feedwater pump discharge piping. This hydrazine, by a chemical reaction, scavenges the oxygen in the feedwater. Maintaining a hydrazine residual near the steam generator feedwater inlet ensures that any dissolved oxygen is removed. Hydrazine also promotes the formation of a protective metal oxide film on carbon steel surfaces.

A combination of amine compounds is used as approved pH additive and added at the condensate pump's discharge header, the effluent of the condensate polishing system, or steam generator feedwater pump discharge piping. It is used to establish and maintain an alkaline pH in the feed train and the steam generator. Alkaline conditions reduce corrosion at elevated temperatures and promote the formation of a protective metal oxide film. The technique of adding hydrazine and volatile amine pH additives is known as All Volatile Treatment (AVT) because these chemicals will not concentrate in the steam generator. Condenser circulating water inleakage is monitored. The monitoring will detect low contaminant levels providing evidence of inleakage. The condensate polishing system may be used to minimize contaminant ingress to the secondary system while preparing to repair identified leaks. Water chemistry monitoring is discussed in Subsection 9.3.2.

The Seabrook Station secondary pH program is based on optimizing the "at" temperature pH (pHt) in various portions of the secondary plant by the use of amines. All amines suitable for use as secondary system pH control agents are classified as weak base compounds. When dissolved in water, these compounds partially ionize forming the hydroxide ion which is responsible for their alkaline properties. Temperature has a marked effect on the ionization constant of each amine depending on the amine itself. As a result the application of a single amine at a constant application rate will result in varying pHt in different regions of the secondary system. It may be necessary to employ a mixture as close as possible to the target pHt for that region in order to maintain the solubility of iron in that region at the lowest level possible.

The concentration of any contaminants that do enter the steam generator is reduced by intermittent and continuous blowdown. This is discussed in Subsection 10.4.8. The plant is equipped with a filtering system that recirculates a portion of the condensate through filters and returns it to the condenser. This procedure removes solid particles from the condenser and the Condensate and Feedwater Systems during low power operation.

10.3.5.2 Effect of Water Chemistry on Iodine Partitioning

The formation of volatile iodine compounds in the steam generator is suppressed by the condition of the secondary water chemistry. The amount of iodine carryover is normally dependent upon the efficiency of the moisture separators within the steam generators.

An approved pH additive is used to regulate the condensate pH. The pH range for effectively eliminating iodine volatility is 8.5-11.0. The "at" temperature pH for condensate is 9-10, maintaining iodine almost completely in its non-volatile state. Thus, only minimal amounts of iodine would be exhausted through the condenser vacuum pumps.

The amount of iodine released to the environment in the event of a steam generator tube rupture is minimal. This is the result of the high degree of iodine partitioning in the steam generator and condenser, and the charcoal filter at the condenser vacuum pump discharge. Refer to Subsection 15.6.3 for further discussion.

10.3.5.3 Control Program

The EPRI Secondary Water Chemistry Guidelines are the basis for the secondary chemistry control program.

A summary of operative procedures which are used for the steam generator secondary water chemistry control and monitoring program is as follows:

a. Procedures are available for sampling for the critical chemical and other parameters and of control points or limits for these parameters for each mode of operation: normal operating, hot startup, cold startup, hot shutdown, cold wet layup. The sampling schedule is expected to closely follow the recommendation of the EPRI Secondary Water Chemistry Guidelines. Critical parameters and specifications for each mode of operation are in accordance with the EPRI Secondary Water Chemistry Guidelines.

b. Procedures for chemical analysis of critical parameters were developed using references such as

   1. American Public Health Association, Standard Methods for Examination of Water and Waste Water

   2. American Society for Testing Materials, Part 31 Water

c. Locations for process instrumentation and sample points are indicted on Updated FSAR Figure 9.3-15. The extensive process instrumentation which monitors critical parameters of the secondary system will result in continuous assessment of the secondary system. Grab samples are taken at critical points (i.e., blowdown, feedwater, condenssate, and makeup) as additional verification of system chemistry control.

d. Procedures for recording and management of data are available. Seabrook Station will have qualified chemistry personnel on station at all times to interpret analysis data of critical parameters on a continuous basis, followed by additional technical review by chemistry department management during normal work hours. All analysis records will be maintained in accordance with station administrative procedures.

e. Procedures for defining corrective action are predicated on the need to maintain condenser integrity by using state-of-the-art techniques for leak detection such as helium-mass spectroscopy to identify air and seawater intrusion. Process instrumentation will result in rapid identification of leaks at Seabrook. Corrective action to identify the location of a verified condenser seawater intrusion in excess of the EPRI chemistry control parameters will be taken promptly. Use of the Condensate Polishing System will limit the chloride concentration in the steam generator blowdown. Actions to minimize degradation of steam generator tubes will be taken as described in the EPRI Secondary Water Chemistry Guidelines.

f. The Seabrook shift chemistry technician is the primary individual responsible to interpret operational chemistry data. Shift chemistry technicians will have completed all training required by the nonlicensed training program for chemistry technicians giving them the expertise to advise the Unit Shift Supervisor on operational chemistry occurrences.

10.3.6 Steam and Feedwater System Materials

10.3.6.1 Fracture Toughness

The test methods and acceptance criteria used to verify the fracture toughness of the ferritic materials used in Class 2 and 3 components of the Steam and Feedwater Systems are in accordance with the applicable requirements of Articles NC-2300 and ND-2300 in Section III of the ASME Boiler and Pressure Vessel Code, 1974 edition.

a. Fracture toughness properties of the steam generator pressure boundary materials are described in Subsection 5.2.3.3 of the Seabrook Updated FSAR. Impact testing of these materials has been performed to verify compliance with ASME III.

b. The main steam and feedwater isolation valves, containment penetrations and the piping between containment penetrations and isolation valves have been reviewed for compliance with General Design Criterion 51, and found to be acceptable. (See SER for Containment Boundary.)

c. The main steam system pipe and fittings inside Containment were fabricated from the materials listed in Subsection 10.3.6.2 below. All welded joints were examined radiographically to ensure minimum weld defects. Impact testing of this material was not considered necessary, since the maximum nil ductility transition temperature for these materials (conservatively taken from NUREG-0577 as 97°F considering the thickness adjustment) was below the minimum service temperature of 100°F established for the hydrostatic test fluid temperature.

The feedwater system pipe and fittings inside Containment and outside Containment up to the check valve beyond the isolation valve were fabricated from the materials listed in Subsection 10.3.6.2 below. All welds were examined radiographically to ensure minimum defects. The piping material, SA-106, was heat-treated to improve impact properties. Impact tests were performed on seven of the eight heats of piping material and met code requirements at the minimum feedwater injection temperature of 50°F.

10.3.6.2 Materials Selection and Fabrication

All Class 2 and 3 pipe, valves and fittings used in the Steam and Feedwater Systems are fabricated from materials that are listed in Appendix I of Section III of the ASME Code.

The following materials are used for Class 2 and 3 service:

Main Steam	Feedwater
SA-106, Grade B and C	SA-106, Grade B (normalized, fine grain)
SA-155, Grade KCF 70	SA-234
SA-234	SA-105
SA-105	SA-193, Grade B7
SA-193, Grade B7	SA-194, Grade 7, 2H, 4 or 3
SA-194, Grade 7, 2H 4 or 3	SA-312, Type 304
SA-216 WCB or WCC (valves)	SA-182, Type 304 SA-403, WP 304

Austenitic stainless steels are used only for the emergency feedwater suction piping. This is a low pressure, ambient temperature, uninsulated section of piping; therefore, Regulatory Guide 1.36, "Nonmetallic Thermal Insulation for Austenitic Stainless Steel," is not applicable.

Regulatory Guide 1.31, "Control of Ferrite Content in Stainless Steel Weld Metal," is applicable to Class 1 and 2 systems. The emergency feedwater piping is Class 3.

The recommendations of Regulatory Guide 1.44, "Control of the Use of Sensitized Stainless Steel," were complied with for the fabrication of emergency feedwater suction piping.

The recommendations of Regulatory Guide 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water Cooled Nuclear Power Plants" and ANSI N45.2.1-73, "Cleaning of Fluid Systems and Associated Components for Nuclear Power Plants" were complied with for cleaning and handling of all Class 2 and 3 components.

The preheat temperatures used for welding low alloy steel are in accordance with Regulatory Guide 1.50, "Control of Preheat Temperature for Welding of Low-Alloy Steel." The preheat temperatures used for welding carbon steel materials are in accordance with ASME Code, Section III.

The steam and feedwater system components are provided with sufficient accessibility so that standard welding procedures are utilized. Nondestructive examination procedures used for tubular products conform to applicable requirements of the ASME Code.

10.4 OTHER FEATURES OF STEAM AND POWER CONVERSION SYSTEMS

10.4.1 Main Condenser

10.4.1.1 Design Bases

The main condenser is not a safety-related component and has no safety design bases. The following are design criteria which apply to the main condenser:

a. The main condenser is designed to function as a steam cycle heat sink, receiving and condensing exhaust steam from the main turbine and steam generator feedwater pump turbines. It has the capability to condense turbine bypass flows up to 40 percent of full-load main steam flow without exceeding turbine exhaust pressure and temperature limitations.

b. The main condenser is designed to serve as a collection point for vents and drains from various components and systems of the heat cycle.

c. The hotwell of the main condenser is designed to provide approximately three minutes of storage capacity without makeup, for the valves-wide-open flow condition, when operating at the normal water level.

d. The main condenser is designed to de-aerate the condensate.

e. Codes and standards applicable to the condenser design are:

    1. Heat Exchange Institute - Standards for Steam Surface Condensers

    2. American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code Section VIII, Division I

    3. Tubular Exchangers Manufacturer's Association Standards, Class C

    4. American Society for Testing and Materials - Standards

    5. American National Standards Institute.

10.4.1.2 System Description

a. Component Description

Each condenser shell assembly is comprised of the following sections:

Shell	Tubes
Hotwell	Tube Stakes
Steam dome	Waterboxes
Extension neck	Internal piping to external connections
Tube sheets	Tube support plates

The main condenser system consists of three de-aerating, double pass, single pressure, radial flow type condensers. Each one-third capacity condenser is located beneath one of three low-pressure turbine cylinders. Each condenser shell is floor-mounted and connected to the turbine exhaust flange by means of a rubber-belt expansion joint to accommodate differential thermal expansion between the turbine and condenser.

The condenser shell is carbon steel, welded construction, with $^{1}/_{16}$" corrosion allowance. Aluminum-bronze tube sheets are bolted to the shell and have provision for allowing thermal expansion of the tubes. The condenser tubes are titanium and are rolled into the holes of the tube sheets.

The condenser tubes are oriented transverse to the turbine-generator axis. Exhaust steam from the turbine discharges down into the condenser from exhaust openings in the bottom of each of the low pressure turbine casings. In addition, the condenser receives steam from the main feed pump turbines.

The condenser tube stakes prevent tube whippage caused by localized sonic steam velocities. The stakes stiffen the titanium tube bundle to reduce vibrational effects which may lead to tube failure. The tube stakes consist of stainless steel Shepard's Crooks for support between tubes, Micarta phenolics for support in the tube lanes, and stainless steel U-clip devices which are installed at certain tube bundle perimeter areas.

The first two stages of feedwater heaters are mounted in the neck of each condenser shell. The extraction steam lines to these heaters are completely enclosed in the condenser neck and dome, while the lines to heaters 3 and 4 pass through the interior of the condenser dome and penetrate the condenser shell.

The waterboxes are coated/lined to prevent corrosion induced by the sea water. The waterboxes are bolted to the condenser shells and are designed for easy removal without disturbing the tube sheets. Design pressure of the waterboxes is 60 psig.

A butterfly valve is provided at the circulating water inlet and outlet nozzles of each shell for isolation. Isolation valves are also provided on all condensate and heater drain lines connecting to each shell, so that each shell can be isolated without shutting down the unit. Though the isolated condenser shell is still exposed to the turbine exhaust steam, it is possible to inspect the tubes and perform minor maintenance and repairs while the unit is operating.

b. System Operation

During normal operation, exhaust steam from the low pressure turbines is directed into the shells of the main condenser. The following auxiliary flows are also discharged into the condenser:

1. Exhaust steam from steam generator feed pump turbines
2. Drains and vents from feedwater heaters
3. Condensate from gland seal system
4. Miscellaneous equipment drains
5. Steam generator blowdown from blowdown flash tank (vapor only) and from the blowdown demineralizers.

In addition to condensing the steam, the condenser also de-aerates the condensate. The noncondensible gases which accumulate in the condenser are removed by the air evacuation system (see Subsection 10.4.2).

Condenser hotwell level is monitored and necessary makeup is provided by the condensate storage tank or discharge to circulating water, respectively (see Subsection 9.2.6). Rejection of condensate normally occurs due to high hotwell level or high contaminant levels. Generally, the condensate storage tank is not used to collect rejected water. Hotwell cation conductivity is also monitored and alarmed in the main control room. The monitoring is arranged so the operator can determine which shell has the inleakage, so that the necessary steps to isolate or plug the leaking tubes can be taken.

The condenser has sufficient capacity to condense 40 percent of the full-load steam flow to the turbine during turbine steam load rejection without exceeding 5" Hg pressure.

Total hotwell capacity is 66,000 gallons at normal water level. This is in excess of three-minute capacity, based on condensate flow at 100 percent load.

There is no provision for control of the circulating water flow except by taking a pump out of service. For a description of the Circulating Water System, see Subsection 10.4.5.

c. System Design Data

Design backpressure	2.0" HgA
Backpressure during full load turbine operation, range	1.5-2.8" HgA
Backpressure during steam dump	5" HgA Max.
Total Heat Load:	
Design	7.90×10^9 Btu/hr
Steam dump	7.15×10^9 Btu/hr
Tubes:	Titanium
	55'-3"
Size	1" OD, 22 gage
Circulating water flow (total)	399,000 gpm
Temperature rise	38°F
Water velocity	7 fps
Head loss	27 ft

10.4.1.3 Safety Evaluation

The main condenser has no safety-related design basis. Due to the plant design (PWR) there is negligible influence of condenser control functions on reactor coolant system operation, and negligible potential for hydrogen buildup in the condenser due to continuous gas removal.

In the event of a steam generator tube leak, radioactivity can be present in the secondary side. See Section 11.1 for expected activity due to steam generator tube leakage.

Due to the location of the condenser in the Turbine Building, any flooding resulting from condenser failure will not affect safety-related equipment.

10.4.1.4 Inspection and Testing

The main condenser shell, tubes, and waterboxes are hydrostatically tested to verify integrity prior to initial plant startup.

For service inspection, access manholes are provided on the outlet and turn-around waterboxes, on both ends of the hotwell, and in the steam dome. It is planned to perform a visual inspection of the condenser internals during each refueling outage as part of the normal station preventive maintenance activities.

10.4.1.5 Instrumentation

Condenser vacuum is indicated and recorded in the control room. Condenser vacuum pressure switches are used to (1) alarm pre-turbine trip vacuum, (2) trip the turbine with two-out-of-three coincidence logic and (3) block steam dump to the condenser, also with two-out-of-three coincidence.

Hotwell level indication and high, low and low-low level alarms are provided in the control room. Hotwell level will control the hotwell water inventory by admitting makeup water from the condensate storage facility.

Sea water inleakage to the condenser is monitored by conductivity cells at the catch trough and cation conductivity cells at the hotwell and is recorded and alarmed in the control room and at a local panel. Inleakage is also monitored and alarmed by measuring sodium ion concentration. The instrumentation is sensitive to leakage resulting in ppb concentration.

Monitoring of radioactive contamination is described in Subsection 10.4.2.5.

Additional instrumentation monitors the performance of the condenser by measuring circulating water inlet and outlet temperature and differential pressure. Display of circulating water outlet temperature at the main control board is used for temperature monitoring during backwash operations.

10.4.2 Main Condenser Evacuation System

10.4.2.1 Design Bases

The main condenser evacuation system, Figure 10.4-1, consists of two independent subsystems: the shell side evacuation subsystem and the waterbox priming subsystem.

The Main Condenser Evacuation System has no safety design basis; however, the following design criteria are applicable:

a. The shell side evacuation subsystem removes noncondensible gases and air inleakage from the steam space of the main condenser shells. This system is sized to achieve and maintain shell side vacuum in the condenser to permit plant startup and operation.

b. The waterbox priming subsystem removes the noncondensible gases from the condenser waterboxes (tube side) during startup and normal operation.

Refer to Subsection 10.4.1.1 for the applicable codes and standards for this system.

10.4.2.2 System Description

a. System Components

The air removal equipment consists of three mechanical vacuum pumps serving the three condenser shells, and two mechanical waterbox priming pumps serving the condenser waterboxes.

The vacuum pumps for both subsystems are of the rotary design and electric motor driven, with the shell side pumps being two stage and the waterbox priming pumps being a single stage. Each pump is skid-mounted with its own moisture separator located downstream of the pump discharge and seal water cooler.

The seals to each pump are provided with a closed cooling system using demineralized water. The seal system on the shell side pumps are cooled by circulating water, and on the waterbox pump by service water.

b. Operation

1. Shell Side Evacuation Subsystem

All three pumps will be operated to evacuate the condenser during startup. Condenser pressure can be reduced to approximately 2" HgA in approximately 76 minutes with zero air inleakage. Once vacuum is attained, one pump will be placed on standby to start automatically at approximately 26" Hg vacuum.

During normal plant operation, the noncondensible gases removed by the shell side evacuation system are piped to the Primary Auxiliary Building (PAB) where they are passed through a HEPA and charcoal filter prior to their discharge to the atmosphere. This is done to minimize the possibility of a radioactive discharge to the environment in the event of a steam generator tube leak. For the hogging or startup mode, the noncondensible gases are not expected to be radioactive, and are discharged directly through the Turbine Building vent to atmosphere. See Subsection 10.4.2.5 for discussion on monitoring of shell side discharge. Also, refer to Subsection 11.3.3 for anticipated release rates of radioactive materials.

Vacuum pump discharge flow is directed either to the filter or to the plant stack by diverting valves which are manually positioned by the operator at the main control board.

2. Waterbox Priming Pumps Subsystem

The waterbox priming pumps are used to remove noncondensible gases from the condenser waterboxes during startup, and to remove gases that are released from the circulating water due to temperature rise during normal operation. These gases are discharged directly to atmosphere through the Turbine Building vent.

The two waterbox priming pumps are each sized for full-load operating conditions. During normal operation, one pump is operating with the other on standby or isolated for maintenance.

10.4.2.3 Safety Evaluation

Normally, there is no radioactivity in the condenser evacuation system. A filtering system for treatment of gaseous waste is provided in the event of steam generator tube leakage. The diverting valves in the discharge of the shell side vacuum pumps are arranged so that on loss of electric or pneumatic power, they will fail to the filter position. The radioactivity released from the condenser evacuation system is discussed in Section 11.3.

10.4.2.4 Inspection and Testing Requirements

Initial testing of the system is performed prior to startup to insure the proper functioning and performance of the system and its components.

10.4.2.5 Instrumentation

Local pressure, level and temperature indication are provided to monitor the operation of the Main Condenser Evacuation System. Vacuum switches are provided for automatic operation of the condenser (mechanical) vacuum pumps. Condenser vacuum is indicated and recorded in the main control room; low vacuum condition is also alarmed in the control room. Total condenser vacuum pump flow indication is also provided locally and on the main plant computer.

A radiation monitor is provided at the discharge header of the mechanical vacuum pumps to monitor releases of radioactivity to the environment. Local and remote radioactivity indication and high alarm are provided in the control room. Refer to Section 11.5 for a discussion on process and effluent radiological monitoring and sampling systems. During startup operation (hogging), the gases removed by the evacuation system are discharged to atmosphere via the Turbine Building vent. During normal operation (holding), the gases removed by the evacuation system are piped to the Primary Auxiliary Building ventilation system, where they are passed through HEPA and charcoal filters prior to discharge to atmosphere. Refer to Subsection 9.4.3, for a description of the Primary Auxiliary Building heating and ventilation system.

10.4.3 Turbine Gland Sealing System

10.4.3.1 Design Basis

The turbine gland sealing system provides sealing steam to the main turbine and the two steam generator feed pump turbines. The sealing system prevents the leakage of steam from the turbine packing glands into the Turbine Building and also prevents the leakage of air into the main condenser.

10.4.3.2 Description

The annular space between the turbine shaft and the turbine casing is sealed with conventional steam-sealed labyrinth packing. Each packing box is provided with a steam supply connection and an air-steam exhaust connection to the gland steam condenser, where the steam seal exhaust is condensed. At loads above minimum, the high pressure turbine packing has an excess of steam in which case the steam supply connection serves as a leakoff.

Relief valves are provided to protect the system from malfunctions resulting in overpressure conditions. Air and noncondensible gases are removed from the gland steam condenser by motor-driven exhauster fans. The cooling medium for the gland steam condenser is provided by the condensate system. The steam seal header pressure is maintained slightly above atmospheric. This system and its components are shown on Figure 10.4-2.

10.4.3.3 Safety Evaluation

The mixture of noncondensible gases discharged to the atmosphere by the gland steam condenser exhaust fans is not normally radioactive. Therefore, a component malfunction or failure in this system will not normally result in any release of radiation to the environment. However, in the event of a steam generator tube leak, it is possible for the exhaust fan discharge to be radioactively contaminated. To allow for an assessment of these potential effluents, the exhaust vent is equipped with the capability to obtain grab samples of iodines and particulates.

A full discussion of the radiological aspects of primary-to-secondary system leakage, including anticipated releases from the turbine gland sealing system and limiting conditions for operation, is included in Chapter 11.

10.4.3.4 Tests and Inspection

Before initial startup, the steam seal regulating valves are tested in accordance with the manufacturer's instruction manuals. During operation, the setpoints and steam pressures are checked periodically.

10.4.3.5 Instrumentation

a. Sealing steam pressure in the header is automatically regulated by air-operated diaphragm control valves which control both supply and spillover.

b. Motor-operated stop and bypass valves are manually operated from the local steam seal and drain control panel; open and closed indicating lights show valve position at the local panel.

c. Sealing steam header pressure is indicated locally at the seal steam control panel and in the control room; low header pressure is annunciated in the control room.

d. Low water level in the steam packing exhauster and low vacuum are annunciated in the control room. Both exhauster fans can be manually operated from the control room. Normally one fan is kept running with the second fan standby in auto mode, so that on loss of first fan the second one comes on line automatically. Fan motor high temperature and fan trip are annunciated in the control room.

10.4.4 Steam Dump System

10.4.4.1 Design Bases

The Steam Dump System is designed to reduce the magnitude of nuclear system transients following large turbine load reductions or turbine trips by dumping steam directly to the main condensers, thereby creating an artificial load on the reactor.

The Steam Dump System has the following functional requirements:

a. Permit direct bypass flow to the condensers of nominally 40 percent of rated turbine flow. The reactor is capable of 10 percent step-load reduction, thereby allowing a turbine step-load reduction of 50 percent without a resultant reactor trip.

b. Permit turbine and reactor trip from full power without atmospheric discharge through the steam generator safety valves.

c. Maintain steam header pressure at required pressure during startup, hot standby, cooldown and reactor physics testing periods.

The maximum capacity of any single valve does not exceed 573,000 lb/hr at 1107 psia to limit steam release if any single valve inadvertently sticks open.

If the condenser is not available, the steam dump is isolated to protect the condenser.

The steam dump system piping is designed in accordance with the ANSI B31.1 Power Piping Code.

10.4.4.2 System Description

a. General

The Steam Dump System is shown on Figure 10.3-7. Steam is discharged to the condensers through twelve air-operated, partial capacity, automatically controlled steam dump valves, (PV-3009 through 3020) which are installed in the steam dump lines connected with the steam system. The branch headers to the steam dump valves tie into the main steam headers between the main steam generator isolation valves and the turbine stop and control valves. The steam dump valves are arranged in parallel so that when combined, they will permit the desired bypass flow to pass. This arrangement will limit the steam bypassed to the condenser, should a valve open accidentally or stick open, thereby minimizing the potential for an uncontrolled cooldown of the primary system. This arrangement also permits the steam dump flow to be evenly shared by the turbine condensers, thus preventing uneven turbine exhaust backpressures.

During normal operation, the system is operated in the T_{avg} mode. A control signal obtained from the difference between primary T_{avg} and a $T_{reference}$ signal which is derived from turbine first stage pressure, is used to determine the number of valves and, thus, the amount of steam to be dumped during a transient.

For a large load reduction, either one-half or all of the dump valves are tripped open immediately, depending upon the magnitude of the transient and resulting control signal. The valves are modulated closed as reactor power approaches turbine power (i.e., T_{avg} approaches T_{ref}) so that the valves are fully closed when the reactor power matches the turbine power. For a plant trip the valves will operate to maintain T_{avg} at the no-load value.

During primary plant cooldown, the Steam Dump System is operated in the steam generator pressure control mode. In this mode, the control signal is generated by comparison of steam header pressure with the pressure setpoint. The pressure setpoint is manually reduced to achieve the required cooldown rate.

All dump valves fail closed on loss of control signals, and are prevented from operating on loss of condenser vacuum. During a loss of condenser vacuum transient, excess steam pressure is relieved to the atmosphere through the power-operated relief valves or the safety valves (see Subsection 10.3.2).

The dump lines are normally stagnant and, therefore, collect condensate continuously. This condensate is automatically removed by an upstream drain system to permit proper valve operation.

b. Steam Dump Valves

Twelve steam dump valves are provided, with each valve having a capacity of approximately 510,000 lb/hr. They are designed to fail closed on loss of control signal, and are split into four groups to ensure even distribution of heat into each condenser. The dump valves are capable of a rapid trip actuation or of a modulating operation.

The steam dump valve actuation characteristics are:

1. The valves are capable of going from full-closed to full-open within five seconds after receiving a trip/open signal. This includes the time required to actuate the solenoid valves associated with each dump valve.

2. The valves are capable of going from full-open to full-closed in five seconds after de-energization of the solenoid valves.

3. The valves are capable of being modulated with a maximum full stroke time of 25 seconds.

10.4.4.3 Safety Evaluation

The Steam Dump System is not essential to safe operation of the plant. It is provided, however, to give the plant flexibility of operation. Each valve is provided with an isolation valve to permit maintenance. The flow capacity of each valve is selected to prevent excessive cooldown rate should the valve fail in the open position.

When all the valves are out of service, the steam generator safety valves provide the relieving capacity required to maintain the steam system within the design limits.

No effects of pipe breaks are considered, since all piping is located in the Turbine Building where the effect of pipe breaks will not jeopardize the safe shutdown of the plant.

10.4.4.4 Tests and Inspections

During preoperational and initial startup testing, the Steam Dump System will be tested to verify proper valve performance and overall system dynamic response as described in Chapter 14.

10.4.4.5 Instrumentation Requirements

The Steam Dump System is controlled by a system which compares turbine power to reactor power by means of temperature and pressure inputs. The specific mode of operation (T_{avg} or steam pressure) can be selected through a selector switch mounted at the main control board (MCB). Valve position indications are also available at the MCB. The Steam Dump Control System is discussed in Subsection 7.7.1.8, and is analyzed for the following control modes:

a. Load rejection

b. Plant trip

c. Steam header pressure.

Interlocks are provided to block steam dump operations on low-low T_{avg} to prevent excessive cooldown of the primary plant and to protect secondary plant equipment if the condenser is unavailable, as sensed by the condenser pressure switches and the circulating water pump breaker positions. Figure 7.2-10 shows the functional details and the interlocks pertaining to the Steam Dump Control System.

10.4.5 Circulating Water System

The Circulating Water System provides cooling water to the main condensers to remove the heat rejected by the turbine cycle and auxiliary systems. Discussions pertaining to the interface between the Circulating Water System, the Service Water System and the ultimate heat sink are found in Subsections 9.2.1 and 9.2.5.

10.4.5.1 Design Bases

a. The circulating water system design is based on a maximum ocean water temperature of 65°F, a condenser heat load of 0.79×10^{10} Btu/hr during normal full-load operating conditions, and an average discharge water temperature maximum increase of 39°F for normal operation with both units. During the summer months, extended hot weather combined with ocean current changes can result in minor ocean temperature excursions above the 65°F design temperature threshold. System analysis has been performed to permit continued plant operation up to a maximum ocean temperature of 68.5°F.

b. The design of the system also includes the capability for furnishing cooling water to the Service Water System, and returning it to the circulating water discharge flow.

c. The Circulating Water System is designed to operate safely at extreme high tide and minimum predicted tide (see Subsection 2.4.11.2), and to permit operation of the turbine generator during condenser steam dump conditions without occurrence of a condenser low vacuum trip.

d. Provisions for continuous low-level chlorination (as shown on Figure 10.4-5) and heat treatment of the tunnels are included for control of fouling by marine organisms.

e. The design of the circulating water system structures is nonseismic Category I, with its components also nonseismic Category I and nonsafety-related.

10.4.5.2 System Description

The general arrangements of the various structures and components comprising the Circulating Water System are shown in Figure 1.2-46, Figure 1.2-47, Figure 1.2-48 and Figure 1.2-52, Figure 1.2-53, Figure 1.2-54, and Figure 1.2-55. The Circulating Water System consists of the following principal structures:

a. Two tunnels connecting the plant site with three submerged offshore intakes and a multiport discharge diffuser

b. An intake transition structure

c. A pumphouse

d. A pair of flumes which join the intake transition structure to the pumphouse

e. A discharge transition structure

f. An underground piping system, interconnecting the pumps in the pumphouse, the condensers, and the transition structures.

The flow diagram of the Circulating Water System is shown in Figure 10.4-3 and Figure 10.4-4. During normal operations, the Circulating Water System provides a continuous flow of approximately 390,000 gpm to the main condenser and 21,000 gpm for the Service Water System.

Starting 260 feet below the plant level (240 feet below mean sea level), at the bottom of vertical 19'-0" finished diameter land shafts, two tunnels extend out under the ocean at an ascending grade of about 0.5 percent until they reach their respective offshore terminus locations about 160 feet below the ocean's surface. The tunnels, which are machine bored through bedrock to a 22'-0" diameter, are concrete-lined to provide the finished 19 foot diameter.

The intake tunnel is approximately 17,000 feet long, and is connected to the ocean by means of three 9'-10½" finished diameter concrete-lined shafts, spaced between 103 and 110 feet apart and located approximately 7000 feet off the shoreline in 60 feet of water. A submerged 30'-6" diameter concrete intake structure intake head is mounted on the top of each shaft to minimize fish entrapment by reducing the intake velocity.

The discharge tunnel is approximately 16,500 feet long, and is connected to the ocean by means of eleven, 5'-1" finished inside diameter concrete-lined shafts, spaced about 100 feet apart, located approximately 5000 feet off the Seabrook Beach shoreline in water up to 70 feet deep. A double-nozzle fixture is attached to the top of each shaft to increase the discharge velocity and diffuse the heated water.

The circulating water portion of the pumphouse encloses six 14' wide circulating water traveling screens and three circulating water pumps. A seismic Category I reinforced concrete wall separates the circulating water portion from the service water portion of the pumphouse structure. The water is pumped through two 11-foot diameter pipes (1 per unit) leading to the condensers, and is returned through two 10-foot diameter discharge pipes (1 per unit) connected with the tunnel transition structures. Water to the service water section of the pumphouse is supplied by two pipelines branching off each of the tunnel transition structures.

Fouling by growth of marine organisms is expected to occur from the point where the sea water enters the intake structures up into the condenser. One process used for control of fouling in the intake structures and inlet tunnel will be continuous low-level chlorination with a sodium hypochlorite solution. Three sodium hypochlorite solution storage tanks, surrounded by a concrete spill containment dike, provide a bulk volume supply which may be pumped on demand to selected chlorination injection points within the CW system. The metered rate of injection will be such that a concentration of 0.2 ppm total residual oxidant, measured as Cl_2 equivalent, is not exceeded in the discharge transition structure. In addition, heat treatment, where the direction of flow in the tunnels is temporarily reversed, and the discharge temperature raised by recirculation is also available as a means of controlling marine growth. In this mode, the warm water from the condenser is returned to the ocean through the intake tunnel, while the discharge tunnel is used to supply ocean water to the plant. To heat treat the discharge pipes and tunnel, the temperature of the condenser outlet water is temporarily raised by recirculating some of the discharge water back to the condensers through the pumphouse.

The pumphouse, pipes leading to the condensers, and the condensers can be dewatered, inspected, and cleaned as required to control fouling.

10.4.5.3 Safety Evaluation

Since the Circulating Water System is considered nonsafety-related, the safety evaluation, therefore, concerns itself with the effect of a failure of this system or any of its components on safety-related systems or components.

If the circulating water flow rate falls below the minimum required amount due to a malfunction in the system, the main condenser may no longer be able to adequately condense main steam, but there will be no effect on the safe shutdown capability of the plant.

The safety evaluation of the Circulating Water System, as it relates to the ultimate heat sink and the Service Water System, is presented in Subsection 9.2.1.

Passage of secondary system condensate from the main condenser into the Circulating Water System through a condenser tube leak is not considered possible during power generation, since the Circulating Water System operates at a higher pressure than the condenser, thus preventing possible contamination of the circulating water by potentially radioactive condensate.

The condenser is equipped with vacuum breaker valves which open if a circulating water pump trips, thus preventing the possibility of overpressure from water hammer.

Clogging of a circulating water traveling screen will trip its corresponding circulating water pump, thus preventing collapse of the screen. None of the controls or instrumentation relates to operations which could affect safety-related systems.

The expansion joints used in the CW system are made of fabric reinforced rubber with circumferential steel reinforcing rings. This makes the rupture of a joint in operation a very unlikely occurrence. To evaluate the consequences of a rubber joint failure, they can be divided into four groups as follows:

Group	Number of Joints	Size ID (in)	Pressure Normal Oper. (ft)	Max (ft)	Location
A	4	102	20	55	Between intake transition structure and pumphouse.
	2	120	20	55	Between discharge transition structure and backwash conduits to the pumphouse.
B	4	120	15	50	At both ends of the discharge pipe section between the two transition structures.

Group	Number of Joints	Size ID (in)	Pressure Normal Oper. (ft)	Max (ft)	Location
C	6	84	50	130	CW pumps discharge.
D	12	84	50	130	Condenser connections.

Group A

Failure of a joint in this group will result in a flood in the valve pit around the transition structure in which the joint is installed, but the water will reach no higher than the water in the pumphouse. The worm gear boxes directly mounted on the butterfly valves will be submerged; however, the electric motors and controls, which are installed above grade, will always be above water. Since the valves remain operational, and no other equipment is affected by this failure, no immediate action will be necessary.

Group B

Failure of a joint in this group will also cause flooding of the valve pit around the transition structure in which the joint is installed, and the water leaking from the joint will drain into the valve pit, eventually fill it, and overflow in the ground unless the CW system of that unit is stopped. It is estimated that the time required to fill the pit will not be less than 45 minutes, even assuming the worst possible failure to be a 2"x24" gap at the lowest side of the joint. A water level alarm will warn the operator of the flooded condition in the valve pit. Failure of a joint in this group will not prevent the Service Water System from providing its function.

Group C

Failure of a joint in this group will flood the pump pit in the Circulating Water Pumphouse between elevations +3' and +20' MSL. A number of openings at elevation 21'-0" of the Circulating Water Pumphouse will prevent the water from building up above the operating floor. The CW pump electric motors and the pump discharge butterfly valve electric motors will always be above water level, and will not be affected by such a failure. Assuming the worst possible failure to be a 2" gap all around, the CW pump pit would fill up in not less than 6.5 minutes. No safety-related equipment is affected by a failure of this equipment.

Group D

Failure of a joint in this group will flood the ground floor pit east of the condensers in the Turbine Building. Assuming the worst possible failure to be a 2" gap all around, the pit would fill up in about 3 minutes, unless prompt action by the operator is taken. There are two level switches in the condenser pit that provide sequential alarms in the control room to warn the operator of the flooded condition. No loss of offsite power is induced by a failure of this equipment provided operator action is taken within 22.2 minutes to mitigate the consequences of the flood.

Summary

The service water pumps are located at ground elevations and their motors are situated above ground elevation. There are no openings in the common wall between the service water pumps and CW pumps. Therefore, flooding which may spread to the open ground has little potential for entering the Service Pumphouse and no potential for interrupting the service water pump safety function.

Flooding caused by a flexible joint failure at the main condenser may spread over the turbine floor and from there to the open ground by passing through Turbine Building doors. However, sufficient physical separation is provided between the areas subject to such a flood and any safety-related equipment whose safety function could be impaired by such a flood.

10.4.5.4 Testing and Inspection

Sufficient manholes and access ways are provided in the circulating water pipes and condenser waterboxes for any necessary inspection or cleaning activities.

10.4.5.5 Instrumentation

Temperature instruments used for condenser performance monitoring and as an upper limit to water temperature for system heat treatment are described in Subsection 10.4.1.5.

Control of heat treatment operation is from the Unit 1 control room. The operator can align all valves (including service water intake valves) for tunnel heat treatment in concert, and adjust the temperature by observing additional discharge water temperature readouts from the computer.

Control and display instrumentation is provided to permit operation of the Circulating Water System from the main control room under all normal and abnormal conditions.

Level instrumentation monitors water level of the intake structure, discharge structure and Circulating Water Pumphouse. The level is indicated at the main control board.

Circulating water pump motor temperatures are monitored and high temperature alarms are provided at the main control board. Once the circulating water pump starts it will be automatically tripped on a loss of offsite power or high pressure drop across the traveling screen.

Controls and position indications are provided on the main control board for the pump operation, motor-operated valves, and condenser vacuum breaker valves.

10.4.6 Condensate Polishing System

10.4.6.1 Design Basis

The Condensate Polishing System (CPS) is an integral part of the Condensate (CO) system. The CPS is intended for standby service and is designed to remove dissolved and suspended impurities from the secondary system. The condensate polishers may be aligned to remove impurities, which could contaminate the secondary system due to a condenser (circulating water) tube leak. The system may also be used during startup to clean the condensate and feedwater systems prior to initiating secondary system flow to the steam generators.

10.4.6.2 System Description

10.4.6.2.1 Operation

The CPS utilizes demineralizers sized to process the equivalent of 100% flow from one of the three condenser hotwells. The CPS is an extension of the condensate system where fluid from a condenser hotwell may be treated to remove contaminants. The CPS removes impurities resulting from condenser tube leakage and corrosion products from the feedwater and condensate systems. The CPS produces a high quality effluent capable of meeting feedwater and steam-side chemistry specifications (see Subsection 10.3.5).

Three lead cation bed (two operating; one spare) demineralizers are utilized to remove system amines prior to processing by the four mixed bed (three operating; one spare) demineralizers. Resin beads are retained in the vessels by a flat, slotted screen underdrain system. Resin bead strainers are located downsteam of each vessel to retain resin in the unlikely event of an underdrain failure.

The CPS is normally in standby. When required for service to mitigate the effects of a condenser leak, the CPS takes the entire flow from the leaking condenser's hotwell, passes this condensate through the demineralizers and returns the condensate to the two clean condenser hotwells. The two clean condenser hotwells remain aligned to the condensate pumps while the leaking condenser hotwell is isolated from the condensate system and aligned to the CPS pump.

The CPS is sized to meet the secondary chemistry requirements for continuous operation as discussed in Subsection 10.3.5 while operating with a condenser leak of 0.04 gpm and to maintain water quality for an orderly unit shutdown with a condenser leak of 0.4 gpm.

The system provides for full flow loop recirculation prior to connection of the CPS to the condensate system. Polisher vessel isolation valves and system interconnection valves are designed to permit a controlled opening and closing to minimize hydraulic surges on the resin bed and within the system piping.

An external resin regeneration and waste processing system is utilized in conjunction with the CPS. The number and sizing of the polisher vessels are such that the functional requirements can be met while permitting the regeneration of resin in one ion exchanger at a time. Sampling of system parameters is provided to ensure correct operation.

Condensate polishing regenerant waste is collected in the low conductivity tank and the water treatment system neutralization tank and sampled prior to release, in accordance with the NPDES permit requirements and the ODCM.

Spent resin will be sluiced to drums for dewatering. The final shipment container and disposal will depend on sampling and whether the resin is determined to require disposal as radioactive waste. The disposal of radioactive resin will meet the requirements as described in Section 11.4. If the waste resin is not determined to be radioactive, then it may be processed as commercial waste in accordance with applicable requirements.

Representative CPS design and operating parameters are listed in Table 10.4-4. The Condensate Polishing System is shown on Figure 10.4-18 and Figure 10.4-19.

10.4.6.2.2 Component Description

a. Cation Demineralizers

The lead cation beds utilize strong acid cation resins to remove dissolved solids from the main condensate stream. The cation beds also remove ammonia, formed in the secondary system as a result of hydrazine degradation, which assists in pH control. The removal of ammonia and other amines allows an extended run length for the mixed beds, which are in series downstream of the cation beds.

Design pressure and temperature for the demineralizer assemblies are 150 psig and 150°F. The expected temperature for ion exchange resins within the demineralizer is the same as condensate pump discharge temperature, which is less than 125°F. Design temperatures and pressures are functions of the condensate pump head, condensate system heat balance criteria, and inherent stability of the resin used. Normal operating conditions are 100 psig and 90 - 105°F.

The sizing of each cation demineralizer is based on adherence to a maximum flow rate of vessel cross section. The final selection of two operating units and one spare unit, each 9 feet-6 inches in outside diameter, provides full CPS flow capability at a unit design flow rate of less than 52 gpm/ft^2.

The cation demineralizer vessels are lined and stainless steel fitted for protection against localized corrosion where ion exchange resins contact internal wetted surfaces. Internal distributors/collectors are stainless steel. The cation demineralizer vessels are designed, fabricated, and code stamped in accordance with ASME Section VIII.

b. Mixed Bed Demineralizers

The mixed beds remove anions and augment the cation beds through further removal of cations.

The design pressure and temperature for the mixed beds are the same as those presented for the cation beds, i.e. 150 psig and 150°F. The expected temperature for ion exchange resins within each demineralizer is less than 125°F. Design temperature and pressures are functions of the condensate pump head, condensate system heat balance criteria, and inherent stability of the resin used. Normal operating conditions are 100 psig and 90 - 105°F.

The sizing of each demineralizer is based on adherence to a maximum flow rate of vessel cross section. The final selection of three operating units and one spare unit, each 8 feet in outside diameter, provides full CPS flow capability at a unit design flow rate of less than 50 gpm/ft^2.

The mixed bed demineralizer vessels are lined and stainless steel fitted for protection against localized corrosion where ion exchange resins contact internal wetted surfaces. Internal distributors/collectors are stainless steel. The mixed bed demineralizer vessels are designed, fabricated, and code stamped in accordance with ASME Section VIII.

c. Condensate Polishing Pump

The condensate polishing pump returns effluent condensate flow from the CPS demineralizers to the CO system. The pump developed head compensates for the losses through the CO system piping, CPS piping, demineralizers, and the CPS control valve.

One 100% condensate polishing pump is provided. The pump is rated for 7,500 gpm at 230 ft. TDH and is driven by a 600 hp motor.

Pump pressure parts are carbon steel; pump shafting and impellers are stainless steel.

d. Condensate Polishing Head Tank

A condensate polishing head tank provides a positive head on the CPS during various operating transients. The tank has a capacity of 7,000 gallons with an approximate normal operating volume of 5,500 gallons.

Design pressure and temperature of the head tank are 150 psig/full vacuum and 150°F. The vertical carbon steel tank has been designed and fabricated in accordance with ASME Section VIII and is code stamped.

e. Separation Tank

The separation tank is used to separate the anion and cation resins hydraulically. The vessel dimensions are 54-inch diameter in the narrow section, expanding to a cone of twice the cross sectional area or 78 inches in diameter. The narrow section is 17 feet long, and the cone is 6 feet.

Design pressure and temperature of the separation tank are 150 psig and 150°F. The vertical lined, carbon steel tank has been designed and fabricated in accordance with ASME Section VIII, and is code stamped. For ease of installation, the tank was fabricated in two pieces that are joined by an ANSI 150lb flange midway in the narrow section.

Table 10.4-4 lists details on the design requirements for all the major CPS components.

10.4.6.3 Safety Evaluation

The CPS has no safety-related design basis. All CPS components are contained in non safety related structures. Failure of any portion of this system will not damage any safety-related component or system.

In the event of a steam generator tube leak, radioactivity can be present in the secondary side and could be conveyed to the CPS. Radioactive contamination of the secondary side by steam generator tube leakage is independent of the CPS and is addressed in Section 11.1.

Due to the location of the CPS in the Turbine Building, Administration Building, and Condensate Polishing Facility, any flooding initiated by the CPS or its components will not affect safety-related equipment.

10.4.6.4 Testing and Inspection

Tests and inspections on the CPS are performed to applicable codes and standards. The testing includes functional testing of the system and its components to ensure structural integrity, leaktightness, and the operability/performance of the integrated systems to function and perform during expected operating conditions. Normal operating system performance monitoring detects deterioration in the performance of the system components.

10.4.6.5 Instrumentation

The instrumentation employed for monitoring the CPS performance includes the following:

a. The condensate polishing pump suction and discharge pressure is locally indicated, and discharge header pressure is monitored to ensure correct system operation. The pump is protected from damage by a low flow permissive, a low condenser hotwell level trip and undervoltage, overcurrent and differential protection.

b. The CPS flow control valve regulates system flow downstream of the demineralizers in either auto or manual mode. Flow is modulated based on condensate polishing pump discharge pressure, condenser hotwell level and condensate system flow.

c. The conductivity of the influent condensate to the cation demineralizers, the effluent from each demineralizer, the common cation bed effluent, and the common mixed bed effluent is measured and recorded. An alarm signal is provided to indicate high conductivity.

d. Sodium levels of the influent condensate to the cation demineralizers, the effluent from each demineralizer, the common cation bed effluent, and the common mixed bed effluent can be measured and recorded. An alarm signal is provided to indicate high levels.

e. Silica levels of the effluent from each mixed bed demineralizer, and the common mixed bed effluent can be measured and recorded. An alarm signal is provided to indicate high levels.

f. Dissolved oxygen levels of the common mixed bed effluent can be measured and recorded. An alarm signal is provided to indicate high levels.

g. Flow transmitters are provided to monitor the throughput of each demineralizer.

h. Measurement of the waste stream for any radioactivity both before and during disposal is provided. Presence of a radionuclide concentration exceeding guidelines automatically terminates disposal to the circulating water outfall line.

10.4.7 Condensate and Feedwater Systems

The Condensate and Feedwater Systems return the condensate from the turbine condenser hotwells through the regenerative feed heating cycle to the steam generators while maintaining the water inventories throughout the cycle.

10.4.7.1 Design Bases

a. The Condensate and Feedwater Systems are designed to provide approximately 16.43×10^6 lb/hr of feedwater at 446.7°F to the steam generators at 100% load. The feedwater system capability is in excess of the turbine control valves wide open (VWO) conditions.

b. The condensate portion of the system is designed to supply approximately 11.22×10^6 lb/hr, with an additional supply of 5.21×10^6 lb/hr from the heater drain pumps to the suction side of the steam generator feedwater pumps at 100% load. The condensate system capability is in excess of the turbine control valves wide open conditions.

c. The feedwater portion of the system is designed to supply the feedwater required for steady-state operation, and to maintain this flow, as required, following a large load reduction. The system is designed to maintain uniform feedwater flow to all steam generators under all conditions and to maintain proper steam generator water levels automatically during steady-state and transient conditions.

d. The Feedwater System from the steam generators, back to and including the check valve upstream of the feedwater isolation valve, is designated as Safety Class 2 and is designed to the requirements of seismic Category I systems. The portion of the system from upstream of the feedwater penetration in the pipe chase, and the Condensate System, is nonseismic Category I, and is designed to ANSI B31.1 requirements. The condensate storage tank is designated Safety Class 3, seismic Category I.

e. The condensate storage tank is designed to store and supply makeup water for the condensate, feedwater, and the Emergency Feedwater Systems (see Subsection 9.2.6).

f. The design parameters for the major components of the Condensate and Feedwater Systems are given in Table 10.4-1.

g. The safety class feedwater lines between the penetrations at the east and west pipe chases and the steam generator nozzles are located in a seismic Category I structure that is tornado, missile and flood protected.

h. The condensate and feedwater system components are designed and constructed in accordance with the following applicable regulations, codes and standards:

1. Code of Federal Regulations, 10 CFR Part 50

2. Branch Technical Positions APCSB 3-1 and MED 3-1

3. ASME Boiler and Pressure Vessel Code

   Section III Nuclear Power Plant Components (Class 2)

   Section XI In-Service Inspection of Nuclear Power Plant Components

   Section VIII Pressure Vessels (Division 1)

4. NRC Regulatory Guides

   1.26 Quality Group Classification and Standards

   1.29 Seismic Design Classification

5. American National Standards Institute (ANSI)

   B31.1 Power Piping Code

   N18.2 Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants

10.4.7.2 Systems Description

The condensate and feedwater system flow diagrams are shown on Figure 10.4-6, Figure 10.4-7, Figure 10.4-8 and Figure 10.4-9.

a. Condensate System

   Three motor-driven, constant-speed, vertical canned-type condensate pumps are supplied, each designed for approximately 50 percent of the total condensate flow.

   These pumps withdraw condensate from the three condenser hotwells via a common header arrangement which cross-connects the three condenser shells. During normal operation, only two pumps will be operating and one will be on standby. The pumps are vented to the condenser shell to prevent air binding. Seal and priming water are supplied to the condensate pumps from the condensate storage tank or the Demineralized Water System. The condensate pumps discharge into a common header that carries the flow to the steam packing exhauster, which condenses the turbine sealing steam and exhausts noncondensibles through blowers to the atmosphere.

   The total condensate flow rate in the common header is measured after leaving the steam packing exhauster. This measurement is used to ensure that the flow rate does not fall below the minimum flow requirements of the steam packing exhauster and the condensate pumps. This is accomplished by a recirculation valve downstream of the steam packing exhauster which will open to maintain the required flow. Recirculation valves are also provided at each pump to protect the pumps if their discharge valve is closed.

   During plant start-up excess condensate from auxiliary steam used for turbine gland sealing and shell warming is returned to the Auxiliary Steam Condensate System.

The common condensate header flow is routed to three parallel groups of low pressure heaters Nos. 1 and 2 mounted horizontally in the three condenser necks. The outlet flow is manifolded into a common header and conveyed to low pressure heaters Nos. 3 and 4, which are also three parallel groups of heaters, located in the turbine heater bay. The outlet flow is manifolded and routed to two parallel low pressure heaters No. 5, also located in the turbine heater bay.

Low pressure heater No. 1 is a four-pass, and heaters 2, 3 and 4 are two-pass U-tube type with integral drain coolers, while heaters No. 5 are two-pass, U-tube without drain cooler. All feedwater heater strings are provided with block valves and bypass piping to take units out of service for maintenance. All heaters are provided with tube-side safety valves to provide protection against possible overpressurization caused by heating of water trapped between closed isolation valves.

Extraction steam from the main turbine and cross-under piping is the heat source for the heaters (see Figure 10.3-10 and Figure 10.3-11). Automatic nonreturn valves are provided in the extraction steam lines for heaters that are not mounted in the condenser neck. The drains from low pressure heaters Nos. 1, 2, 3 and 4 are cascaded and eventually discharged to the condensers. The drains from low pressure heaters No. 5, which also receive the cascaded drains from high pressure heaters No. 6, are piped to the heater drain tank.

The condensate outlets from low pressure heaters No. 5 are manifolded in a common header, along with the discharge of the two vertical heater drain pumps, which account for about 30 percent of the total feedwater flow at full power. The heater drain pumps take suction from the heater drain tank. A branch line off the common condensate header, before the steam packing exhauster, connects to the individual heater drain pump suction lines to protect the heater drain pumps against low NPSH during transient conditions.

The common condensate header distributes the flow equally to the suction side of the two steam generator feed pumps, after passing through a flow measuring device located in each pump suction line. These are used to control the respective feedwater pump recirculation valves.

Condenser hotwell makeup is provided from either the Condensate Storage Tank or the Demineralized Water Storage Tanks upon receipt of a hotwell low level signal.

Demineralized water from the water treatment system can be introduced into the three condenser hotwells via the condensate storage tank or directly from the Demineralized Water System. The condensate storage tank is protected from freezing by a recirculation system which utilizes a heat exchanger and pump controlled by tank temperature. All condensate system connections to the condensate storage tank which are required for normal system operation are located above the tank level required for emergency plant shutdown (see Subsection 9.2.6).

The Condensate Polishing System (see Subsection 10.4.6) may be used during start-up to clean the condensate and feedwater systems prior to initiating secondary system flow to the steam generators. The system may remain on line during ramp-up to 100% power.

b. Feedwater System

The Feedwater System receives water from the Condensate System and a portion of the Heater Drain System, (specifically, drains from high pressure heaters No. 6, low pressure heaters No. 5, MSR shell drains and MSR reheater drains). The feedwater is pumped through the final stage of feedwater heaters (high pressure heaters No. 6) to the four steam generators.

The flow from the two 50 percent, variable speed, horizontal, turbine-driven steam generator feedwater pumps combines into a common header that feeds two parallel high pressure heaters No. 6. The outlets from the high pressure heaters No. 6 are combined into a common header for temperature equalization. From this common header, an individual feedwater line supplies each steam generator. The flow through each individual feedwater line is controlled automatically using two feedwater regulator valves, one designed for low power operation and one designed for high power operation. The regulator valves are pneumatically operated and are designed to fail closed on loss of air. The feedwater regulator valves are located at the south end of the Turbine Building.

The four feedwater lines exit the Turbine Building; two routed east of the Containment and two routed west, where they enter the east and west pipe chases. The east and west pipe chases house the feedwater isolation valves, which are located just upstream of the containment penetrations and connections to the steam generators. Immediately upstream of the feedwater isolation valve is a check valve and a flow measuring device. The emergency feedwater pump discharge connection to each main feedwater line is located between the containment penetration and the feedwater isolation valve.

An ultrasonic feedwater flow measurement system is installed in the common feedwater header just upstream of the feedwater regulating valves. This system is comprised of a 36-inch in line flow measurement spoolpiece and a local system processor panel. The ultrasonic flow measurement system provides high accuracy mass flow, feedwater temperature and feedwater pressure signals to the Main Plant Computer System via a digital communication link. These signals are utilized as inputs to the secondary power calorimetric calculation performed by the Main Plant Computer System.

Each steam generator feedwater pump has a recirculation control system which protects the pumps from damage at low loads by ensuring minimum flow. A feed pump gland seal water system regulates the flow of condensate from the condensate pump discharge header to the feed pump seals. Leak-off from the seals to the seal water receiver tank is returned to the condenser using a tank level controller which operates a control valve in the outlet line from the tank to the condenser.

Individual steam turbines drive the steam generator feed pumps. The turbine drives are of the dual admission type, and each is equipped with two sets of stop and control valves. One set regulates high pressure steam from the Main Steam System, and the other set regulates low pressure steam extracted from the crossover piping. Gland steam is provided to the turbines from the main turbine gland steam supply system. The exhaust steam from the steam generator feedwater pump turbine drives is condensed in main condenser shells A and C.

One steam generator startup feed pump is provided for each unit to provide normal requirements for startup, cooldown and no-load operation. The pump takes suction from the condensate storage tank and discharges through a startup heater into the high pressure feed water heater discharge piping. (The pump suction may also be aligned to the Demineralized Water Storage Tanks as a backup water source. Startup feedwater flow may also be directed through both high pressure feed water heaters in series. The Startup Feedwater System is described in Subsection 10.4.12. The condensate pumps can also be used for startup by using the steam generator feedwater pump bypass piping. A Sampling System is provided and connected to various points in the Condensate, Feedwater and Heater Drains Systems (see Subsection 9.3.2).

Condensate and feedwater chemistry is controlled as described in Subsection 10.3.5.

The chemical feed for the condensate and steam generator wet layup systems is stored in covered tanks for personnel protection.

Leakage in Condensate and Feedwater Systems through valve packings, pump seals, etc., will be collected by the floor drainage system. Makeup required because of leakage will be automatically controlled from the condensate storage tank or the Demineralized Water System.

10.4.7.3 Safety Evaluation

The requirements of 10 CFR Part 50 for containment isolation are satisfied by one feedwater isolation valve in each main feedwater line, located outside the Containment (see Subsection 6.2.4). These valves isolate the steam generators in the event of a steam generator tube rupture or feedwater line break, and prevent the continued input of feedwater to the Containment and resultant continued pressure increase in the event of a steam line rupture upstream of the main steam isolation valves. For analysis of feedwater system malfunctions that result in a decrease in feedwater temperature, increase in feedwater flow or loss of normal feedwater flow, see Subsections 15.1.1, 15.1.2 and 15.2.7.

With loss of main feedwater flow in the normal direction, the emergency feed pumps will supply sufficient flow to satisfy the primary system's cooling requirements (see Section 6.8). The connection for emergency feedwater on the main feedwater lines is downstream of both the feedwater isolation valve and the check valve which will be held closed by backpressure, to prevent flow in the reverse direction.

The circumstances associated with severe feedwater line water hammer and abnormal pipe movement in some PWR plants show consistently that the steam generator feed ring (where feedwater exits the bottom of the feed ring) was uncovered and drained prior to recovering with cold feedwater.

Seabrook station has a steam generator design (Westinghouse Model F, see Subsection 5.4.2) that enables the feed ring to be uncovered without subsequent drainage because feedwater exits the feed ring from the top through inverted J-tubes. This arrangement insures a flooded feed ring for all level transients within the steam generator, thereby eliminating water hammer. Feedwater piping from the feedwater isolation valve to the steam generator is routed to be self-venting, with the steam generator being the high point of the system.

Any failure in the nonsafety class portion of the Condensate and Feedwater Systems has no effect on the safety of the reactor, which can be shutdown in an orderly manner. A source of feedwater supply to the steam generators is required for decay heat removal from the reactor following a unit shutdown. In the event that the Condensate and Feedwater Systems are not available, the Emergency Feedwater System (see Section 6.8) provides the required emergency supply of feedwater. Normal reactor cooldown is accomplished by dumping steam to the main condenser. When the reactor coolant system temperature and pressure are reduced to or below the design values for the Residual Heat Removal (RHR) System, steam dump can be secured. The RHR design conditions (600 psig and 400°F) correspond to a secondary system pressure of 233 psig.

If for any reason the normal cooldown mode cannot be utilized, the reactor can be cooled down using the Emergency Feedwater System (see Section 6.8) and the power-operated steam generator relief valves.

10.4.7.4 Inspection and Testing

During preoperational testing, the various components of the feedwater, condensate, and associated portion of the heater drain system are functionally tested to verify their performance to the extent practical. The systems are operated during hot functional testing at normal no-load conditions as a final check prior to plant operation. The specific testing is described in Chapter 14.

In-service inspection of the Class 2 feedwater piping will be performed in accordance with ASME B&PV Code, Section XI.

10.4.7.5 Instrumentation

The instrumentation employed for monitoring the Condensate and Feedwater System performance consists of the following:

a. Each condensate pump suction and discharge pressure is locally indicated, and discharge header pressure is indicated at the MCB. Pump controls are at MCB. Normally, two pumps are running with one in standby. The standby pump runs automatically on low discharge header pressure or on trip of any one of the running pumps. Each pump is protected from overheating by being interlocked to pump cooling water flow. Each pump is protected from damage by individual recirculation flow on low pressure. The steam packing exhauster recirculation valve opens on low flow in the condensate discharge header to provide minimum flow protection for the condensate pumps and closes on low header pressure to prevent pump run out and maintain feed pump NPSH.

Pump trip, bearing and motor winding high temperature, and low cooling water flow are alarmed in the control room.

b. The condenser hotwells are interconnected, and each hotwell is equipped with a level transmitter. The level signals from the three hotwells are processed by an auctioneering circuit which selects the lowest value for the hotwell level control system. The level is maintained within a preselected control band by admitting makeup water via the makeup control valve from the condensate storage tank or the Demineralized Water System. Low hotwell level is alarmed in the control room.

c. All feedwater heaters are provided with level controllers and drain control valves for normal drain disposition. High level drain control valves are provided for heaters Nos. 2, 3 and 4 to discharge into the condenser. Low, high and high-high levels are annunciated in the control room. High-high level actuates nonreturn and isolation valves in the extraction steam lines, or condensate isolation valves as applicable, to prevent water carryover to the turbine. Valve position monitoring lights are provided at the MCB for the feedwater heaters spill valves to the condenser and the extraction steam nonreturn and isolation valves.

Gauge glasses are provided for local direct observation of heater liquid levels.

Feedwater heater inlet, outlet and drain temperatures and shell side pressures are monitored and used for the performance computation.

d. The two heater drain pumps are controlled from the MCB. Suction and discharge pressure are indicated locally, and flow to the feed pump suction header is indicated in the control room. Instruments monitor heater drain tank level for indication, control, interlocks, and alarms. The heater drain tank is provided with level controllers which regulate the heater drain pump discharge, recirculation and condenser spill valves. The condenser spill valve is also opened by a high level switch. A high-high level switch initiates turbine water induction protection. Heater drain pump trip, motor bearing, and motor winding high temperatures are alarmed in the control room. The 10-inch spill valve opens on high level.

e. Steam generator feed pumps are provided with local suction and discharge pressure gauges. The indications of these pressures for the steam generator feed pumps are also provided at the MCB.

A flow element is provided in the feedline to each steam generator. Pairs of taps provide for two independent metering measurements from the flow element. The feedwater flow measurements are indicated in the control room and utilized in the steam generator level control system. The details of the safety-related display instrumentation are presented in Section 7.5. The details of the steam generator water level control system are presented in Subsection 7.7.1.7. Manual-auto stations for all the feedwater control valves are provided at the MCB.

For computer trending, separate flow measuring loops utilizing ultrasonic flow transmitters are provided in each feedline to the steam generators.

To ensure that minimum flow requirements are met, a recirculation valve has been provided for each pump, controlled by the suction flow measuring venturi.

f. The steam generator feed pump turbine provides variable speed feed pump operation. The variable speed feed pump control system is described in Subsection 7.7.1.7. Feed pump bearing high temperatures are alarmed in the control room.

10.4.8 Steam Generator Blowdown System

10.4.8.1 Design Bases

The Steam Generator Blowdown System is designed to limit the concentration of dissolved and suspended solids in the shell (secondary) side of the steam generators, which are introduced into the steam generators through the feedwater. Removal of these solids minimizes chemical deposition on steam generator tube surfaces, thus limiting the reduction in heat transfer capability, as well as reducing the rate of steam generator tube corrosion. The potential sources of solids in the steam generators can be any, or a combination of, the following:

a. Chemical additions to the secondary system for corrosion control

b. Reactor coolant boric acid due to primary to secondary leakage through the steam generator tubes

c. Secondary system corrosion products

d. Seawater, due to leakage through the condenser tubes into condensate returning to the steam generator

e. Impurities in condensate makeup water.

The unit has an independent steam generator blowdown and sampling system. Initial processing (depressurization and cooling) of steam generator blowdown liquid is accomplished using the flash tank and bottoms coolers. If the radioactivity in the blowdown liquid is insignificant, then the liquid is returned to the Condensate System after processing through demineralizers, or discharged directly to the Circulating Water System.

The method to process radioactive secondary liquid from the steam generators is to direct steam blowdown flash tank bottoms cooler discharge to the floor drain tanks. If no secondary pressure is available, the steam blowdown and wet lay-up pumps can be used. From the floor drain tanks, processing takes place using the installed vendor system (WL-SKD-135) to the waste or recovery test tanks. (Reference Subsection 11.2.2.1.)

The unit has the continuous blowdown capability of 400 gpm (100 gpm per steam generator). This very high flow rate is used when high feedwater contaminant concentrations require such a blowdown rate. This high rate of blowdown is expected to occur at startup, after a unit shutdown in which maintenance has taken place on the secondary side of the plant. A high rate of blowdown may also be required at other times to control chemistry conditions. In the flash tank, 30 percent of this blowdown is flashed to steam, leaving 70 percent to be processed by the demineralizers, or discharged directly to the Circulating Water System.

Blowdown may be discharged directly to the Circulating Water System when cleanup capabilities are unavailable, as long as the limits of 10 CFR 20 (Appendix B, Table 2, Column 2 - instantaneous release) and 10 CFR 50 Appendix I (average annual release) are not exceeded.

Isolation valves are provided outside the Containment to close all blowdown lines on a "T" signal, Auto emergency feed pump start, or on a HELB signal.

Piping and valves from the steam generator up to and including the containment isolation valves, are Safety Class 2, and are designed to ASME Section III, Code Class 2 (see Section 3.2). Other piping and equipment in the steam generator blowdown systems are nonnuclear safety class, and are designed to ANSI B31.1.B-1971 and ASME Section VIII.

Blowdown water chemistry control parameters are established using the EPRI Secondary Water Chemistry Guidelines.

10.4.8.2 System Description and Operation

a. Normal Operation

Figure 10.4-10, Figure 10.4-11, Figure 10.4-12 and Figure 10.4-13 are flow diagrams of the system. Each of the four steam generators is provided with a bottom blowdown connection on the secondary side above the tube sheet. During normal operation, each steam generator undergoes continuous blowdown with the blowdown water passing through a containment isolation valve, flow meter, and system valves. A small quantity of blowdown is continuously drawn off automatically into the sample system through a sample heat exchanger for monitoring of the activity in the blowdown. If the activity in the blowdown discharge is higher than allowable (see Subsection 9.3.2), blowdown is automatically secured. The blowdown liquid then flows through a manual control valve which establishes the blowdown rate. Some of the liquid flashes upon passing through the control valve, and two-phase flow then enters the flash tank. There, approximately 30 percent of the blowdown flow exits the top of the tank as saturated steam. The remaining 70 percent exits the bottom of the tank as saturated water. The flash tank operates at 70 psia. Steam and water leaving the flash tank are directed as follows depending upon the existence and size of primary to secondary leakage.

1. If No Primary to Secondary Leakage Exists

   Flash tank steam will normally be directed back to the No. 3 feedwater heater or to the main condenser for that unit, or can be exhausted to the atmosphere if the heater and condenser are not available. Liquid from the flash tank will be cooled in the flash tank bottoms cooler and directed through a radiation monitor and flow totalizer into demineralizers. The demineralizers will remove chemical contaminants and radioactivity (as explained later), and the liquid can be transferred back to the condenser. Filters upstream of the demineralizers prevent the transport of corrosion products into the beds. An alternate path to discharge the blowdown liquid to the environment is available, that is, into the service water system discharge line via the Waste Liquid System (see Section 11.2). Flow into the flash tank under these circumstances can vary from 20 gpm (5 gpm per steam generator) to 400 gpm (100 gpm per steam generator).

2. If Primary to Secondary Leakage Exists

The early indication of a primary to secondary leak may be observed as a result of routine grab sampling (at very low leak rates) or on any of the following radiation monitors:

- Condenser air removal, RM 6505
- Blowdown Flash Tank, RM 6519
- Individual Blowdown line, RM 6510-6513 (see Subsection 9.3.2).

Each of the setpoints on these radiation monitors is established, using station procedures, such that plant personnel can detect indications of small leaks, during the early phases of the leak.

If activity in excess of the alarm setpoint is measured on the Flash Tank RM, the blowdown system will automatically isolate. The individual monitors on the blowdown lines will provide indication of which steam generator is leaking. New RM setpoints may be established to continue system operation, once a steady state has been achieved.

EPRI PWR Primary to Secondary Leak Guidelines identifies the significance levels of primary to secondary leaks. The operational plan for how to continue plant operation when leakage exists is located in plant procedures for management overview as well as individual departmental procedures for increased monitoring.

These administrative and operational procedures dictate how the steam generator blowdown liquid will be processed during a primary to secondary leak. The following options are available to operations personnel:

- Continue to use the blowdown demineralizers as in normal operation. This requires additional radiation monitoring in the Demineralizer Building to ensure general area dose rates are within Zone II limits, as well as increased grab sampling frequency to closely monitor the leak rate change.
- Use the Waste Liquid System to treat all or part of the flashed liquid, bottoms or both, from the blowdown flash tank. This may require a reduction in blowdown flow rate.

- Discharge to the environment within the confines of 10 CFR 20 and Appendix I of 10 CFR 50 limits.

If the activity in the flash tank steam is low, the steam may be allowed to return to the No. 3 feedwater heater for reuse in the plant without processing. Flashed steam is not intentionally released to the atmosphere. If significant quantities of radionuclides are contained in the flash tank distillate, the liquid may be processed through the flash steam condenser to the waste test tanks. This limits the flow from the four steam generators to approximately 75 gpm as the cooler capacity is only 25 gpm.

Processing of the steam generator flash tank bottoms liquid through the Waste Liquid System will also necessitate a total blowdown flow limit of 71 gpm for all four steam generators (the vendor-operated Liquid Waste Treatment System capacity for process flow is 50 gpm). The basis for this flow is further discussed in Appendix 11A.

b. Operation With the No. 3 Feedwater Heater or Main Condenser Not Available

When the No. 3 feedwater heater or main condenser for the unit is not available, system flows are realigned as follows:

1. If No Primary to Secondary Leakage Exists

Steam from the flash tank may be directed to the flash steam condenser/cooler and then pumped to the waste test tanks in the Liquid Waste System (see Section 11.2). Although under these conditions this liquid would not contain radioactivity, the contents of the waste test tank would be sampled before discharge to the service water system discharge, since the tanks could contain other processed liquid waste.

If blowdown flow requirements result in steam flow from the flash tank above the capacity of the flash tank condenser/cooler (which can process steam only equivalent to a maximum of 25 gpm of water), an additional path is provided to discharge the flash tank steam via the atmospheric exhaust.

Water from the flash tank will be handled in the same manner as in Subsection 10.4.8.2a.1.

2. If Primary to Secondary Leakage Does Exist

Steam from the flash tank is handled in the same manner as explained in b.1. However, the blowdown capacity is limited as explained in Subsection 10.4.8.2a.2. Radioactive steam is not charged to the atmosphere (see Subsection 10.4.8.2a.2).

Liquid from the flash tank will be handled in the same manner as explained in Subsection 10.4.8.2a.2.

c. Cooling Water Shortage Case

During low heat removal capability of the Primary Component Cooling Water (PCCW) System, steam generator blowdown capacity may be limited. Such circumstances may arise during heat treatment of service water system tunnels and initial phases of plant cooldown for a short duration. Additionally, on a "T" signal, cooling water to the Waste Processing Building is isolated. However, the evaporators are automatically shutdown on a "T" signal.

10.4.8.3 Component Design

a. Flow Control Valves

Initial blowdown liquid flashing will occur due to the pressure drop across the flow control valve associated with each steam generator's blowdown line. The flow control valves are sized to pass zero to 100 gpm flow and are designed to minimize noise, vibration and erosion.

b. Blowdown Flash Tank

The blowdown flash tank is sized to permit 400 gpm (200,000 lbm/hr) two-phase flow. Flow enters the tank through four tangential nozzles. A stainless steel wear plate is used to prevent tank erosion. Steam exiting the tank must pass through a mesh style deentrainment separator (demister pad) to limit carryover. The vessel is carbon steel and will operate at 70 psia with overpressure protection provided to limit any pressure excursion while passing maximum steam flow.

Vessel operating pressure is used to force water flow to the demineralizers, or to the service water system discharge piping. Vessel operating pressure is maintained by pressure control valves in the steam line or cooling water line to the flash tank steam condenser, depending upon mode of system operation. Operating level is maintained by one of two level control valves in the liquid discharge line.

c. Flash Tank Bottoms Coolers

The flash tank bottoms coolers are shell and tube heat exchangers sized so that flashing will not occur downstream under maximum blowdown conditions. Each cooler will handle 50 percent maximum flow.

d. Flash Steam Condenser/Cooler

The flash steam condenser is a shell and tube heat exchanger sized to condense approximately 11,500 lb/hr steam. Cooling water flow (PCCW) through the condenser is regulated to maintain 70 psia in the flash tank. The subcooling section at the bottom of the condenser cools the condensate to approximately 120°F.

e. Flash Tank Distillate Pumps

Two 50 gpm (nominal) centrifugal pumps are provided to pump liquid from the flash steam condenser to the main condenser or waste test tanks depending upon the system mode of operation. These pumps can also be used to evacuate the flash tank through a separate valve (normally kept closed) if necessary.

f. Steam Generator Blowdown Evaporators

Each of the calandria blowdown evaporators has the process liquid on the tube side and heating steam on the shell side. The condensers and coolers with the evaporators subsystem are similar to those with the Boron Recovery System (Subsection 9.3.5) and the Liquid Waste System (Section 11.2). However, the heating element is contained within the evaporators for the Steam Generator Blowdown System, whereas it is separate for the Boron Recovery System.

g. Demineralizers

The unflashed liquid from the flash tank is cooled and directed to demineralizers. The three beds are configured to be compatible with secondary system chemistry. The effluent quality will be in accordance with EPRI Secondary Water Quality Guidelines.

h. Booster Pumps

The booster pumps are required to pump the liquid from the flash tank through heat exchangers and demineralizers into the Condensate System. The booster pumps are especially required to overcome the hydraulic resistance in the downstream circuit at high flow rates.

i. Steam Generator Blowdown Recovery (SGBR) Heat Exchanger

The heat exchanger cools down the unflashed liquid from the flash tank to less than 110°F before it enters the demineralizers. This temperature is required to maintain the characteristics of the demineralizer resins.

j. Iron Filters

The iron filters are located upstream of the demineralizer beds and serve to prevent the transport of corrosion products into the beds, which can reduce bed efficiency. The filters comprise two disposable cartridge housings connected in a duplex piping arrangement. This configuration allows for continuous SB operation during cleaning of a filter.

Table 10.4-2 lists the design and operating conditions of the steam generator blowdown system components.

10.4.8.4 Safety Evaluation

The Steam Generator Blowdown System has no safety function, nor is its performance required during or after an accident. Accordingly, the system is designed as nonnuclear safety (NNS), non-Category I. However, from the connection on the steam generator to the containment isolation valves, just outside the Containment, the system is Safety Class 2 and seismic Category I. Some parts of the system may contain radioactive fluids, depending on the presence of steam generator tube leakage. Closure of the blowdown lines is accomplished by air-operated valves that close on high pressure or level in the flash tank or startup of the emergency feed pumps signal (Refer to Section 6.8). These valves are closed on loss of air pressure or electrical power to the solenoids, thus assuring the performance of the safety function under all failure conditions. Liquid discharge from the flash tank is automatically terminated on a high radiation signal in the discharge line or in the sample withdrawn from each steam generator.

Electrical power is provided at 460 volts, 3 phase, 60 Hz. Emergency electrical power is not provided. Each combination motor-starter incorporates thermal elements to protect against overloads and a magnetic molded case circuit-breaker to protect against faulted conditions.

Monitoring devices are provided to measure conditions of pressure, temperature, radiation, conductivity, flow, and liquid levels to ensure that the system is operated safely and within design limits. The design bases listed in Subsection 10.4.8.1 are met using the flash tank, demineralizer, and Waste Liquid System capabilities. A failure analysis is presented in Table 10.4-3.

10.4.8.5 Test and Inspections

Prior to initial plant startup, the Steam Generator Blowdown System is tested to verify proper operation of system equipment.

During normal plant operation, calibration of the radiation monitors (see Section 11.5) and surveillance testing of the containment isolation valves will be performed in accordance with Technical Specification requirements.

10.4.8.6 Instrumentation and Control

a. Flash Tank Subsystem

1. Containment Isolation Valves

These valves are controlled from the MCB. The outboard valves close automatically on a "T" signal, emergency feed pumps running, or a HELB signal (see Subsection 7.6.10). Flow of individual blowdown lines is indicated at the MCB and locally near the blowdown throttle valves.

2. Flash Tank Instrumentation

Level and pressure are indicated locally and at the MCB. Temperature of the tank is indicated at the MCB. High and low level, as well as hi and hi-hi pressure is alarmed at the MCB.

3. Flash Tank Control

(a) Pressure Control

In normal operation, the pressure of the tank is maintained at 70 psia by throttling the pressure control valve in the steam line to the condenser. During this time the line to the atmosphere is kept closed. When the main condenser or No. 3 feedwater heater is not available, the flash tank is aligned to the flash steam condenser. Pressure control is then achieved by throttling the cooling water valve at the outlet of the flash steam condenser. In case the main condenser, No. 3 feedwater heater, and flash steam condenser are not available, the flash tank steam is processed to atmosphere in a controlled manner.

(b) Level Control

In normal operation, the pressure of the tank is maintained at 70 psia by throttling the pressure control valve in the steam line to the condenser. During this time the line to the atmosphere is kept closed. When the main condenser or No. 3 feedwater heater is not available, the flash tank is aligned to the flash steam condenser. Pressure control is then achieved by throttling the cooling water valve at the outlet of the flash steam condenser. In case the main condenser, No. 3 feedwater heater, and flash steam condenser are not available, the flash tank steam is processed to atmosphere in a controlled manner.

When the flash tank distillate is aligned to the main condenser or waste test tank (WTT), which may be required to drain the steam generators, the level of the tank is controlled by throttling the level control valve at the common discharge header of the flash tank distillate pumps. High level will alarm and then isolate the discharge from the steam generators to the flash tank.

4. Flash Steam Condenser

The level of the flash steam condenser is maintained by throttling the control valve at the common discharge header of the flash tank distillate pumps. Temperature of the distillate is monitored by the plant computer. High and low level are alarmed in the control room.

5. Flash Tank Distillate Pumps

These pumps are used in two different stages of plant operation:

(a) Draining the steam generators via the flash tank, transferring flash tank concentrates to waste test tanks (WTT).

(b) Transferring the distillate from flash steam condenser to WTT or main condenser.

These pumps are controlled from the MCB. In both modes of operation, only one pump is run at a time. In mode (a) the pump is started manually and is stopped automatically on low-low flash tank level. In mode (b), the starting of the aligned pump is automatically initiated by high level in the distillate condenser and stopped manually. When pressure in the flash steam condenser reaches normal operating condition, the distillate circuit is manually aligned to the main condenser (or WTT) and the pump is stopped and isolated. Pump trip is alarmed in the control room. Rotation of pump duty is administratively controlled.

6. Flash Steam Condenser Vent Valve

The valve remains normally closed, and opens automatically at predetermined pressure and closes on high-high pressure to prevent ingress of steam in the vent gas system. The valve can be manually opened from the MCB by overriding pressure interlocks.

7. Radiation Monitoring

The flash tank liquid discharge to the environment is measured continuously and recorded and totalized at the MCB. When the discharge is complete, the totalized flow reading is recorded and forwarded to the chemistry department. Chemistry in turn utilizes this data to satisfy the monthly discharge surveillance requirements of the Technical Requirements Manual. This stream is continuously monitored for radioactivity and is isolated on high radiation and on high flash tank concentrates discharge temperature. High radioactivity is alarmed locally and at MCB. Additionally, high radioactivity in the blowdown sample lines isolates this flash tank liquid discharge stream. All these high radioactivity interlocks, with the exception of high radiation and temperature flash tank concentrate discharge stream, can be manually overridden. This flash tank liquid discharge radiation monitor is reset by flushing the sampling line and draining the trapped liquid to the floor drain tanks. The high temperature interlock is reset by re-establishing cooling water flow through the flash tank bottoms cooler.

b. <u>Evaporator Subsystem</u>

The evaporator subsystem, of the Steam Generator Blowdown System, is not immediately available for use. Prior to any anticipated startup of this subsystem, plant management would be notified to plan for any training, procedure updates, and pre-operational testing required.

Control and instrumentation for all these evaporators are located at the waste management system (WMS) control panel in the Waste Processing Building. Each evaporator can be individually controlled. Normally, the evaporators operate automatically. The initial starting is, however, manual via a master selector switch.

The prime objective of the evaporator control system is to maintain the level constant by manipulating the feed and the auxiliary steam. This is achieved by controllers at the WMS control panel. Controllers also maintain auxiliary control loops, such as evaporator pressure, distillate and concentrates cooling water temperature, at stable values. In case any one of these controllers are lost, a manual control with aid of backup instrumentation maintains evaporator operation. Additionally, instrumentation for feed, temperature, pressure and level of evaporation is provided at the WMS control panel.

The following conditions are alarmed on the local control panel:

1. High, high-high, low pressure and low-low level in the evaporators
2. High and low level in the distillate condensers
3. High vent temperature in the distillate condensers
4. High and low temperature of concentrate outlet at bottoms coolers
5. High distillate cooler outlet temperature and conductivity
6. High auxiliary steam flow
7. Pumps trip
8. Evaporator bottoms pump seal water pressure low.

   All low, low-low level, temperature and pressure alarms are suppressed in shutdown mode.

c. Demineralizer Subsystem

The demineralizer subsystem, along with its regenerative system and controls, is located in a separate room next to the Waste Processing Building. Cooled unflashed liquid is directed to the demineralizer. If the liquid temperature is higher than a predetermined value. The service valve will be closed and the liquid will bypass the demineralizer skid (SB-SKD-95). The influent and effluent conductivities of the demineralizer are measured by conductivity cells to monitor bad performance.

Regeneration is initiated by push-button and carried forward automatically until completed. Units are returned to service manually. Since the acid and caustic systems are the same for all mixed beds, interlocks are provided to allow one unit at a time to go on regeneration. The SG blowdown regenerate will be monitored for radioactivity and may be processed through the Liquid Radwaste System. If the radioactivity in the regenerate solution is determined to be of such low concentration and total quantity that the Technical Specification dose limits for demonstrating compliance with the "As Low As Reasonably Achievable" of 10 CFR Part 50, Appendix I can be met, the regenerate solution may be discharged to the Circulating Water System through the liquid waste discharge header which includes a radiation monitor that will terminate the release if higher than expected activity is detected.

Level loops indicate the levels of the acid and caustic tanks, and control the regeneration process. Low level in the acid or caustic tank prevents regeneration, as does high level in the waste holdup sump.

The contents of the waste holdup sump are sampled to insure that the radionuclide concentration does not exceed the limits of the Offsite Dose Calculation Manual (ODCM). If it does, the contents of the sump are discharged to the chemical drain treatment tanks for processing as radioactive liquid waste. If it does not, the contents of the sump are discharged to the liquid waste systems (LWS) test tank discharge header upstream of the LWS test tank discharge monitor (see Subsection 11.5.2.1).

The following conditions are alarmed on the local control panel:

1. High inlet temperatures to demineralizers

2. High differential pressure on demineralizers

3. High demineralizer influent and effluent conductivity

4. High waste holdup sump level

5. Pumps trip

6. Demineralizer skid service air pressure low.

10.4.9 Auxiliary Feedwater System

A typical pressurized water reactor plant includes an Auxiliary Feedwater System that normally operates during startup, hot standby, and hot shutdown. In addition, it also functions as an emergency heat removal system to transfer heat from the primary system when the Main Feedwater System is not available.

For the Seabrook plant, the functions of the Auxiliary Feedwater System are fulfilled by the startup feed pumps during startup, hot standby and hot shutdown conditions (see Subsection 10.4.12) and the Emergency Feedwater System during loss of normal feedwater flow (see Section 6.8).

10.4.10 Secondary Component Cooling Water System

10.4.10.1 Design Bases

The Secondary Component Cooling Water (SCCW) System is a nonsafety-related system and has no safety design basis. The following design criteria are applicable to this system:

a. The system is designed to remove heat from the turbine-generator accessories and other auxiliary equipment located in, and adjacent to, the Turbine Building. This heat is transported by the recirculating water of the SCCW system to the SCCW heat exchanger or auxiliary SCCW heat exchanger, where it is transferred to the Service Water System.

b. The system is designed for a maximum service water temperature of 80°F. The minimum service water temperature is 35°F. Refer to the Service Water System (Subsection 9.2.1) for further information.

c. The system is designed to provide a maximum secondary component cooling water temperature of 95°F to the equipment coolers. However, the actual cooling water temperature is dependent on the varying SCCW heat loads and the service water inlet temperature.

d. The following codes and standards are applicable for the system:

1. American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section VIII, Division I, Unfired Pressure Vessels
2. Heat Exchanger Institute (HEI) Standards for Heat Exchangers
3. Tubular Exchanger Manufacturers Association (TEMA) Standards, Class "C"
4. American Society for Testing and Materials Standards for Centrifugal Pumps
5. Hydraulic Institute Standards for Centrifugal Pumps
6. American National Standard Institute (ANSI)
7. Power Piping Code B31.1

10.4.10.2 System Description

The Secondary Component Cooling Water System, as indicated on the system flow diagrams, Figure 10.4-14 is designed as a closed loop through the secondary component cooling water pumps, heat exchangers and piping to and from the turbine generator accessories and auxiliary equipment. A head tank is also included to provide makeup water and to act as a surge tank for the system.

The turbine generator accessories and other auxiliary equipment are equipped with auxiliary coolers that have been designed for a maximum cooling water (SCCW) temperature of 95°F. The actual temperature of the water supplied to these coolers will vary from 85°F to 95°F, depending on the SCCW system heat load and service water inlet temperature. This temperature range for the recirculating water will be maintained by a thermally controlled bypass around the SCCW heat exchanger. Uncooled recirculating water will be piped through the bypass and allowed to mix with the SCCW heat exchanger cooled water. A sensor located downstream of the mixing point will be set to maintain a minimum SCCW temperature of 85°F. These temperature controls are on the SCCW side, rather than on the service water side, to avoid interaction with the service water pumps and the Primary Component Cooling Water System.

Each component auxiliary cooler has inlet and outlet valves for isolation and throttling during initial startup. The secondary headers to the various component auxiliary coolers have branch isolation valves to isolate these headers from the main header for maintenance.

The system is filled with demineralized water from the demineralized water makeup system. A corrosion inhibitor is added to the SCCW system through the chemical feed tank, located near the SCCW pumps. In addition the system is provided with a bypass filter to remove foreign particulates from the water.

The system contains three horizontal centrifugal pumps. Only two pumps are required for full load operation, with the third pump on standby. Two main heat exchangers are supplied. These heat exchangers are horizontal, tubular, counterflow, two-pass, shell-and-tube, with fixed tube sheets and removable covers. The shell is constructed of carbon steel; the tube sheets are carbon steel with 90-10 copper nickel cladding. All internal carbon steel surfaces on the tube side of the exchangers will be protected against seawater corrosion by a neoprene lining. Tubes are fabricated of 90-10 copper nickel due to its resistance to seawater corrosion. Two smaller auxiliary heat exchangers are installed in parallel to the main heat exchangers. The auxiliary heat exchangers are sized for very low loads, such as during outages, up to approximately 5 percent power. These heat exchangers are horizontal, shell and tube, counterflow, single pass. The shell is constructed of carbon steel, which will be exposed to the SCCW flow. The tubes and tubesheets are 90-10 CuNi for corrosion resistance from seawater. The channel and cover are coated cast iron.

10.4.10.3 Safety Evaluation

The design and operation of the SCCW system has no safety-related function, and no safety-related evaluation is required. However, since the SCCW system is cooled by the Service Water System, it will be affected to a degree by changes in the Service Water System. The supply of service water to the SCCW heat exchanger will be isolated upon receipt of either a safety injection signal, a cooling tower actuation signal, or a loss of offsite power (see Subsection 9.2.1).

10.4.10.4 Inspection and Testing Requirements

Prior to initial plant startup, the SCCW system is operationally tested to insure that it will perform properly.

10.4.10.5 Instrumentation

a. SCCW Pumps

These pumps are controlled from the MCB. Normally, two pumps are running, with the third pump in standby. A loss of either of the running pumps will automatically start the third pump. Once started, this pump will continue to run until secured by plant personnel.

The pump discharge header pressure is indicated in the control room. Low discharge header pressure, as well as the loss of either pump, is alarmed in the control room. Pump suction and discharge pressure is locally indicated.

b. SCCW Head Tank

Tank level is locally indicated by a level gauge. The low and high level are alarmed in the control room. The level of the head tank is controlled automatically by a level controller that regulates the flow of the demineralized makeup water supply.

c. SCCW Heat Exchangers

Temperature measured at the discharge of the heat exchangers is used to control SCCW temperature by blending the cooled water through the heat exchangers and the warm water through the bypass line. The temperature controller is local. High and low heat exchanger discharge temperatures are alarmed in the control room.

Each heat exchanger's inlet and outlet temperature is locally indicated, and shell pressure of each heat exchanger is also locally provided.

d. Miscellaneous Heat Exchangers

For miscellaneous heat exchangers, outlet temperature and inlet and outlet discharge pressures are locally indicated.

Individual local temperature controllers are provided to control the SCCW flow through the various heat exchangers serviced by the SCCW system. SCCW flow to the air compressor is isolated in the event of a loss of offsite power.

e. Radiation Monitoring

No radiation monitoring is required in the SCCW system, since it does not service any heat exchanger used in systems carrying primary coolant or other radioactive fluid systems.

10.4.11 Auxiliary Steam System

10.4.11.1 Design Basis

The Auxiliary Steam System provides low pressure saturated steam for various plant components and systems, satisfying the following requirements:

a. Building heating and miscellaneous plant requirements.

b. During plant operation and shutdown, auxiliary steam is supplied to the steam generator blowdown evaporator, the boron recovery and waste processing evaporators, letdown and boron recovery degasifiers and various tanks. During plant shutdown, the auxiliary steam requirements for the above systems may exceed the capacity of the auxiliary boilers, requiring the systems to operate at reduced capacity.

c. During plant startup, auxiliary steam is used for turbine gland sealing.

The Auxiliary Steam System is designed in accordance with the Power Piping Code, ANSI B31.1. The auxiliary boilers are designed in accordance with ASME Code Section I, Power Boilers. The deaerator and blowdown tank are designed in accordance with ASME Code Section VIII, Division 1. The Safety Class 3 portion of the system is designed to ASME Code Section III, Subsection ND, Class 3 Components.

10.4.11.2 System Description

The Auxiliary Steam System shown on Figure 10.4-16 is comprised of the following equipment:

a. Two package boilers, each rated at 80,000 lbs/hr of saturated steam at 150 psig, complete with forced draft fans, breeching and common stack

b. One 170,000 lb/hr de-aerating heater with storage tank

c. Three motor-driven boiler feed pumps rated at 180 gpm each (one spare)

d. Triplex fuel oil pumping set (one spare pump)

e. One blowdown tank, one fuel oil storage tank and two skid-mounted chemical feed units

f. Interconnecting piping

g. Safety-related PAB isolation valves.

The Auxiliary Boiler System is shown on Figure 10.4-15 and the Auxiliary Steam Condensate System is shown on Figure 10.4-17.

During plant start-up excess condensate from auxiliary steam used for turbine gland sealing and shell warming is returned to the Auxiliary Steam Condensate System.

Feedwater from the de-aerator is pumped to the auxiliary boilers and evaporated. Steam is piped to building heating units and operating equipment. Building heating system condensate and the equipment steam and/or drains are added to the main cycle or returned to the auxiliary boiler de-aerator.

The boilers are fired by No. 2 fuel oil. Steam atomization is used during normal boiler operation. Air is the atomizing medium for startup.

During normal plant operation, a branch line from main steam lines can supply the required steam to the Auxiliary Steam System. A pressure-reducing valve reduces the main steam pressure to that equivalent to the output of the auxiliary boilers. The pressure reducing station is closed during station startup, when the auxiliary boilers furnish the required steam.

The auxiliary steam PAB isolation valves are operable from the MCB and close automatically on a HELB signal.

10.4.11.3 Safety Evaluation

In the event that any of the systems being supplied with auxiliary steam become contaminated, the auxiliary condensate will in turn become contaminated. To prevent the auxiliary boiler from becoming contaminated, each unit is equipped with a radiation monitor which samples the condensate in the condensate return line. If the radionuclide concentration exceeds a preselected level, the monitor automatically terminates the condensate return. This device is described in greater detail in Subsection 11.5.2.1 and Table 11.5-1.

The operation of the Auxiliary Steam System is not required under emergency conditions. However, on a high energy line break in the PAB, safety-related valves isolate this system.

10.4.11.4 Tests and Inspections

The Auxiliary Steam System is hydrostatically tested in accordance with ANSI B31.1 requirements, and functionally tested to verify control functions. Safety Class 3 portions of the system will be hydrostatically tested in accordance with ASME Code III, Subsection ND, Class 3 Components.

The system is functionally tested to ensure proper operation.

10.4.11.5 Instrumentation and Control

Instrumentation and control associated with the Auxiliary Steam System are located primarily at the auxiliary boiler control (ABC) panel in the Administrative Building. The principal subsystems are identified as follows:

a. Combustion Control: This subsystem provides for automatic control of fuel and combustion air. Automatic correction of fuel-air ratio is provided by O_2 measurement of the flue gas and is used to bias the demand for air flow.

b. Boiler Safety: This subsystem provides for safe operation of the boiler, where a sequence of purging, operation of the fuel system and ignition of burner flame is coordinated in a manner which will preclude any potentially explosive situation, thereby ensuring boiler safety at all times.

c. Auxiliary Boiler: This subsystem utilizes a two-element control system which regulates feedwater flow as a function of steam flow demand. Drum level signal corrects steam-feedwater flow mismatch, and adjusts feedwater flow accordingly. Pressure is indicated locally and remotely at the ABC panel. Flue gas temperature, air velocity, oxygen concentration in flue gas, steam flow and drum level are recorded at the ABC panel. Deviant parameters are alarmed at the ABC panel, and are grouped together in the control room as an "auxiliary boiler trouble" alarm.

d. Boiler Feed Pumps: These pumps are controlled from the ABC panel, are sequenced to start automatically on selected pump, and are interlocked with the de-aerator tank level.

e. Auxiliary Steam Distribution: Auxiliary steam distribution to branch lines is reduced in pressure and fed to the plant for various uses. Local pressure indication is available at each branch line.

10.4.12 Startup Feedwater System

10.4.12.1 Design Bases

The Startup Feedwater System is a nonsafety-related system and, therefore, has no safety design bases. The system is designed to:

a. Provide a supply of feedwater to the steam generators during plant startup to fill and pressurize the steam generators

b. Provide sufficient feedwater flow to the steam generators to allow steam to be utilized for turbine plant warmup and turbine operations up to 5 percent of full load, prior to operation of the main feed pumps

c. Provide sufficient feedwater flow to the steam generators to allow the reactor plant to operate at low load (hot standby) while the turbine plant is not operating

d. Minimize thermal transients in the steam generator feedwater lines

e. The system is designed to the following codes and standards:

   1. ANSI B31.1, Code for Power Piping

   2. Hydraulic Institute Standards.

10.4.12.2 System Description

The Startup Feedwater System consists of a startup feed pump and interconnecting piping. For a diagram of this system, see Figure 10.4-8 and Figure 10.4-9.

The startup feed pump is a horizontal, centrifugal, motor-driven pump which takes suction from the condensate storage tank and discharges into the main feedwater piping at the discharge of the main feed pumps. If the main condenser is not under vacuum, the startup feed pump can also take suction from the condenser hotwell. (If desired, the pump suction may also be aligned to the Demineralized Water Storage Tanks as a backup water source.) Provisions for an auxiliary steam supply to the No. 6 heaters and to the startup heater allow a degree of feedwater heating to minimize thermal transients in the steam generator feedwater lines.

During operation of the Startup Feedwater System, flow to the steam generators is controlled by the feedwater control bypass valves, either manually or automatically, using a steam generator level signal.

The startup feed pump (SUFP) motor can be powered by nonemergency Bus 4 or emergency Bus E5. For additional electrical design details, refer to Subsection 8.3.1.1.b.9.(a).

During no-load and low load plant operations, the SUFP is aligned to non-emergency Bus 4 to provide its startup and shutdown functions. After the SUFP completes its startup function, its power supply will be transferred to emergency Bus E5 as plant power is increased. The SUFP will remain aligned to Bus E5 during 100% power operation. As power is decreased in preparation for a plant shutdown, the SUFP power supply will be transferred back to Bus 4. If the SUFP is required to perform its EFW contingency function while aligned to Bus 4 coincident with a loss of offsite power, it will have to be manually transferred to Bus E5 and manually started.

Valves FW-V163 and FW-V156 are normally closed, but can be opened to permit the startup feed pump to supply SG feedwater via the EFW supply headers. These valves are equipped with motor operators which are powered by a Bus E5 power supply and are controlled from the main control board.

10.4.12.3 Safety Evaluation

The Startup Feedwater System is not part of the Engineered Safety Systems, and is not required for maintenance of plant safety in the event of an accident.

10.4.12.4 Tests and Inspection

The system will be performance tested prior to initial plant startup.

10.4.12.5 Instrumentation

Control of feedwater flow to the steam generators during startup feedwater operation uses a portion of the same narrow-range steam generator level channels used during normal feedwater operation.

The startup feed pump is started from the main control room for startup operations. The startup feed pump motor can be powered from non-emergency Bus 4 or emergency Bus E5. The status of Bus 4 and Bus E5 is monitored in the main control room. While aligned to Bus 4, the startup feed pump will automatically start if both steam-driven feed pumps trip. Automatic starting of the startup feedpump is blocked if an "S" signal or hi-hi steam generator level signal is generated to trip the feed pumps. This blockage of the startup feed pump is not safety-related, but takes place because the feedwater control bypass and isolation valves are also closed. While aligned to Bus E5, the SUFP control switch will be maintained in pull-to-lock to prevent inadvertent start on Bus E5. The SUFP will only be manually started on Bus E5 to perform its EFW contingency function, for quarterly SUFP surveillance testing, to support shutdown after a plant trip, or as needed to support maintenance retest. The SUFP will only be manually started, while the emergency diesel generator (EDG) is supplying Bus E5, to perform its EFW contingency function or for 18 month EDG surveillance testing. The startup feed pump motor bearing and motor winding high temperature are alarmed in the main control room.

SEABROOK STATION
UPDATED FINAL SAFETY ANALYSIS REPORT

CHAPTER 10
STEAM AND POWER CONVERSION SYSTEM

TABLES

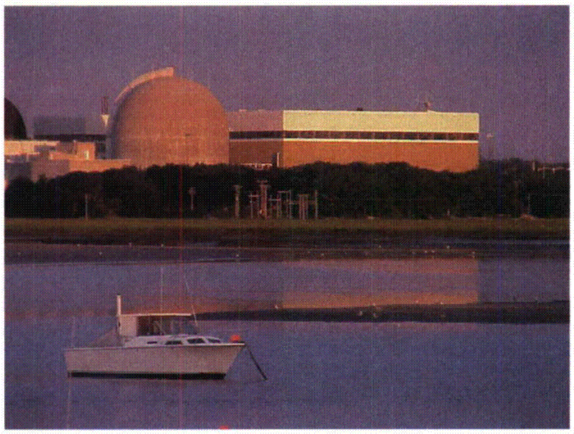


TABLE 10.3-1 Deleted

TABLE 10.4-1 CONDENSATE AND FEEDWATER SYSTEM COMPONENT DESIGN DATA

1. Condensate Pumps

Number	3
Type	Vertical, multistage
Capacity, per pump	11,300 gpm
Total developed head @ design capacity	1,065 feet
Temperature of condensate	100°F
Motor rating	3,500 hp

2. Steam Packing Exhauster

Number	1
Tube material	304 SS
Channel design pressure	600 psig
Channel design temperature	125°F

3. Feedwater Heaters

Number	16
Type	Closed, horizontal U-tube
Tube Material	304 SS

Heater No.		Channel Design Pressure (psig)	Channel Design Temperature (°F)
1.	3	600	370
2.	3	600	370
3.	3	600	370
4.	3	600	370
5.	2	600	425
6.	2	1500	600

4. Steam Generator Feedwater Pumps

Number	2
Type	Horizontal, single stage
Capacity, per pump	18,662 gpm*
Total developed head @ design capacity	2,196 ft*
Temperature of feedwater	373°F*

5. Steam Generator Feedwater Pump Turbines

Number	2
Turbine drive rating	12,417 hp

6. Heater Drain Pumps

Number	2
Type	Vertical, multistage
Capacity, per pump	5,917 gpm*
Total developed head @ design capacity	691 ft*
Temperature of condensate	373°F*
Motor rating	1,250 hp

7.	Startup Heater	Tubeside Design Pressure (psig) 1500	Tubeside Design Temperature (°F) 400
		Shellside Design Pressure (psig) 225	Shellside Design Temperature (°F) 250

* Reflects post-uprate design heat balances.

TABLE 10.4-2 STEAM GENERATOR BLOWDOWN SYSTEM COMPONENT DATA

Flash Tank

Number	1
Design Pressure	150 psig to full vacuum
Operating Pressure	55 psig
Design Temperature	366°F
Operating Temperature	303°F
Capacity (Input)	200,000 lb/hr
Capacity	2700 gal.
Materials	Carbon Steel (Shell), Type 304 SS (Internals)
Design Code	ASME Sec. VIII, Div. 1
Safety Class	NNS

Flash Tank Bottoms Cooler

Number	2
Heat Exchange Rate	7,220,000 Btu/hr
Design Code	ASME Sec. VIII and TEMA C
Safety Class	NNS

	Shell Side	Tube Side
Design Temperature, °F	366	366
Design Pressure, psig	150	150 to full vacuum
Operating Pressure	85	55
Design Flow, lb/hr	240,300	70,000
Fluid	Cooling Water	Blowdown
Temperature in, °F	102 (max.)	303
Material	Carbon Steel	304L SS

Flash Steam Condenser/Cooler

Number	1	
Heat Exchange Rate	12,520,000 Btu/hr	
Design Code	ASME Sec. VIII and TEMA C	
Safety Class	NNS	
	Shell Side	Tube Side
Design Temperature, °F	366	366
Design Pressure, psig	150 to full vacuum	150
Operating Pressure, psig	55	85
Design Flow, lb/hr	11,500	417,000
Fluid	Steam	Cooling Water
Temperature in, °F	303	102
Temperature out, °F	130	132
Material	304L SS	Carbon Steel with 304L tubes SS

Flash Tank Distillate Pumps

Number	2
Design Flow	50 gpm
Design TDH	110 ft
Material	316 SS
Design Pressure	250 psig
Design Temperature	366°F
Design Code	Mfg. Standard
Safety Class	NNS

Vapor Body (Evaporator)

Number	2
Design Throughput	13,511 lb/hr (Feed)
Reflux Rate	2,574 lb/hr
Volume Holdup	1300 gals.
Safety Class	NNS
Design Pressure	50 psig to full vacuum
Operating Pressure	15 psig
Design Temperature	300°F
Material	Incoloy 825 (Tower 316L SS)

Heating Element (Evaporator)

Heat Transfer Rate	17,153,000 Btu/hr
Design Code	ASME Sec. VIII and TEMA C

	Shell Side	Tube Side
Design Temperature, °F	375	300
Design Pressure, psig	150 to full vacuum	150 to full vacuum
Operating Pressure, psig	125	15
Design Flow	20,618 lb/hr	16,085 lb/hr
Fluid	Steam/Condensate	Process Liquid
Temperature in, °F	353	200
Temperature out, °F	353	253
Material	Carbon Steel	Incoloy 825

Evaporator Distillate Condenser

Number	2	
Heat Exchange Rate	15,200,000 Btu/hr	
Design Code	ASME Sec. VIII, Div. 1 and TEMA C	
Safety Class	NNS	
	Shell Side	Tube Side
Design Temperature, °F	300	200
Design Pressure, psig	150 to full vacuum	150
Operating Pressure, psig	15	85
Design Flow	16,085 lb/hr	1,022 gpm
Fluid	Distillate	Cooling Water
Temperature in, °F	250	85
Temperature out, °F	250	115
Material	304 LSS	Carbon Steel with 304L SS Tubes

Evaporator Distillate Accumulator

Number	2
Capacity	300 gals.
Material	304L SS
Design Pressure	Full Vacuum to 50 psig
Operating Pressure	15 psig
Design Temperature	300°F
Operating Temperature	250°F
Design Code	ASME Sec. VIII, Div. 1
Safety Class	NNS

Evaporator Distillate Pump

Number	2
Design Flow	35 gpm
Design TDH	210 ft
Material	316 SS
Design Pressure	150 psig
Design Temperature	300°F
Design Code	Manufacturer's Standards
Safety Class	NNS

Evaporator Distillate Cooler

Number	2
Heat Exchange Rate	1,762,200 Btu/hr
Design Code	ASME Sec. VIII, Div. 1, and TEMA C
Safety Class	NNS

	Shell Side	Tube Side
Design Temperature, °F	200	300
Design Pressure, psig	150	150
Operating Pressure, psig	85	50
Design Flow	177 gpm	13,511 lb/hr
Fluid	Cooling Water	Distillate
Temperature in, °F	85	253
Temperature out, °F	105	120
Material	Carbon Steel	304L SS

Evaporator Bottoms Pump

Number	2
Design Flow	15 gpm
Design TDH	35 ft.
Material	Gould-Alloy
Design Pressure	150 psig
Design Temperature	300°F
Design Code	Manufacturer's Standards
Safety Class	NNS

Evaporator Bottoms Cooler

Number	2
Heat Exchange Rate	463,560 Btu/hr
Design Code	ASME Sec. VIII, Div. 1
Safety Class	NNS

	Shell Side	Tube Side
Design Temperature, °F	200	300
Design Pressure, psig	150	150
Operating Pressure, psig	85	50
Design Flow	47 gpm	15 gpm
Fluid	PCCW	Concentrate
Temperature in, °F	85	253
Temperature out, °F	105	180
Material	Carbon Steel	Incoloy 825

Piping

Material	Austenitic and High Nickel Stainless Steel
Design Code	ANSI B31.1
Safety Class	NNS

Demineralizer

Number	3
Type	Mixed bed
Orientation	Vertical
Size	4'-0" diameter x 8'-0" straight
Resin Volume/Manufacturer	
Cation	22 cu. ft/Diamond Shamrock (Duolite) No. C-26TR
Anion	22 cu. ft/Diamond Shamrock (Duolite) No. A-161TR
Inert	6 cu. ft/Diamond Shamrock (Duolite) No. S-3TR
Bed Depth	4 ft
Design Flow	50 gpm normal, 280 gpm maximum
Backwash Flow	31 gpm
Rinse Flow	75 gpm
Design Pressure	300 psig
Design Temperature	150°F
Materials	316 SS vessel & internals
Design Code	ASME Sect. VIII
Safety Class	NNS

Booster Pump

Number	1
Design Flow	350 gpm (includes 60 gpm min. circulation flow)
Design TDH	200 feet
Material	316 SS
Design Pressure	150 psig
Design Temperature	300 °F
Design Code	Manufacturer's Standards
Safety Class	NNS

SGBR Heat Exchanger

Number	1
Heat Exchange Rate	11,200,000 Btu/hr.
Design Code	ASME Sect. VIII & TEMA BFU
Safety Class	NNS

	Shell Side	Tube Side
Design Temperature, °F	200	300
Design Pressure, psig	150	300
Operating Pressure, psig	85	150
Design Flow, lb/hr.	280,000	140,000
Fluid	Cooling Water	Blowdown
Temperature in, °F	85	185
Temperature, out, °F	125	110
Material	Carbon Steel	Steel with Stainless Tubes

Resin Traps	Process Flow	Waste Flow
Number	1	1
Design Flow	50 to 280 gpm	75 gpm
Design Pressure	300 psig	(Later)
Design Temperature	300 °F	100 °F
Differential Pressure Clean/Dirty	4/5 psi	4/5 psi
Size	0.01 slots	0.05 slots
Material	304 SS	304 SS
Design Code	Manufacturers Std.	Manufacturers Std.
Safety Class	NNS	NNS

Iron Filters	
Number	2
Design Flow	280 gpm
Design Pressure	240 psi
Design Temperature	100°F
Material	Carbon Steel
Design Code	ASME Sect. VIII
Safety Class	NNS

TABLE 10.4-3 STEAM GENERATOR BLOWDOWN SYSTEM MALFUNCTION/FAILURE ANALYSIS

Component	Accident or Malfunction	Comments and Consequences
System	LOCA/LOOP**	The system does not have to be operational during accident conditions. Therefore it can be shutdown.
Pressure Vessels	Overpressure	Automatic controls and safety relief valves are provided.
System	Failure to Function	If the flash tank is out of service for repairs, the blow-down will have to be stopped. Evaluation of secondary chemistry will need to be used for outage time; this might eventually result in a unit shutdown
Tanks & Piping	Rupture	The safety relief valves on the pressurized systems are set at pressures below the design pressures considering reasonable transients in the system. In spite of this, should a rupture occur, safety related structures and equipment will not be flooded. The portion of the piping within the Containment is designed to safety class 2 and is designed/supported for the corresponding seismic and other loads.
Instrumentation	Malfunction	Two level instruments, one for process control and indication and the other for indication and the other for indication and alarm, are provided on all the essen-tial equipment of the process. Moreover, the I&C are located outside the boron concentration areas to provide easy access during operation.

Component	Accident or Malfunction	Comments and Consequences
Containment Isolation Valves	Air or Electrical power failure to solenoid valve	Valves are designed to fail closed.
	Failure of one train "T" signal	Dual solenoid valves are provided. Each independently receive a "T" signal from the A and B Train respectively.

**OP = Loss of Offsite Power.

TABLE 10.4-4 CONDENSATE POLISHING SYSTEM COMPONENT DATA

<u>Cation Bed Demineralizer</u>

Number	3
Orientation	Vertical
Size	9' 6" diameter x straight
Resin Volume	212 cu ft
Design Flow, cross-sectional	52 gpm/ft^2
Design Pressure	150 psig
Design Temperature	150°F
Materials	Carbon steel, lined, with stainless steel internals
Design Code	ASME Section VIII
Safety Class	NNS

<u>Mixed Bed Demineralizer</u>

Number	4
Orientation	Vertical
Size	8' diameter x straight
Resin Volume	38 cu ft cation; 113 cu ft anion
Design Flow, cross-sectional	50 gpm/ft^2
Design Pressure	150 psig
Design Temperature	150°F
Materials	Carbon steel, lined, with stainless steel internals
Design Code	ASME Section VIII
Safety Class	NNS

<u>Cation Bed Resin Strainer</u>

Number	3
Design Differential Pressure	165 psig
Design Temperature	150°F
Design Flow	3,750 gpm
Particle Retention	100 mesh
Material, vessel	Stainless steel, with stainless steel element

<u>Mixed Bed Resin Strainer</u>

Number	4
Design Differential Pressure	165 psig
Design Temperature	150°F
Design Flow	2,500 gpm
Particle Retention	100 mesh
Material, vessel	Carbon steel, with stainless steel element

<u>Low Conductivity Tank</u>	
Number	1
Design Pressure	Atmospheric
Design Temperature	120°F
Capacity	32,000 gal
Materials	Carbon steel, lined
Design Code	ANSI/AWWA D100
Safety Class	NNS
<u>Condensate Polishing Head Tank</u>	
Number	1
Design Pressure	150 psig/Full vacuum
Design Temperature	150°F
Capacity	7,000 gal
Materials	Carbon steel
Design Code	ASME Section VIII
Safety Class	NNS
<u>Condensate Polishing Pump</u>	
Number	1
Design Flow	7,500 gpm
Design TDH	230 ft
Material	Carbon steel with stainless steel shaft and impeller
Design Pressure	150 psig
Design Code	Manufacturer's Standard
Safety Class	NNS
<u>Recycle Pump</u>	
Number	2
Design Flow	3,000 gpm
Design TDH	116 ft
Material	Carbon steel, with stainless steel shaft and impeller
Design Pressure	275 psig
Design Code	Manufacturer's Standard
Safety Class	NNS

Low Conductivity Pump

Number	2
Design Flow	300 gpm
Design TDH	60 ft
Material	Carbon steel, with stainless steel shaft and impeller
Design Pressure	150 psig
Design Code	ANSI B73.1M
Safety Class	NNS

Acid Pump

Number	2
Design Flow	10 gpm
Material	Alloy 20
Design Pressure	150 psig
Design Code	Manufacturer's Standard
Safety Class	NNS

Caustic Pump

Number	2
Design Flow	10 gpm
Material	316 stainless steel
Design Pressure	150 psig
Design Code	Manufacturer's Standard
Safety Class	NNS

Separation Tank

Number	1
Design Pressure	150 psig
Design Temperature	150°F
Materials	Carbon steel, lined
Design Code	ASME Section VIII
Safety Class	NNS

Anion Regeneration Resin Mix and Storage Tank

Number	1
Design Pressure	150 psig
Design Temperature	150°F
Materials	Carbon steel, lined
Design Code	ASME Section VIII
Safety Class	NNS

Mixed Bed Cation Regeneration Tank

Number	1
Design Pressure	150 psig
Design Temperature	150°F
Materials	Carbon steel, lined
Design Code	ASME Section VIII
Safety Class	NNS

Cation Regeneration Tank

Number	1
Design Pressure	150 psig
Design Temperature	150°F
Materials	Carbon steel, lined
Design Code	ASME Section VIII
Safety Class	NNS

Cation Storage Tank

Number	1
Design Pressure	150 psig
Design Temperature	150°F
Materials	Carbon steel, lined
Design Code	ASME Section VIII
Safety Class	NNS

Sulfuric Acid Tank

Number	1
Design Pressure	Atmospheric
Design Temperature	110°F
Capacity	4,170 gal
Materials	Carbon steel, lined
Design Code	ASME Section VIII, not stamped
Safety Class	NNS

Sodium Hydroxide Tank

Number	1
Design Pressure	Atmospheric
Design Temperature	110°F
Capacity	4,170 gal
Materials	Carbon steel, lined
Design Code	ASME Section VIII, not stamped
Safety Class	NNS

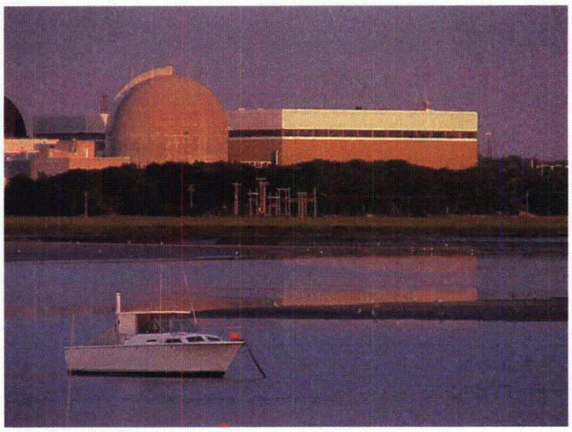
Cation Dilution Water Heater Tank

Number	1
Design Pressure	150 psig
Design Temperature	250°F
Capacity	4,000 gal
Materials	Carbon steel, lined
Design Code	ASME Section VIII, Div. 1
Safety Class	NNS

SEABROOK STATION
UPDATED FINAL SAFETY ANALYSIS REPORT

CHAPTER 10
STEAM AND POWER CONVERSION SYSTEM

FIGURES



See 1-NHY-B202116

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Heat Balance - 100% Load	
		Figure 10.1-1

See 1-NHY-202115

See PID-1-HG-B20888

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Hydrogen Gas System	
		Figure 10.2-1

See PID-1-MS-B20579

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Main Steam System Overview	
		Figure 10.3-1

See PID-1-MS-B20580

See PID-1-MS-B20581

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Main Steam System - Main Steam Headers Detail	
		Figure 10.3-2 Sh. 2 of 2

See PID-1-MS-B20582

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Main Steam System - Emergency Feedwater Pump Supply Detail	
		Figure 10.3-3

See PID-1-MS-B20583

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Main Steam System - Main Steam Manifold and HP Turbine Piping Detail	
		Figure 10.3-4

See PID-1-MS-B20584

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Main Steam System - Low Pressure Steam Piping Detail	
		Figure 10.3-5

See PID-1-MS-B20585

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Main Steam System - High Pressure Steam Piping Detail	
		Figure 10.3-6

See PID-1-MS-B20586

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Main Steam System - Steam Dump Piping Detail	
		Figure 10.3-7

See PID-1-MS-B20587

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Main Steam System - Main Steam Drains Detail	
		Figure 10.3-8

See PID-1-MS-B20588

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Main Steam System - Miscellaneous Vents and Drains Detail	
		Figure 10.3-9

See PID-1-EX-B20324

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Extraction Steam Overview	
		Figure 10.3-10

See PID-1-EX-B20328

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Extraction Steam Main Turbine and Steam Piping Drains (MSD)	
		Figure 10.3-11

See PID-1-AR-B20744

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Condenser Air Evacuation System P&ID
	Figure 10.4-1

See PID-1-SSS-B20759

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Turbine Steam Seal System Detail	
		Figure 10.4-2

See PID-1-CW-B20672

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Circulating Water Overview	
		Figure 10.4-3

See PID-1-CW-B20673

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Circulating Water Detail	
		Figure 10.4-4 Sh. 1 of 2

See PID-1-CW-B20674

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Circulating Water Detail	
		Figure 10.4-4 Sh. 2 of 2

See PID-1-CL-B20680

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Chlorination System Overview	
		Figure 10.4-5

See PID-1-CO-B20420

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Condensate System Overview	
		Figure 10.4-6

See PID-1-CO-B20422

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Condensate System Detail [5 Sheets]	
		Figure 10.4-7 Sh. 1 of 5

See PID-1-CO-B20423

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Condensate System Detail [5 Sheets]	
		Figure 10.4-7 Sh. 2 of 5

See PID-1-CO-B20424

See PID-1-CO-B20425

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Condensate System Detail [5 Sheets]	
		Figure 10.4-7 Sh. 4 of 5

See PID-1-CO-B20426

See PID-1-FW-B20684

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Feedwater System Overview	
		Figure 10.4-8

See PID-1-FW-B20686

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Feedwater System Detail	
		Figure 10.4-9 Sh. 1 of 2

See PID-1-FW-B20687

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Feedwater System Detail	
		Figure 10.4-9 Sh. 2 of 2

See PID-1-SB-B20625

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Steam Generator Blowdown Overview	
		Figure 10.4-10

See PID-1-SB-B20626

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Steam Generator Blowdown (Blowdown Flash) Detail
	Figure 10.4-11

See PID-1-SB-B20629

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Steam Generator Blowdown Blowdown Evaporation Detail [3 Sheets]	
		Figure 10.4-12 Sh. 1 of 3

See PID-1-SB-B20630

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Steam Generator Blowdown Blowdown Evaporation Detail [3 Sheets]	
		Figure 10.4-12 Sh. 2 of 3

See PID-1-SB-B20631

See PID-1-SB-B20627

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Steam Generator Blowdown Blowdown Recovery Detail	
		Figure 10.4-13 Sh. 1 of 2

See PID-1-SB-B20628

See PID-1-SCC-B20072

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Secondary Component Cooling Water Overview	
		Figure 10.4-14

See PID-1-AB-B20012

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Auxiliary Boiler Overview	
		Figure 10.4-15

See PID-1-AS-B20564

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Auxiliary Steam Overview	
		Figure 10.4-16

See PID-1-ASC-B20900

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Auxiliary Steam Condensate System Overview,	
		Figure 10.4-17 Sh. 1 of 2

See PID-1-ASC-B20901

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Auxiliary Steam Condensate System Overview,	
		Figure 10.4-17 Sh. 2 of 2

See PID-1-CPS-B20150

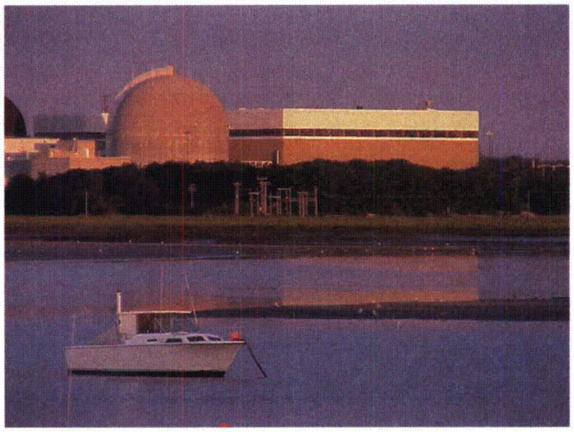
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Condensate Polishing System Overview	
		Figure 10.4-18

See PID-1-CPS-B20152

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Condensate Polishing System Condensate Polishing Pump	
		Figure 10.4-19

SEABROOK STATION
UPDATED FINAL SAFETY ANALYSIS REPORT

CHAPTER 11
RADIOACTIVE WASTE MANAGEMENT



11.1 SOURCE TERMS

The ultimate source of radioactivity for all plant systems and radioactive releases from the site is the reactor core. The fission product inventory in the reactor core is presented in Chapter 15. Additional sources of radioactivity considered in the design of the Radioactive Waste Management Systems and for the determination of plant releases are corrosion and activation products. These sources of radioactivity are generated by the unavoidable interaction of neutrons from the reactor core with corrosion particles in the coolant system and with air molecules in the Reactor Containment. Quantitative values for these radionuclides are taken from reactor operating experience and are considered as part of the overall source term.

11.1.1 Design Basis

The concept of radioactive waste management involves the examination of all potential pathways of radioactive release to the environment and the provision of appropriate processing and treatment equipment to ensure that release of radioactivity to the environment is kept as low as is reasonably achievable (ALARA) in compliance with Section 50.34a of 10 CFR Part 50. Appendix I to 10 CFR Part 50 provides numerical guides for those design objectives to meet the criterion ALARA. The plant operates within the limits of radiation levels set forth in 10 CFR Part 20.

On May 21, 1991, a complete revision to 10 CFR 20 was issued. Several design bases reference 10 CFR 20 and specific terms or parts of 10 CFR 20. Design bases information provides a historical perspective of the information used to formulate a particular design. References to 10 CFR 20 when used in a historical or design bases context have not been changed to reflect the revised 10 CFR 20.

The transport of radioactivity from the primary Reactor Coolant System to various parts of the plant during normal operation is traced and evaluated in order to determine the performance of each process interposed between the source of radioactivity and the subsequent pathways to the environment.

There are three radioactive waste treatment systems: the liquid radwaste systems (Liquid Waste System, Boron Recovery System and Steam Generator Blowdown System), the Radioactive Gas Waste System, and the Solid Radwaste Handling System. Potentially radioactive liquids, gases and solids are collected and processed according to physical and chemical properties and radioactive concentrations as deemed necessary. Care is taken in design to minimize the mechanical leakage paths in these systems in order to limit unprocessed leakage.

The calculation of potential offsite doses considers all possible pathways to the environment, including estimated leakage of radioactive process streams into the Containment, and into the auxiliary buildings, which are eventually released to the environment through building ventilation systems. The radioactive source terms used for design of shielding and building ventilation systems have been closely reviewed in conjunction with effluent releases to the environment to ensure that these source terms are not in excess of the acceptable release guidelines set forth in 10 CFR Part 50, Appendix I.

11.1.1.1 Failed Fuel[1]

The concept of failed fuel determines what fraction (or percentage) of the reactor core fission product inventory is assumed to be released to and contained within the Primary Coolant System. Three sets of source terms (reactor coolant radionuclide concentrations) have been determined, using three different values for the assumed failed fuel fraction:

a. The coolant radionuclide concentrations based on a 1 percent failed fuel fraction to establish design basis values to be used for systems and shielding calculations

b. Reactor coolant radionuclide concentrations based on 0.25 percent failed fuel for design of the plant ventilation systems

c. Reactor coolant radionuclide concentrations based on 0.12 percent failed fuel to conform to the models and procedures described in Regulatory Guide 1.112 "Calculation of Releases of Radioactive Materials in Liquid and Gaseous Effluents from Pressurized Water Reactors (PWRs)." The inventories calculated in this manner represent "expected basis" activities and will be used for the evaluation of environmental impacts during normal operation.

11.1.1.2 Fission, Corrosion and Activation Product Activities

a. Activity in the Primary Coolant System

"Design basis" (1 percent and 0.25 percent failed fuel values) activity inventories in the reactor coolant are provided in Table 11.1-1. These inventories have been used for designing the waste management and ventilation systems components.

"Expected basis" (0.12 percent failed fuel values) activity inventories are also provided in Table 11.1-1.

[1] With the exception of the Primary Coolant System source term based on 1% failed fuel, the analysis and parameters described in this section are historical and are the basis of the 10CFR Part 50 Appendix I program. It is noted that for an operating plant, the actual performance and operation of installed equipment, the reporting of offsite releases and doses, and compliance with the regulatory limits of 10CFR50 Appendix I, 10CFR20, and 40CFR190 are controlled by the Offsite Dose Calculation Manual.

b. Radioactivity in the Secondary Coolant System

Primary-to-secondary leakage of reactor coolant through the steam generator tubes will result in the contamination of steam generator secondary side liquid with radionuclides from the reactor coolant. The secondary side radioactivity levels are dependent on the reactor coolant radionuclide concentration, the primary-to-secondary leak rate, and the rate of steam generator secondary side blowdown. The anticipated values for these parameters for design and normal operating conditions are as follows:

1. Primary coolant radionuclide concentrations as given in Table 11.1-1 (0.12 percent values)
2. Primary-to-secondary leak rate of 100 lbm per day (total for 4 steam generators)
3. Steam generator blowdown rate of 75 gpm (total)
4. The carryover due to mechanical entrainment in the steam generators is 0.1 percent for all particulates except halogens, and 1 percent for halogens
5. Volatile chemistry is used with condensate demineralization available during startup and condenser leakage events.

These conditions combine to generate equilibrium steam generator secondary side activity concentrations as given in Table 11.1-4 and Table 11.1-5. Both design level activities (0.25 percent failed fuel) and expected levels of activity for normal operation (0.12 percent failed fuel) are presented.

11.1.1.3 Tritium

The liquid waste design principle of recycling all the reactor coolant from letdown and leakage that meets the reactor water chemistry specifications has potential long-term ramifications in regard to tritium levels within the plant systems and work areas. Tritium will be produced by ternary fission in the fuel, with subsequent escape from the fuel to the reactor coolant through the zircaloy cladding. Tritium will also be produced by neutron activation of boron and lithium present in the reactor coolant. Analysis shows that the tritium buildup in reactor coolant and associated liquids, as a result of recycling the intended amounts of reactor coolant, must be controlled so that containment accessibility is not unduly limited. This can be accomplished by periodic discharge or by feed and bleed of the reactor coolant.

The total tritium release through the combined liquid and vapor pathway is 0.40 Ci/yr/MWt, based on operational experience in PWRs. (NUREG-0017, "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Pressurized Water Reactors," April 1976.)

The quantity of tritium released through the liquid pathway is based on the calculated liquid release volume, assuming a tritium concentration in the released liquid of 1.0 μCi/cc. Secondary system wastes are excluded. In accordance with Regulatory Guide 1.112, only a maximum of 50 percent of the total quantity of tritium calculated to be available for release can be assumed to be released via the liquid pathway. The remainder of the tritium is assumed to be released through the vapor pathway.

11.1.2 Mathematical Models and Parameters

11.1.2.1 Design Bases Activity Levels

The parameters used in the calculation of the design level inventories are summarized in Table 11.1-2. In these calculations, the defective fuel rods are assumed to be present at the initial core loading and to be uniformly distributed throughout the core; thus, the fission product escape rate coefficients are based upon average fuel temperature. It is also assumed that all isotopes are formed directly from fission and/or from the decay of parent isotopes that are also fission products. Two isotope decay chains are assumed throughout. These products are then released into the reactor coolant through small defects in the fuel cladding. Such release is assumed to be continuous and equal to some constant fraction of the core inventory of each individual isotope. The mechanisms by which radioactive material is removed from the coolant are:

- Radioactive decay
- Material flow through a purification system
- Leakage and letdown.

The differential equations governing the numbers of atoms of the various isotopes in the core and the coolant are as follows:

In the fuel

$$\frac{dA}{dt} = P_A - \lambda_1 A$$

$$\frac{dB}{dt} = P_B + \lambda_A A - \lambda_2 B$$

In the coolant

$$\frac{d\bar{A}}{dt} = f \alpha_A A - \lambda_3 A$$

$$\frac{d\bar{B}}{dt} = f \alpha_B B + \lambda_A A - \lambda_4 B$$

where:

A = total number of atoms of parent in core

B = total number of atoms of given isotope in core

A = total number of atoms of parent in coolant

B = total number of atoms of given isotope in coolant

P_A = production rate of parent in core = $(Y_2 - Y_1)F$ (atoms/sec)

P_B = production rate of isotope in core = Y_1 F (atoms/sec)

λ_A = radioactive decay constant of parent (sec^{-1})

λ_B = radioactive decay constant of isotope (sec^{-1})

Y_1 = direct fission yield of isotope in question

Y_2-Y_1 = direct fission yield of parent of give isotope

σ = thermal neutron absorption cross section of given isotope (cm^2)

$\varnothing$ = average thermal neutron flux in core (n/cm^2 - sec)

α_A = fuel escape coefficient of parent (sec^{-1})

α_B = fuel escape coefficient of isotope (sec^{-1})

β_A = removal rate of parent from coolant (fractions/sec)

β_B = removal rate of isotope from coolant (fractions/sec)

f = failed fuel fraction

F = fissions per second in the core = $3.121 \times 10^{10} \times (P \times 10^6)$

$\lambda_1 = \lambda_A$

$\lambda_2 = \lambda_B + \varnothing\sigma$

$\lambda_3 = \lambda_A + \beta_A$

$\lambda_4 = \lambda_B + \beta_B$

The activities of the various isotopes in the fuel and coolant are:

$D_f(t) = \lambda_B B(t)/3.7 \times 10^{10}$ (curies)

$D_c(t) = \lambda_B B(t)/(V_c \times 3.7 \times 10^{10})$ (curies/m^3)

where V_c is the coolant volume (m^3)

11.1.2.2 Expected Bases Activity Levels

Expected bases activity levels for primary and secondary coolant water are calculated based on Regulatory Guide 1.112, NUREG-0017 and the USNRC PWR GALE code. Input parameters used to derive coolant activity levels and plant liquid and gaseous releases are given in Appendix 11A, and are summarized in Table 11.1-3.

11.1.3 Source Terms for Shielding and Component Failures

See Section 12.2.

11.1.4 Radioactive Leakage and Estimated Contributions

11.1.4.1 Sources of Leakage

Westinghouse has surveyed various PWR facilities to identify design and operating problems influencing nonrecyclable reactor coolant leakage, and hence the load on the Waste Disposal System. Leakage sources have been identified in connection with pump shaft seals and valve stem leakage.

11.1.4.2 Estimated Contribution to Total

As discussed in Appendix 11A, Section 2d, the equipment leakage rate of primary coolant is estimated to be 300 gallons per day (0.21 gpm). This source of leakage is assumed to occur within the Primary Containment and is collected in the primary drain tank (PDT) before processing by the PDT Degassifier and Boron Recovery System.

Additional sources of recyclable primary grade leakage are expected to occur outside the Containment (within the Primary Auxiliary Building), and will be collected in the aerated waste recovery tank. The quantity of leakage assumed is 160 lbs. of primary coolant per day.

Leakage rates from the Reactor Coolant System and other fluid systems containing radioactivity into individual cubicles and areas that may require access are discussed in Section 12.2.

11.1.5 Effluent Releases from Other than Radioactive Waste Systems

This section discusses the effluent releases to the environment from other than radioactive waste systems during normal operations. Estimated releases from anticipated operational occurrences are discussed in Subsections 11.2.3 and 11.3.3.

11.1.5.1 Gaseous Effluent Releases

The Radioactive Gaseous Waste System has been designed to process the gaseous wastes generated in the plant, and are discussed in Section 11.3. Normally anticipated effluent releases from the system will be on a controlled basis. However, any leakage of primary coolant or the process stream either in the Containment or in the auxiliary buildings is collected in the buildings and vented through filtration systems to the environment. Any steam/water leakages in the Turbine Building are directly vented to the environment. The noncondensable gases will be also discharged through the main condenser vacuum system exhaust.

a. <u>Containment Purges</u>

Four purges are expected annually for shutdown, annual fuel loading, and planned maintenance. The duration of shutdown and pre-entry containment purges is expected to be 24 hours per purge. In addition, an online purge system is available for use during power operation. This system will be used to reduce containment airborne activity levels in anticipation of containment entry by plant personnel. To evaluate airborne releases from containment venting, a continuous 1000 scfm online purge rate is used.

b. <u>Steam Generator Blowdown System</u>

Steam Generator Blowdown System waste liquids are processed either through the blowdown demineralizer subsystem or the Liquid Waste Processing System. The normal method of processing blowdown fluids is through the blowdown demineralizer subsystem, which returns the liquid to the main condenser. This includes conditions of minor primary-to-secondary leakage. If necessary, this liquid may also be processed through the installed vendor system (WL-SKD-135). Blowdown flash tank bottoms would be transferred to the floor drains tanks and then through this skid for treatment prior to discharge. The flash tank steam can be either returned to the No. 3 feedwater heater, or processed through the flash steam condenser cooler to the waste test tanks, prior to discharge. Noncondensable gases contained within the Secondary Coolant System are released via the main condenser vent system and as such are considered as part of the gaseous releases source term for the main condenser evacuation system.

c. <u>Primary Auxiliary Building Ventilation</u>

The activity from leakage into the PAB is assumed to be released directly to the environment through the filtration system. The partition factors are 0.0075 for iodines and 1.0 for noble gases. The leakages are mostly equipment leakages, and take place at valves, flanges and pump seals. It is assumed that there is an inleakage of 160 lbs. of primary coolant per day into the Primary Auxiliary Building.

d. <u>Turbine Building Ventilation</u>

One hundred percent of the steam leakage is assumed to be released to the environment. Steam leakage from the Secondary Coolant System to the Turbine Building is assumed to be 1700 lbs. per hour.

e. Main Condenser Vacuum Pump

An iodine partition factor of 0.15 is used for volatile iodine species in the main condenser. All noble gases and a small part of the iodines in the condensing steam are considered to be released through the Primary Auxiliary Building charcoal filtration system. The main condenser vacuum pump is assumed to operate at 2 inches Hg and 60 scfm. Annual releases from the main condenser vacuum pump are calculated using an equilibrium model.

11.1.5.2 Liquid Effluent Releases

The Radioactive Liquid Waste System is designed to process liquid wastes generated in the plant, and is discussed in Section 11.2. In this section, the liquid effluent releases for tritium control in the Primary Coolant System and the condensate leakage in the Turbine Building are discussed.

a. Processed Reactor Coolant Releases for Tritium Control

It is assumed that 200,000 gallons of reactor coolant is released per year for tritium control. This liquid is treated by the Boron Recovery System prior to release. The process and discharge period is 120 hours.

b. Unprocessed Liquid Waste from the Turbine Building Floor Drain Sump and Water Treatment Liquid Effluent that includes Condensate Polishing System

It is assumed that the Turbine Building floor drains and Water Treatment Liquid Effluent that includes Condensate Polishing System will collect leakage/discharge at 7550 gallons per day at main steam activity. No credit for decay is taken in these calculations.

11.1.6 Operations Resulting in Radioactive Releases to the Environment

Operational activities resulting in radioactive liquid and gaseous releases to the environment are discussed in Subsections 11.2.3 and 11.3.3, respectively. Expected and design conditions are discussed, including anticipated releases and projected doses.

11.1.7 Operating Reactor Experience

Operating reactor experience has been incorporated into the development of Seabrook's source terms, by using the methodology described in NRC Regulatory Guide 1.112, NUREG-0017 and the PWR GALE computer program. These regulatory guides and computer codes are based on standardized primary and secondary coolant activity derived from American Nuclear Society (ANS) 18.1 Working Group recommendations. These recommendations, in turn, are based on several years of reactor operating experience.

11.1.7.1 Fuel Performance

Seabrook's "realistic" source terms have been developed using the ANS standard referenced above, and as such represent the radionuclide inventories and releases which are expected with 0.12 percent fuel cladding defects, and the escape and release rate coefficients given in Table 11.1-2. A value of 0.12 percent is the weighted average of the fuel performance data, based on the burnup rates, for operational experience with zircaloy-clad fuels in PWRs collected for the operating years of January 1970 to July 1973.

11.1.7.2 Leakage Sources and Pathways

Leakage sources and pathways for liquid and gaseous radioactive releases to the environment are discussed in Subsections 11.2.3 and 11.3.3, respectively.

11.1.7.3 Releases of Radioactivity to the Environment

Releases of liquid and gaseous radioactivity to the environment from planned and anticipated operational occurrences are presented in Subsections 11.2.3 and 11.3.3, respectively.

11.2 LIQUID WASTE SYSTEM

This section describes the manner in which radioactive liquid wastes are collected, processed and directed for either reuse or release from the site. The equipment involved and its operation is described, the amount of radioactivity in liquid effluents is estimated and, finally, the radiological environmental impact from such releases is evaluated.

11.2.1 Design Bases

The Liquid Waste System (WL) is nonnuclear safety class (NNS) and nonseismic Category I, in accordance with Regulatory Guides 1.26 and 1.29.

The Liquid Waste System is designed to meet applicable requirements specified in 10 CFR, Parts 20 and 50, as follows:

a. Provide a central collection point for radioactive liquid waste. This includes approximately 1200 gallons per week of reactor grade and nonreactor grade leakage from various systems (see Subsection 9.3.3) and approximately 400 gallons per week of floor drainage from area wash down.

b. Provide preliminary processing through the use of a strainer and filters.

c. Concentrate nonvolatile and, to some extent, volatile radioactive liquid contaminants, through evaporation, with a minimum decontamination factor (D.F. = Ratio of specific activity in the bottoms and distillate) of 10^4, at a bottoms concentration of 12 percent by weight.

d. Concentrate the residual contaminants (bottoms) up to 12 percent total dissolved solids (TDS) for transfer to the Waste Solidification System (Section 11.4).

e. Produce up to 25 gpm of distillate from the evaporator/condenser. The distillate is demineralized (if necessary) and tested in the WL waste test tank before disposal offsite.

f. Maintain, during normal operation, the radioactivity content of liquid effluents from the Seabrook site within the concentration limits expressed in 10 CFR 20, Appendix B, Table II, Column 2, on an instantaneous release basis and on an annual average release basis to maintain the radioactive liquid effluents so that the dose guidelines expressed in the Appendix I to 10 CFR 50 are not exceeded.

g. Provide processing equipment and capacity sufficient to maintain radioactivity in liquid effluents within the applicable flexibility provisions of Appendix I to 10 CFR 50 during anticipated operational occurrences.

11.2.2 System Description

11.2.2.1 Operation

The primary means of processing liquid radioactive water is through a vendor-supplied demineralizer system. Other systems or components may be used if necessary.

The Liquid Waste System, Figure 11.2-1, Figure 11.2-2, Figure 11.2-3 and Figure 11.2-4, is located in the Waste Processing Building, a seismic Category I structure. Backup processing capability is available from the Boron Recovery System (BRS) evaporator BRS-EV-3A (Subsection 9.3.5). Collection of the waste liquid is continual as the waste is generated. Processing is in batch mode. No feed occurs into the evaporator when it is discharging the concentrated liquid. The system can be used during normal operation, plant startup, shutdown, and refueling operations as long as electrical power and component cooling water systems do not fail. No emergency power or cooling water is available to this system. In the case of unavailability of main plant steam during shutdown, a limited amount of steam is available from the auxiliary boiler, depending upon other uses of steam at that time. In the event that steam and cooling water are not available during a shutdown, the waste evaporators will not be run. Residual waste liquid will be processed via a skid-mounted waste liquid processing system or transferred to the floor drain tanks for future processing.

The major system functions are discussed below.

a. Liquid Waste Storage

The following sources pass through a strainer and are stored in the floor drain tanks for further processing:

1. Liquid from the chemical drain treatment tanks

2. Effluent from the Resin Sluicing System

3. Liquid from the Steam Generator Blowdown System when that system requires additional processing capacity

4. Liquid from boron waste storage tanks when that liquid is unacceptable for reuse in the reactor plant

5. Recycled effluent from liquid waste evaporation and testing and demineralization when reprocessing is required

6. Demineralized water for system startup and flushing operations

7. Waste liquid from the spent fuel pool skimmer pump

8. All sumps in contaminated plant areas, and the Administration and Service Building sump and the RCA walkway B&C sumps.

Prior to further processing, the pH of the liquid can be adjusted with the liquid waste chemical addition pump by recirculating the tank contents with the floor drain tank pump.

From the floor drain tanks, liquid is transferred by one of the two floor drain tank pumps to one of the following:

1. The waste evaporator for processing

2. The waste test tanks for direct discharge offsite, if the quality and radioactivity levels are within design limits

3. The boron recovery system evaporator (Subsection 9.3.5.2)

4. The waste feed tanks of the Waste Solidification System (Section 11.4)

5. The skid-mounted Waste Liquid Processing System

6. The boron waste storage tanks if additional waste water storage capacity is required prior to processing.

Overflow and recirculation lines are provided on the floor drain tanks. The operation is manual, except for the automatic shutdown of the floor drain tank pump on low liquid level in the tank.

b. <u>Evaporation</u>

The evaporation equipment of the Liquid Waste System is identical to the Boron Recovery System (see Subsection 9.3.5.2) except that only one liquid waste evaporator is employed, versus two for the BRS, and the concentration in the liquid waste evaporator is generally up to 12 percent TDS, as mentioned in the design bases.

c. Testing and Demineralization

The liquid waste evaporator distillate and the skid-mounted Waste Liquid Processing System effluent are the normal sources of liquid to the testing and demineralization subsystem. Other sources of liquid which enter the waste test tanks are listed below.

1. Liquid directly from the floor drain tanks, when that liquid does not require processing in the evaporator
2. Distillate from the boron recovery evaporator, when that evaporator is substituting for the waste evaporator
3. Flashed steam or bottoms from the steam generator blowdown flash tank and flash steam condensers, when that system must discharge liquid off site (see Subsection 10.4.8)
4. Liquid from chemical drain liquid tanks, when that liquid does not require processing in the evaporator.
5. Liquid waste from the RWST and SFP clean up skid 1-CBS-SKD-161.

Liquid collected in the waste test tanks is transferred by one of two waste test tank pumps to the Circulating Water System intake and discharge transition structures for final disposal. Radiation levels and flow rates are also monitored on this line. If purification is required prior to discharge, the liquid is circulated through the waste demineralizer and filter. If reprocessing is required, the waste test tank contents are pumped back to the floor drain tanks, the BRS recovery test tanks, or the boron waste storage tanks. Overflow and recirculation lines are provided on the waste test tanks.

Liquid from the BRS testing and demineralization subsystem (Subsection 9.3.5.2d) and blowdown from the steam generator blowdown system flash tank (Subsection 10.4.8) can be transferred directly to the Circulating Water System intake and discharge transition structures for final disposal when water quality permits. Wide-range flow control is achieved by two parallel control valves in 3" and ¾" lines. One valve always remains closed when the other is open. The flow of filtered demineralized water which is discharged to the Circulating Water System is recorded as well as totalized at the Waste Management System (WMS) control panel.

Operation is essentially manual, with only two protective functions being automatic, i.e., securing of the waste test tank pumps on low test tank level, and termination of offsite discharge on high radiation levels.

11.2.2.2 Component Description

The detailed data is given in Table 11.2-1.

11.2.3 Radioactive Liquid Release

The amount of radioactivity projected to be released in liquid effluents from normal operation, including anticipated operational occurrences, is described below. The radiological impact from such releases is evaluated based on the dose models in Regulatory Guide 1.109 (Revision 1).

The analysis and parameters described in this section are historical and are the basis of the facility's 10 CFR 50 Appendix I analysis, utilizing the assumptions and methodology of NUREG-0017 and a core power level of 3654 MWt. For the licensed core power level, the core power level for analysis purposes is 3659 MWt. Since 3659 MWt represents only an approximate 0.1% increase above 3654 MWt, use of the core power level of 3659 MWt in the Appendix I analysis discussed herein will have an insignificant impact on radiological releases.

Continued compliance with the annual dose limits to an individual in an unrestricted area set by 10 CFR 50 Appendix I and 40 CFR 190, resulting from gaseous and liquid effluents released to the environment following operation at the licensed core power level was also demonstrated by utilizing five (5) years (1998-2002) of effluent/dose impact data and using scaling factors to estimate the impact of operation at the licensed core power level.

It is noted that for an operating plant, the actual performance and operation of installed equipment, the reporting of actual offsite releases and doses, and compliance with the regulatory limits of 10 CFR 50 Appendix I, 10 CFR 20, and 40 CFR 190, are controlled by the Offsite Dose Calculation Manual.

11.2.3.1 Normal Operation Release

The sources of radioactive wastes that are to be released are as follows:

- Boron Recovery System discharges for tritium control
- Nonrecyclable liquid releases from the Liquid Waste System
- Secondary system condensate leakage
- Nonrecyclable liquid releases from the steam generator blowdown waste holdup sump

The list of potential sources of liquid to be discharged is reduced to the above because of the processing system design principle to segregate, process, and recycle as much of the liquid extracted from the Reactor Coolant System as possible. The systems provided to carry out this design principle are the Boron Recovery System (Subsection 9.3.5) and the Liquid Waste System, described above. These descriptions illustrate the manner in which the collecting and handling of liquid letdown and leakage from the Reactor Coolant System are segregated from nonrecyclable liquids and sent, after processing, to the reactor makeup water storage tanks for reuse within the Reactor Coolant System.

a. Release Assumptions

The main assumptions and parameters used in estimating the magnitude of radioactive liquid releases are as follows:

1. The radionuclides and their concentrations within the Reactor Coolant System are as listed in Table 11.1-1 under the heading of "0.12 percent cladding defects."

2. The radionuclides and their concentrations within the secondary side of the steam generators are as listed in Table 11.1-4. The feed and condensate system activities are equivalent to the steam activities, excluding noble gases.

3. The decontamination factors (DFs) within the Boron Recovery System and the Liquid Waste System are given in Appendix 11A.

4. The times of radioactive decay between collection, processing and discharge are listed in Appendix 11A for each stream of liquid waste.

5. The reactor is assumed to be operating at 3654 MWt, with an 80 percent capacity factor.

b. Releases

1. Releases from Boron Recovery System

As described in Subsection 11.1.1.3, tritium control considerations anticipate the need for discharging reactor coolant letdown after processing by the Boron Recovery System. The expected volume required for this measure is 200,000 gallons per year. With the input liquid containing radionuclides at Table 11.1-1 values (0.12 percent clad defect), the processing DFs and the radioactive decay times in Appendix 11A, the annual release from this source is 0.033 Ci/year, except tritium. This release is shown by radionuclide in Table 11.2-2.

2. Nonrecyclable Releases from Liquid Waste System

The estimated average volumetric generation rates of nonrecyclable primary system leakage are 40 gallons/day inside the Containment at primary coolant activity (PCA), and 200 gallons/day in the Auxiliary Building at 0.1 PCA. This liquid is collected in the floor drain tanks, which are at the head end of the floor drain portion of the Liquid Waste System. It is also estimated that there are 400 gpd of liquid waste from laboratory drains at 0.002 PCA, 15 gpd from sampling drains at 1.0 PCA, and 700 gpd from miscellaneous waste at 0.01 PCA released into the Liquid Waste System. (There will be no liquid waste generated from laundry operations.) This liquid is collected in the chemical drain treatment tanks (refer to UFSAR Subsection 9.3.3.2). The total input rate to the floor drain tank is less than 1355 gallons/day with the effective composite activity concentration of 0.061 PCA. With the processing DFs and radioactive decay times in Appendix 11A, the annual release from this source is less than 0.035 Ci/year, except tritium. This release is shown by radionuclide in Table 11.2-3.

3. Steam Generator Blowdown

With reactor coolant radionuclide concentrations as listed in Table 11.1-1 and an estimated average leakage of 100 lbs/day of reactor coolant through the steam generator tubes, equilibrium secondary side steam generator radionuclide concentrations are calculated as listed in Table 11.1-4. The steam generator secondary side blowdown rate associated with this leakage is 75 gpm, total from all four steam generators.

Blowdown from the four steam generators is processed by the blowdown flash tank, where approximately 30 percent of the blowdown volume is flashed to steam. This steam is normally routed to the No. 3 feedwater heater or to the main condenser. With no primary-to-secondary leakage, steam may be exhausted to the atmosphere if the heater and condenser are not available. The vent gases from the flash steam condenser are processed before discharge. Radioactive steam is not processed directly to the atmosphere. In the presence of a primary-to-secondary leak, flash tank steam may be sent to the No. 3 feedwater heaters or processed through the flash tank distillate coolers to the waste test tanks, prior to discharge.

The remaining volume of blowdown liquid (70 percent) can be released directly to the environment via the plant Circulating and Service Water System when no primary-to-secondary leakage exists or processed via the blowdown demineralizer subsystem. With significant primary-to-secondary contamination of the secondary side water, the primary method to process radioactive secondary liquid from the steam generators is to direct steam blowdown flash tank bottoms cooler discharge to the floor drain tanks. If no secondary pressure is available, the steam blowdown and wet lay-up pumps can be used. From the floor drain tanks, processing through the installed vendor system (WL-SKD-135) to the waste test tanks may be performed (reference Subsection 11.2.2.1). Additional methods that could be used, with station management approval and planning, include the liquid volume within the flash tank being processed by the Blowdown Evaporator System (Subsection 10.4.8). Two evaporators in parallel are available to process a maximum of 50 gpm of blowdown liquid. Distillate from the evaporators is condensed and directed to the waste test tanks of the Liquid Waste System. Further processing is available within the Liquid Waste System, if required, prior to discharge to the environment via the Plant Circulating Water System. Bottoms from the blowdown evaporators are routinely

released to the Solid Waste Processing System for processing and shipment offsite. No credit for collection and processing decay times has been assumed to calculate the liquid releases from this pathway. With the DFs presented in Appendix 11A for the installed vendor system, the annual release from this source is 0.02 Ci/year. This release by radionuclide is presented in Table 11.2-4.

4. Secondary System Condensate Leakage

   The estimated average liquid leakage rate of the secondary system is 7200 gallons/day. The leakage is assumed from liquid sources at main steam activity. The concentrations of the main steam activity are listed in Table 11.1-4. This liquid is collected in the Turbine Building floor drain and then is discharged from the plant unprocessed, which results in the annual release of 0.00658 Ci/year, except tritium. This release is shown by radionuclide in Table 11.2-5.

5. Summary of Radioactive Liquid Release From Normal Operation

   The total estimated radioactivity to be released due to the generation and release of the liquid streams described above is shown in Table 11.2-6. The total annual release of 0.08 Ci except tritium and 730 Ci of tritium are the expected discharge levels due to normal operation. The tritium releases are discussed in Section 11.1.

11.2.3.2 Releases from Anticipated Operational Occurrences and Design Basis Fuel Leakage

The additional unplanned liquid release due to anticipated operational occurrences is estimated to be 0.15 Ci/year based on reactor operating data over a 2.5 year period, January 1973 through June 1975, representing 102 reactor-years of operation (NUREG-0017). These releases are assumed to have the same isotopic distributions for the calculated source term of the liquid wastes. The annual release from the anticipated operational occurrences is shown by radionuclide in Table 11.2-7.

Table 11.2-8 shows the total annual release by radionuclide from normal operation, including anticipated operational occurrences. The discharge concentrations are compared with (MPC)w, the concentration limits of 10 CFR Part 20, Appendix B, Table II, Column 2. Table 11.2-9 shows the total annual release by radionuclide from design basis fuel leakage (i.e., 1 percent failed fuel).

11.2.3.3 Release Points

The release routes of radioactive liquids generated in operation are shown in simplified flow sheet form in Figure 11.2-5. Liquid processed for discharge by the Boron Recovery System ultimately accumulates in the waste or recovery test tanks, located adjacent to the Waste Processing Building, as shown in Figure 1.2-22 and Figure 1.2-25. After sampling for radioactivity analysis, the liquid is discharged from the waste or recovery test tank through a process radiation monitor to the Circulating Water System. The tie-in point with the Circulating Water System is as shown in Figure 11.2-4.

Nonrecyclable reactor coolant leakage is collected by the drain system within the Containment and Primary Auxiliary Building and is directed to the floor drain tank of the Liquid Waste Processing System. This tank is located within the Waste Processing Building as shown in Figure 1.2-29. This liquid is processed and directed to either the waste or recovery test tanks. The liquid is then sampled and released as described above for boron recovery system releases.

Miscellaneous liquid wastes (decontamination water, laboratory, drains, etc.) are collected in the chemical drain treatment tank where they can be processed and directed to the waste or recovery test tanks for final sampling prior to discharge to the Circulating Water System.

Steam generator secondary side blowdown is directed to the flash tank. If steam generator blowdown activity is low, liquid from the flash tank may be transferred directly to the waste test tank or to the Circulating Water System. Otherwise, liquid from the flash tank is directed to the blowdown demineralizers or the installed vendor system for treatment.

Secondary system condensate leakage is collected in the Turbine Building sumps. From here the liquid is directed to an oil separator and transferred to the circulating water system discharge. A radiation monitor is located on the sump effluent line which automatically isolates the sump at a predetermined radioactive concentration.

All these waste liquids, once released to the Circulating Water System, experience the same release path to the Atlantic Ocean. The radioactive liquid wastes then reach the environment via the circulating water discharge line.

11.2.3.4 Dilution Factors

The discharge route of radioactive liquid to the environment, as described in the above section, provides onsite dilution by the flow of the Circulating Water System, conservatively assumed to be 390,000 gpm (for a discussion of CWS operation, see Subsection 10.4.5). With the assumption of 80 percent operating capacity of the Circulating Water System, this flow dilutes the normal liquid radioactivity releases from 0.24 Ci/year except tritium and 730 Ci/year of tritium, to 3.9×10^{-10} μCi/ml except tritium and 1.2×10^{-6} μCi/ml of tritium.

Further dilution of the circulating water discharge plume will occur after leaving the discharge pipe. A near-field dilution factor of 8 was used for calculating the estimated doses reported in the next section. This dilution factor estimate is based on a multi-port, deep water discharge concept and is conservatively calculated for the tidal cycle surface area above the discharge pipe.

The information above provides assurance that the original plant design would meet the requirements of 10 CFR Part 20 and 10 CFR Part 50. The Offsite Dose Calculation Manual (ODCM) contains actual plant data that is used for plant releases to comply with 10 CFR Part 20.

11.2.3.5 Estimated Doses

Radionuclides in liquid effluents from the site pose a potential environmental radiation source to certain individuals or segments of the general public. In general, many possible exposure pathways exist for liquid effluents, however, detailed consideration can be focused on the pathways that pose the greatest risk to public exposure - the "Critical Pathways." The critical pathways are considered to be the internal exposure by the ingestion of fin fish or other seafood harvested for the area affected by released radionuclides and the external exposures from recreational activities along the shoreline.

The radiological assessment of the ingestion pathways and the external exposure pathway is calculated based on the models, parameters, assumptions, and dose conversion factors in Regulatory Guide 1.109 (Revision 1). Table 11.2-10 shows the parameters used in the dose calculation.

The maximum annual doses from all pathways received by an individual in a particular age group due to the estimated liquid discharge radioactivities from normal operation (including anticipated operational occurrences) listed in Table 11.2-8 are shown in Table 11.2-11. Among three age groups, the highest maximum annual doses, 2.5×10^{-3} mrem/yr to the total body and 2.4×10^{-2} mrem/yr to the thyroid (the most critical organ), are small fractions of the numerical design dose objectives of Appendix I to 10 CFR 50.

11.2.4 Design Evaluation

a. General

The Liquid Waste System performs no safety function and is not required for the safe shutdown of the reactor. Accordingly, the system is designated NNS class. Also, because of the noncritical nature of this system, emergency electrical power is not provided.

All waste liquid generated within the plant is processed through the floor drainage system (Subsection 9.3.3), Boron Recovery System (Subsection 9.3.5) or Liquid Waste System, when necessary. Process discharge to the environment is only after testing of the effluent quality.

The evaporators are designed throughout with low-flow velocity and liquid vapor separators to reduce any entrainment of vapor. The evaporators are designed to yield a minimum decontamination factor of 10^4 for nonvolatiles. This will reduce the amount of radioactivity in the distillate to acceptable levels for disposal. The concentrated liquid is processed by the Solid Waste System before disposal offsite. Backup evaporator capacity from the BRS is available. All essential portions of the system are located away from any high energy lines. A negative pressure is always maintained in the vent header so that gases go into the vent header and not into the tank overflow lines. Relief valves and tank overflow lines have been provided to protect against overpressures. Should the high level point be exceeded, the tank overflow is channeled to the floor sump. Each floor sump includes one or more sump pumps which transfer the excess liquid to the floor drain tank. Pumps in the system have low level shutoffs. Filters and demineralizers have pressure indication upstream and downstream to indicate fouling.

Two sets of level instruments, one for process control and indication and the other for indication and alarm, are provided on all the essential equipment of the process. Moreover, the instrumentation and controls are located outside the boron concentrating areas to minimize radiation exposure to operating personnel.

The skid-mounted Waste Liquid Processing System is connected to permanent plant connections in the Waste Liquid System (located (-)3' elevation WPB).

Radiological consequences from postulated failures of components in the Liquid Waste System are bounded by the results of the liquid waste system failure analyses presented in Subsections 15.7.2 and 15.7.3.

b. <u>Equipment Redundancies</u>

Only one of the two floor drain tank pumps, waste test tanks, waste test tank pumps, and floor drain filters is required at a time. If the liquid waste evaporator subsystem is not available at any time, the liquid can be concentrated in one of the two boron recovery system evaporators, or it can be discharged directly to the waste feed tanks.

c. Faults of Moderate Frequency

The system is designed to handle the occurrence of equipment faults of moderate frequency such as:

1. Malfunctions in the Liquid Waste Processing System

Malfunctions in this system could include pump or valve failures or evaporator failure. Failure of a pump is acceptable since a standby is provided. Sufficient surge capacity is provided to accommodate waste in the event of evaporator failure.

2. Excessive Leakage in Reactor Coolant System Equipment

Excessive hydrogenated reactor coolant leakage can be accepted by the equipment and floor drainage system (Subsection 9.3.3) and processed by the BRS (Subsection 9.3.5).

Similarly, excessive rates of either aerated, recyclable or nonrecyclable reactor coolant leakage would be handled by the WL system by collection in the floor drain tanks (10,000 gallons each) and processing by the 25 gpm evaporator.

A design feature that enhances the surge processing capability of the BRS and WL system is the fact that the three evaporators are cross-connected to allow flexibility in evaporation capability during abnormal periods.

3. Excessive Leakage in Auxiliary System Equipment

Leakage of this type could include water from steam side leaks and fan cooler leaks inside the Containment which are collected in the containment sump and sent to the floor drain tank. Other sources could be component cooling water leaks and service water leaks. This water will enter the floor drain tank and be processed and discharged as during normal operation.

4. Overflow or Rupture of a Tank in the Liquid Waste System

Floor drains and curbing are provided around all tanks in the WL system to minimize the flow of radioactive liquid to uncontrolled areas in the event of tank rupture.

11.2.5 Testing and Inspection

Prior to startup, the WL system was tested to verify proper operation of system equipment, establish setpoints and test flow rates, temperatures, and pressures. During plant operation the system is inspected frequently to ensure proper performance and operability.

11.2.6 Instrumentation Requirements

The instrumentation and control for this system is similar to that for the BRS (Subsection 9.3.5.5). Control and remote instrumentation is located on the Waste Management System (WMS) panel in the Waste Processing Building. Pump trips, high/low levels, and temperatures are alarmed as in the BRS. High pressure differentials on filters and demineralizers are also alarmed, as is the high radiation of the final effluent from the waste test tank.

Discussed below is the instrumentation and control which differ from that of the BRS:

Radioactivity of the fluid passing through the line discharging to the Circulating Water System is continuously monitored. Upon high radiation, the discharge valves are automatically closed and an alarm is simultaneously actuated locally in the control room and at the WMS panel. Details of the radiation monitoring instrumentation are provided in Section 11.5.

Radioactivity and flow rate of the effluent are continuously monitored to assure that the effluent release is in compliance with the Technical Specification requirements.

When a predetermined high liquid level in any tank is reached, an alarm is sounded at the WMS panel alerting the operator of the potential for tank overflow. An annunciator on the waste management system control panel indicates the source of the alarm.

11.3 RADIOACTIVE GASEOUS WASTE SYSTEM

11.3.1 Design Basis

Hydrogenated fission product gases from the reactor coolant letdown stream and from the liquids collected in the primary drain tank and the reactor coolant drain tank are processed in the Radioactive Gaseous Waste System (RGWS). An iodine guard bed and a molecular sieve dryer reduce the contamination level of the gases before further processing by the carbon delay beds. The carbon delay beds provide a minimum of 60 days xenon delay and 85 hours krypton delay. Low activity aerated gas streams from the reactor plant aerated vent header (Subsection 9.3.6), and condenser vacuum pump units (Subsection 10.4.2) are filtered, monitored, and discharged to the plant unit vent.

The RGWS is designed to provide sufficient processing so that gaseous effluents are discharged to the environment at concentrations below the regulatory limits of 10 CFR 20 and within the "as low as is reasonably achievable" guidelines set forth in 10 CFR 50, Appendix I, (see Subsection 11.3.3 and Section 11.5). The RGWS also provides sufficient holdup and control of gaseous releases, as specified in 10 CFR 50, Appendix A, General Design Criterion 60. The RGWS can process a maximum surge flow of 1.2 scfm from the degasifiers, which is based on the maximum letdown flow of 120 gpm from the Reactor Coolant System to the Chemical and Volume Control System. This represents the most limiting plant operating condition for the RGWS.

The portion of the Waste Processing Building which houses the RGWS is seismic Category I.

The RGWS is designated NNS. Table 11.3-1 lists RGWS components and their design parameters. Redundant components are provided to minimize operator exposure and enhance system reliability. The RGWS is designed in conformance with Regulatory Guide 1.143.

Hydrogen concentration is monitored in cubicles containing RGWS components to detect a leak in the system. Monitoring of hydrogen concentration is not required while the RGWS is inerted with nitrogen. Subsection 15.7.1 discusses RGWS leak or failure. Dual oxygen monitors are provided to sample the process stream to monitor formation of explosive mixtures. An alarm is initiated at a predetermined setpoint prior to reaching a potentially explosive mixture. The RGWS is designed to withstand a H_2 explosion. A radiation monitor is provided on the line to the plant unit vent to detect an excessive release of radiation to the environment. An automatic isolation valve terminates flow in this line immediately upon receiving a high radiation signal. The RGWS monitors are discussed in Section 11.5. Monitoring of radioactive releases is provided in accordance with 10 CFR 50, Appendix A, General Design Criterion 64.

The Waste Processing Building Ventilation System is designed to discharge radioactive waste gases to the plant unit vent, and is discussed in Subsection 9.4.4.

The Primary Auxiliary Building Ventilation System is designed to discharge the radioactive waste gas to the atmosphere via the plant unit vent and is discussed in Subsection 9.4.3.

Shielding in separate cubicles is provided for the gas chillers, iodine guard beds, carbon delay beds, molecular sieve dryer, regeneration compressor package, hydrogen compressors, particulate filters, and hydrogen surge tank.

11.3.2 System Description

The Radioactive Gaseous Waste System, Figure 11.3-1, Figure 11.3-2, Figure 11.3-3, and Figure 11.3-4, processes noncondensable gases from the letdown degasifiers (Subsection 9.3.4), the primary drain tank degasifier (Subsection 9.3.5), the hydrogenated vent headers (Subsection 9.3.6), and the sample vessel purges (Subsection 9.3.2). Effluent gases from the operating letdown degasifiers are the major input to the RGWS. The gases from the hydrogenated vent header are processed as RGWS capacity is available. A pressure-regulating valve in the hydrogenated vent header maintains a constant pressure of 2 psig in the influent line of the RGWS, and serves to isolate this line from hydrogenated vent header pressure surges. Valves with special trim, seat and stem materials are used throughout the WG system to ensure tight shutoff and minimize leakage to the atmosphere. During normal operation, the expected influent flow rate from the degasifiers is 0.8 scfm which allows approximately 0.4 scfm for processing hydrogenated vent header gas. These influent gases consist primarily of hydrogen and water vapor with trace amounts of xenon, krypton, and iodine. Table 12.2-26 from Subsection 12.2.1 provides radioactivity content in the process stream. These gases enter the RGWS at the gas chiller compressor unit. A second chiller unit is provided for redundancy.

The chilled gas then enters the iodine guard bed which removes iodine from the gas stream (prior to further processing) to reduce radiation levels on downstream components. Since the guard beds will eventually become saturated with moisture and fission product iodines, it is anticipated that periodic replacement of the carbon in the guard beds will be necessary. In order to minimize operator exposure, "cartridge-type" carbon elements are used and the cartridges are changed remotely. Two guard beds are provided for redundancy. Each bed is capable of operating at maximum anticipated system flow.

The gas flows from the guard bed to the drying train. A three-bed dehydration system is provided. Each bed is capable of operating at maximum anticipated system flow. Normally, one bed is in operation, the second bed is heated for regeneration, and the third bed is in standby. A thermally regenerable molecular sieve is used as the drying agent. Instrumentation on the effluent side of the drying train automatically diverts the gas stream to the standby bed if an abnormally high dew point is experienced. The drying train is designed to remove vapor to -40°F dewpoint. Normal outlet dewpoint will be -80°F to -60°F.

Regeneration of a drying bed is accomplished in a closed loop system, consisting of a single-stage diaphragm compressor, an electric heater, and a purge gas condenser. The RGWS is designed with three waste gas dryers. One of them is in operation, the second is in regeneration, and the third is on standby and ready for operation. In case the purge gas condenser is inoperable, the regeneration loop will be shut off; on-stream time of the standby dryer is 96 hours which will be enough time for any maintenance work on the condenser. The regeneration compressor is also used to handle excess flow of up to 11.3 scfm when the RGWS is purged with hydrogen or nitrogen. Hydrogen from the hydrogen gas service system (see Figure 10.2-1) or nitrogen from the nitrogen gas service system (see Figure 11.3-3 and Figure 11.3-4) is directed to the hydrogenated vent header and enters the RGWS at the gas chiller. The gas stream bypasses the dryers and carbon delay beds and discharges to atmosphere via the equipment vent system.

The dried gas passes into a carbon delay bed. A total of five carbon beds are used, each containing 42.4 ft^3 (1600 lbs.) of 6x8 mesh type MBQ (or equivalent) carbon. Each carbon delay bed is approximately 3 feet in diameter by 8.5 feet in height (with 1.5 feet freeboard). Each bed provides 12 days of xenon delay and 17 hours of krypton delay based on conservatively estimated dynamic adsorption coefficients of 772.5 cc/gm. atm. for xenon and 45.4 cc/gm. atm. for krypton at a design flow rate of 1.2 scfm. Normal expected flow through the adsorbers is 0.8 scfm. The beds operate in series to provide a total delay of 60 days for xenon and 85 hours for krypton at a design flow rate of 1.2 scfm. In the event of an upset condition in the dryer, the first and second carbon beds can be operated in parallel, or the first bed can be bypassed. Bypassing of more than one bed is not permitted and would require shutoff of all input streams.

The particulate filter downstream of the carbon delay beds is designed to remove 99.97 percent of all particles 0.3 micron and larger released from the carbon delay beds. Two HEPA filters are provided for redundancy.

The waste gas stream is then compressed by a single stage diaphragm compressor. The compressor is furnished with an after-cooler to cool the discharge gas. A second compressor and after-cooler are provided for redundancy.

The compressor waste gas stream is either:

a. Returned directly to the Reactor Coolant System via the volume control tank, or the hydrogen injector,

b. Stored in the hydrogen surge tank,

c. Released to the environment via the equipment vent system, or

d. Recycled to the hydrogenated vent header as makeup gas.

The hydrogen gas compressor discharge and the hydrogen surge tank are common to a 150 psig header. Gas flows from this header through a pressure-reducing valve into 100 psig header which supplies the hydrogen injector.

Gas flow pressure is further reduced to 25 psig to supply the chemical and volume control tank. Additional gas flow pressure reduction is provided to supply gas to the hydrogenated vent header at 3 psig. Hydrogen from the hydrogen gas service system supplies makeup hydrogen to the 100 psig header if the RGWS is unable to supply the header demand, and supplies hydrogen directly to the volume control tank and hydrogen injection when RGWS hydrogen purity is below acceptable limits. A monitored line from the 150 psig header directs gas through a pressure-reducing valve and into a 3 psig header. This header then directs excess gas to the hydrogenated vent header where it is released to the atmosphere via the Primary Auxiliary Building normal ventilation cleanup exhaust unit. All releases comply with the "as low as is reasonably achievable" requirements of 10 CFR 50, Appendix I. The PAB exhaust unit is discussed in detail in Subsection 9.4.3.

The annual gaseous release due to lifting of relief valves on the evaporators or degasifiers is insignificant, since it is expected to occur less than once per year and will result in a small release to the environment for each occurrence.

Liquid drainage from the gas chillers, iodine guard beds, molecular sieve dryers, or the regeneration package is collected by a small drain pot and then pumped to the primary drain tank by the drain transfer pump.

Prior to operation, the entire system, including drain lines and the drain pot, is thoroughly purged with nitrogen to eliminate the possibility of obtaining a flammable mixture of hydrogen and oxygen.

The RGWS is taken out of operation during a "T" signal because primary component cooling water is lost to the regeneration and hydrogen gas compressors and the purge gas condenser.

The main condenser evacuation system and the turbine gland sealing system are described in Subsections 10.4.2 and 10.4.3, respectively.

11.3.2.1 Tests and Inspections

Periodic testing of the RGWS is not required as the system is in continuous operation. Inspection is performed in accordance with normal maintenance procedures. Samples can be drawn from lines downstream of the waste gas chiller dryers and downstream of the first and second carbon delay beds to check for oxygen, nitrogen and hydrogen concentrations, as desired, and for radioactivity.

11.3.2.2 Instrumentation

The following RGWS functions are instrumented and/or controlled:

- Measure concentration of oxygen in process stream
- Measure concentrations of hydrogen in cubicles
- Record and total flow of gas in main process path

- Measure the dryness at the outlet of molecular sieve dryer and hydrogen gas compressors
- Isolate the system in case of release of radioactive gases to the environment.

Further discussion on the instrumentation and control of the major components is presented below:

a. Waste Gas O_2 Monitors

The waste gas stream at the carbon delay bed inlet is sampled for trace oxygen to monitor explosive buildup. Two sensors are mounted in parallel. Oxygen concentration is indicated locally and high oxygen concentration is alarmed at the WMS panel.

b. Molecular Sieve Dryer

The three dryers are controlled from the Waste Management System (WMS) control panel. Normally the WMS is operated in auto with two drying towers alternating drying/regeneration cycles, with the third drying tower in standby. If failure occurs, the standby tower is auto-selected and it enters the cycle in place of the failed tower. This allows for a minimum of 72 hours to correct the malfunction. High humidity automatically shuts off the dryer. Humidity and radioactivity are monitored continuously at the outlet of the dryer bank. Humidity is recorded and radioactivity is indicated at the WMS panel. High humidity and radioactivity are alarmed at the WMS panel.

c. Regenerative Compressor

This compressor is a part of the dryer subsystem and maintains the flow of gas through the dryer column during regeneration. Starting of the compressor is always manually initiated from the WMS level. Deviant conditions such as low suction, lube oil and oil-gas leakage pressure automatically trip the compressor. Regenerative compressor gas flow and temperature are indicated at the WMS panel. Additionally, a trip signal from the dryer control logic stops the compressor on dryer trip.

d. Carbon Delay Bed

Temperature in each bed is recorded on a multi-point recorder at the WMS control panel. A radiation monitor is located at the outlet of the carbon delay beds. Radioactivity is indicated and high radioactivity is alarmed at the WMS panel.

e. Hydrogen Compressors

Starting of these compressors is always manual. Only one compressor runs at a time. Deviant operating conditions such as low lube oil level, generator cooler high temperature, etc., trip the compressor. The compressor discharge flow is recorded, totaled, and humidity is indicated at the WMS panel. High humidity is alarmed at the WMS panel.

f. Vent Isolation Valve

The release of radioactive waste gases is controlled by the vent isolation valve. A radiation monitor in this stream and another at the plant unit vent continuously monitor the stream for radioactivity. High radiation at the waste gas (WG) monitor automatically closes the vent isolation valve. This monitor also indicates radioactivity and alarms high radiation at the WMS panel. Flow can be re-established by the operator from the main control board. Should the WG radiation monitor malfunction for any reason, the radiation monitor at the plant unit vent alarms at the control room. The vent isolation valve is designed to fail closed on loss of air or power.

g. Monitoring of Leakage in Cubicles

Confined cubicles housing the primary drain tanks, the iodine guard beds, dryers, carbon delay beds, regenerative and hydrogen gas compressors, particulate filters and hydrogen surge tank are continuously ventilated and maintained at a slight negative pressure by the WPB normal ventilation system. These cubicles are also continuously monitored for hydrogen concentration except while the RGWS is inerted with nitrogen. The normal continuous ventilation is sufficient to maintain hydrogen concentration from normal minor leakage to a level well below 4 v/o. Should leakage increase and approach the 4 v/o concentration limit, the following will occur:

1. High hydrogen concentration will be alarmed at the WMS panel and at the main control board
2. Operator will take action to isolate and purge with nitrogen the leaking component
3. Operator will evacuate unnecessary personnel from the area.

In addition to the above, a supplementary exhaust fan will evacuate the hydrogen surge tank cubicle atmosphere when the hydrogen concentration reaches 2 v/o.

These steps will insure that the hydrogen concentrations will never exceed 4 v/o in the RGWS cubicles.

h. Hydrogen Gas Service

The makeup hydrogen flow from the hydrogen gas service system is recorded and totaled at the WMS panel.

i. Drain Transfer Subsystem

The drain transfer pump is operable from the WMS panel. In normal operation the pump cycles automatically on level switch actuation on a locally mounted drain pot. High pressure and time-delayed high level alarms are available at the WMS panel. The latter alerts the operator of the failure of the automatic start logic, so that corrective measures and/or manual starting of the pump can be initiated.

11.3.3 Radioactive Gaseous Release

Subsections 11.3.1 and 11.3.2 describe the Radioactive Gaseous Waste System. Since the system reduces fission product gas concentration in the reactor coolant during normal operation, it significantly reduces the escape of radioactive gases arising from any possible reactor coolant leakage.

Design is based on continuous operation with reactor coolant radioactivities associated with 1 percent failed fuel at rated thermal power. The estimated releases for normal operating conditions, however, are based on continuous operation with reactor coolant activities associated with 0.12 percent failed fuel at rated core thermal power. Provisions are also made to process gases from the reactor plant hydrogenated vent header. The radiological impact from such release is estimated by using the dose model in NRC Regulatory Guide 1.109 (Revision 1).

The analysis and parameters described in this section are historical and are the basis of the facility's 10 CFR 50 Appendix I analysis, utilizing the assumptions and methodology of NUREG-0017 and a core power level of 3654 MWt. For the licensed core power level, the core power level for analysis purposes is 3659 MWt. Since 3659 MWt represents only an approximate 0.1% increase above 3654 MWt, use of the core power level of 3659 MWt in the Appendix I analysis discussed herein will have an insignificant impact on radiological releases.

Continued compliance with the annual dose limits to an individual in an unrestricted area set by 10 CFR 50 Appendix I and 40 CFR 190, resulting from gaseous and liquid effluents released to the environment following operation at the licensed core power level was also demonstrated by utilizing five (5) years (1998-2002) of effluent/dose impact data and using scaling factors to estimate the impact of operation at the licensed core power level.

It is noted that for an operating plant, the actual performance and operation of installed equipment, the reporting of actual offsite releases and doses, and compliance with the regulatory limits of 10 CFR 50 Appendix I, 10 CFR 20, and 40 CFR 190, are controlled by the Offsite Dose Calculation Manual.

11.3.3.1 Normal Operation

The sources of radioactive gaseous releases (see Figure 11.3-5) are as follows:

a. Containment venting

b. Primary Auxiliary Building vent

c. Main condenser air evacuation pumps

d. Turbine Building leakage

e. Waste gas system release.

The release assumptions and parameters for each source are shown in Table 11.3-5. The estimated releases, by isotope, from each source, are shown in Table 11.3-2 for normal operation. This table is based on the source term information presented in Section 11.1, assumptions and parameters in Table 11.3-5, and an overall operating capacity factor of 80 percent.

11.3.3.2 Releases from Anticipated Operational Occurrences

The anticipated operational occurrences include the following:

a. Operation at 0.5 percent failed fuel for one year

b. Operation with 500 gallons/day of steam generator tube leakage for 90 days

c. Operations with 1 gallon/minute of hot reactor coolant leakage to the Containment for 12 days, followed by a containment purge

d. Operation with 200 gallons/day of reactor coolant leakage to the Primary Auxiliary Building for 90 days.

Each of the above anticipated operational occurrences has been evaluated by assuming that all the other parameters remain the same. The release rates from the anticipated operational occurrences are shown in Table 11.3-6, Table 11.3-7, Table 11.3-8 and Table 11.3-9.

11.3.3.3 Release Points

a. Primary Radioactive Gaseous Release Pathways

Primary gaseous release pathways for Seabrook Station are identified as releases from the following sources:

1. Waste Gas Processing System

2. Steam Generator Blowdown Processing System (included as part of main condenser releases)

3. Main Condenser Air Evacuation System

4. Primary Containment Purge Exhaust

5. Normal ventilation exhaust air from the Auxiliary and Turbine Buildings.

These sources are consistent with those presented in Reference 1. These primary release paths of potentially radioactive gaseous wastes are shown in Figure 11.3-5 and are discussed in Appendix 11A. With the exceptions of main condenser air evacuation pumps (hogging mode) and Turbine Building leakage, all significant releases to the environment are made through the primary vent stack. Releases from the main condenser air evacuation pumps are directed either to the Turbine Building vent or through the primary vent stack after filtration. Releases from the Turbine Building will be made through building ventilators.

For each primary release point, the height of release, inside dimensions of release point exit, effluent temperature and effluent exit velocity are given in Appendix 11A.

b. Secondary Potentially Radioactive Gaseous Release Pathways

The releases of radioactive materials in gaseous effluents from the following sources are considered as secondary sources and have not been included in the evaluation of station releases.

1. The releases of radioactivity materials in gaseous effluents from the following sources are calculated to be less than 1 Ci/yr of noble gases and 10^{-4} Ci/yr of iodine-131. Therefore, the following sources are considered negligible:

   (a) Steam releases due to atmospheric steam dump operation during startup and low power physics testing and

   (b) Ventilation air from buildings not covered in 11.3.3.3a.5 above (Reference 1).

2. Potentially radioactive materials in gaseous effluents from secondary pathways which are not routed to the plant unit vent include the following:

   (a) Radiochem lab fume hoods exhaust

   (b) Administration Building RCA exhaust

   (c) Thermostatically controlled room-type ventilators for equipment safeguards in PAB

   (d) Gland seal steam condenser exhaust.

Releases from these sources are anticipated to result in offsite doses which are less than 1 percent of the dose limits of 10 CFR 50, Appendix I. Sampling and/or monitoring of these potential release pathways is discussed in Updated FSAR Section 11.5.

11.3.3.4 Dilution Factors

After the radioactive gases discharge from the release points, they are diffused and further diluted by the atmosphere. The dilution factors are evaluated by using annual average meteorological data. These dilution estimates are contained in Subsection 2.3.5. The maximum annual average X/Q at the site boundary is 3.90E-07 sec/m^3 for elevated stack releases and 5.65E-06 sec/m^3 for Turbine Building releases (ground level release point).

The information above provides assurance that the original plant design would meet the requirements of 10 CFR Part 20 and 10 CFR Part 50. The Offsite Dose Calculation Manual (ODCM) contains actual plant data that is used for plant releases to comply with 10 CFR Part 20.

11.3.3.5 Estimated Doses

The estimated doses given in the following subsections are based on the gaseous releases listed in Table 11.3-2 due to normal operation. The additional releases from the anticipated operational occurrences discussed in Subsection 11.3.3.2 would increase the maximum annual doses by a factor of 5 or less.

a. Estimated External Exposure From Noble Gaseous Releases

Radioactive noble gases released to the atmosphere will result in external radiation doses to individuals in the population. From Table 11.3-2, all the noble gases will be released from the plant stacks. The radiation from these gases consists of gamma rays, which result in a dose to deep tissues or total body dose, and beta particles which result only in a surface or skin dose, and which is effectively reduced by normal clothing. Total body dose refers to the gamma ray component of the dose, and skin dose refers to the beta plus gamma doses at the outer surface of the skin and assumes no dose reduction as a result of clothing.

The gamma and beta air doses from the stack releases are evaluated at the highest dose point on the site boundary, and the total body dose and the skin dose are evaluated at the highest residential dose location based on the models in Regulatory Guide 1.109. The highest dose point on the site boundary is 2999 feet from the station, in the ESE sector, with an annual average $(X/Q)_{\text{undepleted}}$ = 3.90E-07 sec/m^3. The highest residential dose location is approximately 7867 feet from the station, in the ESE sector, with an annual average (X/Q $_{\text{undepeleted}}$ = 2.08E-07 sec/m^3.

Residential doses take credit for a 0.7 shielding and occupancy factor as suggested in Regulatory Guide 1.109. Table 11.3-3 shows the estimated maximum air doses, total body dose and skin dose based on the noble gaseous releases in Table 11.3-2.

b. Estimated Doses from Radioiodines and Particulate Radionuclides

The releases of radioiodines and particulate radionuclides (not including noble gases) to the atmosphere will result in internal exposure through inhalation, ingestion pathways and external exposure to contaminated ground to individuals in the population. The maximum individual doses at the highest residential dose point are calculated based on the models in Regulatory Guide 1.109 for each age group and the following pathways:

1. Inhalation of air at the highest residential dose point
2. Ingestion of leafy vegetables and other food products grown at the highest residential dose point
3. Ingestion of milk and meat produced by the animals at the highest dairy farm dose point
4. External exposure to the contaminated ground plane at the highest residential dose point.

The release points and dilution factors are discussed in Subsections 11.3.3.3 and 11.3.3.4 above. All the parameters, usage factors for maximum individuals and dose conversion factors are taken from the suggested values in Regulatory Guide 1.109, except (1) fraction of year animals being on pasture = 0.5, and (2) absolute humidity = 8.0 g/m^3.

The calculated highest residential dose point from all release points is approximately 7867 feet from the station, in the ESE sector, with annual average dilutions of the following:

1. Stack releases:

   $(X/Q)_{depleted}$ = 2.0E-07 sec/m^3; D/Q = 1.11E-09/m^2.

2. Turbine Building releases:

   $(X/Q)_{depleted}$ = 1.06E-06 sec/m^3; D/Q = 3.17E-09/m^2.

The highest dairy farm dose point is approximately 2.4 miles from the station, in the NW sector, with an annual average D/Q = 2.64E-10/m^2 from the stack release, and 3.31E-10/m^2 from the Turbine Building vent release.

Table 11.3-4 shows the estimated maximum organ doses for each age group from all pathways due to the annual gaseous releases in Table 11.3-2. Among four age groups, the estimated maximum doses, 0.08 mrem/yr to total body, 0.16 mrem/yr to thyroid and 0.23 mrem/yr for the child bone pathway (the most critical organ), are small fractions of the numerical design dose objectives in Appendix I to 10 CFR 50.

11.3.4 References

1. USNRC NUREG-0017, "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Pressurized Water Reactors (PWRs)," (PWR Gale Code) April 1976.

11.4 SOLID WASTE MANAGEMENT SYSTEM

11.4.1 Design Basis

11.4.1.1 Design Objective and Criteria

A Solid Waste Management System is provided to serve at the nuclear generating unit.

The Solid Waste Management System processes waste liquids, spent resins and dry wastes for onsite storage or disposal off site. The system is designed and operated to meet the limits for controlled releases of radioactive liquids from the site set forth in the Offsite Dose Calculation Manual (ODCM).

The Solid Waste Management System is designed in accordance with 10 CFR 20; Regulatory Guides 1.143, 8.8, and 8.10; Standard Review Plan 11.4; Branch Technical Position 11-3; and ANSI/ANS 55.1. A generic topical report discussing the WasteChem volume reduction and solidification system was accepted by the NRC on April 12, 1978. Considerations of operation, maintenance, and accident conditions have been factored into the system design to maintain radiation levels as low as reasonable achievable. Equipment layout and shielding are designed to limit radiation levels in areas accessible by the operator during a solidification operation to less than 15 mrem/hr, and in the radwaste control room levels are less than 2.5 mrem/hr. Handling, storage, and shipping of radioactive waste will be performed in conformance with 10 CFR 50, 61, and 71.

The containers used for solid waste storage and offsite shipment will meet the appropriate requirements of 49 CFR 171-179 (Department of Transportation Radioactive Material Regulations), and 10 CFR 71 (Packaging of Radioactive Materials for Transport).

System design incorporates backup processing capability for the WL, BRS and SB evaporators via the liquid waste volume reduction subsystem.

A 10 CFR 61 wasteform qualification program has been conducted (Reference 6) to demonstrate asphalt wasteform compliance with the NRC branch technical position for Class B and C wastes prior to using the asphalt system for waste processing. A process control plan (PCP) will be implemented to ensure compliance with 10 CFR 61 wasteform and burial site requirements. The wasteform qualification program (Reference 6) was submitted to, and accepted by, the NRC.

Waste will be classified pursuant to 10 CFR 61 classification requirements by use of isotopic inferral/correlation techniques in conjunction with periodic calibration programs.

The Solid Waste Management System is comprised of several subsystems as follows:

- Spent Resin Sluicing
- DAW Volume Reduction
- Alternate Solidification
- Volume Reduction and Solidification in Asphalt

Volume reduction and solidification in asphalt consists of:

- Waste concentrates handling
- Spent resin handling
- Liquid waste volume reduction
- Material handling

The system also provides the necessary instrumentation and connections for adequate control and monitoring of wet radwaste for delivery to contracted mobile solidification services equipment or to the WPB truck bay and/or adjacent shielded storage area for processing.

These subsystems are all interrelated with the exception of the DAW Volume Reduction which operates independently of any other subsystem. The various wet and dry solid radioactive wastes may be processed by either a permanently installed or mobile Solid Waste Management System.

a. Waste Processing Subsystems Using Asphalt

The permanently installed Solid Waste Management System, located in the Waste Processing Building (WPB), provides the following functions:

1. Collection and storage of spent resins from plant demineralizers or vendor-supplied systems
2. Collection and storage of concentrated wastes containing up to 12 wt% boric acid
3. Volume reduction to the maximum extent practical of all liquid concentrates and spent resins produced
4. Solidification of volume reduced waste, using an asphalt binder, as necessary, for offsite disposal in accordance with the requirements of 10 CFR 61
5. Encapsulation/solidification, using an asphalt binder, of noncompactible waste such as spent filter cartridges and similar items generated during plant operation and maintenance

6. Limited temporary onsite storage for solidified in asphalt low level wet waste, with a capacity of 15 months based on design volume production rates. Expected volume production rates yield capacities of 3.6 years.

The Solid Waste Management System is nonnuclear safety and nonseismic, and is designed in accordance with the requirements as set forth in NRC Regulatory Guide 1.143. The operation of the system has no effect on the capability to bring the plant to a safe shutdown condition.

b. Alternate Waste Processing Via Mobile Contractor

The processing of wet wastes and encapsulatable dry waste, in final preparation for onsite storage and eventual offsite shipment for disposal, may be accomplished using the services of a contractor using mobile equipment. Available contractor mobile equipment will utilize an approved solidification agent. The mobile solidification vendor wasteform qualification program has been placed on file. Permanently installed solid waste management system equipment needed for proper connections to and monitoring of waste inputs made to the mobile solidification services contractor equipment is listed in Table 11.4-1.

The station services and equipment necessary for the operation of the mobile solidification services to process liquid concentrates, spent resin or spent filter cartridge type waste are designed to:

1. Deliver liquid radwaste to an alternate Mobile Solidification System in the truck bay or in the adjacent shielded storage area.
2. Solidify completely all radioactive waste concentrates and chemical wastes.
3. Solidify noncompressible contaminated items such as spent filter cartridges and other items generated during plant operation and maintenance.
4. Package the solidified radioactive waste in containers suitable for transportation to a licensed burial site.

Temporary onsite storage of waste is available in the Waste Processing Building. Additional onsite storage will be provided as required.

The Mobile Solidification System has the capability to solidify an input volume of at least 16,000 ft^3/yr of waste consisting primarily of spent resin and 12 wt% boric acid concentrates, and to encapsulate and solidify 600 ft^3/yr of spent filter cartridges, contaminated and/or activated tools and other equipment. The resin sluice tanks (2) can accommodate no more than one year's supply. Spent resin and process filters can be dewatered with a vendor-developed dewatering process.

A Process Control Program (PCP) is in place that identifies the programs and procedures necessary for the processing, disposal and burial of dewatered resins and process filters.

c. Portable Solidification System Interface

1. Wet waste storage tanks are limited to the inplant installation.
2. Piping used to interface the plant's waste lines to the portable equipment complies with the requirements of Regulatory Guide 1.143.
3. The waste processing components of a portable system is appropriately arranged in the area bounded by lines A and D and columns 5 and 6 at elevation 25 feet of the Waste Processing Building. The area is provided with floor drains and monitored building ventilation exhaust ducts. Spill control arrangements will be evaluated on a case basis, since they would be component and equipment arrangement specific.

11.4.1.2 System Inputs

The Solid Waste Management System has several input sources. The following subsections discuss these inputs on a yearly operational basis, including maintenance and anticipated transients. Table 11.4-2 and Table 11.4-3 list the maximum and expected volumes and expected activity, respectively, for each source. The bases and assumptions used in determining the solid waste activities for offsite shipment for each waste type are listed in Table 11.4-4. The bases and assumptions used for determining the waste activities inside of each major subsystem component are presented in Section 12.2. Radionuclide concentrations and volumes are consistent with reactor operating experience presented in NUREG/CR-0144, ORNL-4924, NUREG/CR-1759, and NUREG/CR-1992. Process flow diagrams for the Solid Waste Management System are shown in Figure 11.4-11, sh1, and Figure 11.4-11, sh.2. Note that process flow diagrams, Figure 11.4-11, sh1, and Figure 11.4-11, sh.2 are provided for historical information only. Flow parameters are original plant design. P&IDs for the system are given in Figure 11.4-1, Figure 11.4-2, Figure 11.4-3, Figure 11.4-4, Figure 11.4-5, Figure 11.4-6, Figure 11.4-7, Figure 11.4-8, Figure 11.4-9 and Figure 11.4-10. Layouts identifying packaging, shipping, and storage areas are given by Figure 1.2-22, Figure 1.2-23, Figure 1.2-24, Figure 1.2-25, Figure 1.2-26, Figure 1.2-27, Figure 1.2-28, Figure 1.2-29 and Figure 1.2-30.

The estimated solid radwaste activity/volume values presented in the following sections represent data that were part of the original license application and is retained here for historical purposes.

Operation at the licensed core power level is expected to have minimal impact on installed equipment performance, system operation, and maintenance. Consequently, only minor, if any, changes are expected in waste generation volume. The activity levels for most of the solid waste will, however, increase proportionately to the increase in long half-life coolant activity, bounded by the percentage increase in power level to the licensed core power level.

a. Dry Active Wastes

Dry active wastes are classified into two categories. The two categories are (1) noncompactible and (2) compactible. Examples of noncompactible wastes would include small items such as used hand tools which cannot be economically decontaminated, electrical connectors, wood, et al., from contaminated areas. Examples of compactible wastes would include paper, polyethylene, tape, anti-contamination clothing, gloves, and shoe covers that are contaminated and/or beyond repair. The activity concentrations for offsite shipment, after processing, are listed in Table 11.4-5.

b. Spent Demineralizer Resins

The resin sluice tanks, located in the WPB at elevation (-)31' provide the collection point for spent resins. Normally spent resins are transferred to the shielded storage area or the WPB truck bay for processing, dewatering, and/or storage, as an alternative, spent resins may be solidified in asphalt using a permanent or alternate solidification system. The spent resin transfer pump takes suction on either of these tanks and transfers resin to the resin hopper at elevation 53'. After the resin hopper is filled, resin is processed by dewatering via the resin dewatering pump. The hopper fill operation is repeated until the proper resin slurry density is obtained in the resin hopper.

The resin process path is then to the resin centrifuge. Transport water is driven from the resin in the centrifuge and the resin free falls into the evaporator/extruder where it is homogeneously mixed with asphalt binder in an approximate one-to-one ratio, by weight, and is then discharged into an appropriate shipping container.

The function of the centrifuge is to remove transport water from the resin slurry, and thereby increase the system throughput by reducing the evaporative demand on the evaporator/extruder. As an alternative, resin slurry can also be fed directly to the evaporator/extruder via the concentrates metering pump. Resin slurries can also be pumped to the alternate solidification station located in the truck bay, via the resin centrifuge metering pump, for either solidification or dewatering in a disposable container.

At the alternate solidification station, a contractor with mobile waste processing equipment would then proceed to homogeneously mix the slurries with a solidification binder and then package them in an appropriate shipping container.

Spent resins from the following demineralizers are collected in the spent resin sluice tanks prior to processing:

1. Spent fuel pool demineralizer
2. Chemical volume control system demineralizers
3. Liquid waste system demineralizers
4. Boron recovery system demineralizers

The expected activity concentrations for offsite shipment, after processing, are listed in Table 11.4-6.

c. <u>Evaporator Bottoms and Other</u>

The waste concentrates tank at elevation (-)31' collects the evaporator bottoms and chemical drains from evaporators and drain tanks located throughout the plant. Concentrates are transferred to the waste feed tanks on elevation 53'. Once in these tanks, pH is adjusted and the concentrates can be fed to the crystallizer, or directly to the evaporator/extruder or to the alternate solidification station.

Chemically neutralized concentrates fed to the evaporator/extruder are homogeneously mixed with asphalt binder in a one-to-one ratio of solids to asphalt by weight, and discharged into appropriate approved containers. At the alternate solidification station, a contractor with mobile waste processing equipment would homogeneously mix the concentrates with an acceptable binder and package them into appropriate shipping containers.

The normal design operating mode is to process these concentrates through the crystallizer producing a concentrated bottoms of 35-50 percent total dissolved solids. These bottoms are then fed to the evaporator/extruder for further volume reduction by the removal of water, and solidification of the radsalts.

The concentrates collected for processing in the Solid Waste Management System on a batch basis, include evaporator bottoms and chemical drains from the following systems:

1. Liquid Waste
2. Boron Recovery
3. Steam Generator blowdown
4. Floor and Equipment Drains.

Other waste inputs handled by the system include the spent filters from various radioactive and/or potentially radioactive process systems from throughout the plant. Typically spent filter cartridges and other similar waste items are processed via a technique of encapsulation and solidification, using either cement or asphalt binder within a filter basket placed inside a 55-gallon drum or other container as needed. The expected activity concentrations of evaporator bottoms, chemical drains, and encapsulated and solidified waste for offsite shipment, after processing and binding with either asphalt or cement, are listed in Table 11.4-7.

11.4.2 System Description

The Solid Waste Management System is a plant system designed for the management of the final processing of wet or dry solid radwastes within the Waste Processing Building. Wet solid radwastes include spent demineralizer resins, concentrates from evaporator bottoms, chemical wastes, spent filter cartridges, and miscellaneous wastes from floor and equipment drains. Spent demineralizer resins are handled primarily by dewatering and storage prior to shipment offsite for burial. As an alternative, dewatered resins may be solidified using a permanent or alternate solidification system prior to disposal by one or more dewatering steps prior to being mixed with asphalt or other binder using a screw evaporator/extruder or fed directly to mobile equipment for dewatering and/or solidification with either cement or other approved binder. Concentrates at 12-50 wt% solids concentration are chemically adjusted, as necessary, then fed to the evaporator/extruder for the removal of water and mixing with the asphalt binder or fed directly to mobile equipment for mixing with an alternate binder. Dry solid waste includes both compactible and noncompactible trash. Transfer of wet wastes to contracted mobile services equipment is accomplished via use of the alternate solidification concentrates feed and the centrifugal metering pump and is controlled from a dedicated alternate solidification control panel located in the loading dock at elevation 25' of the Waste Processing Building.

The Solid Waste Management System is located in a separately shielded areas of the Waste Processing Building at the (-)31', 25', 42'-5", 53', and 55' elevations in the southwest corner of the building. Radwaste originating from sources from within the plant is controlled by this system for processing either by permanently installed equipment and or by contracted mobile solidification services. Wet radwastes for ultimate solidification and/or volume reduction into appropriate storage/shipping containers is collected into various tanks located at elevation (-)31' prior to transfer to the Solid Waste Management System for processing. Personnel exposure is kept as low as reasonably achievable by the use of shielding, by the use of closed circuit television cameras, by the provision of a separately shielded processed waste container storage area, and by the provision of a separately shielded loading dock area. Radioactive solid waste equipment parameters are summarized in Table 11.4-8.

11.4.2.1 Component Descriptions

The following are descriptions of major permanently installed mechanical system components.

a. Spent Resin Hopper (WS-TK-81)

The spent resin hopper is located at elevation 53' of the WPB. It has a working capacity of 934 gallons. The hopper accepts radwaste in the form of a spent resin slurry directly from the spent resin sluice tank. A volume reduction of this waste is accomplished in the hopper using an integral mixer, a dewatering screen, and decantation. A solids concentration of 15 wt% is expected. During recirculation pH is adjusted, if required. The normal disposition of the dewatered slurry produced by the hopper is feed to the centrifuge. Decant is returned to one of the resin sluice tanks. The hopper includes permanently installed internal sparging nozzles to permit remote decontamination after each use.

b. Resin Centrifuge (WS-MM-611)

The resin centrifuge is located above floor elevation 55' of the WPB. The unit has a maximum processing capacity of 4.7 gpm of resin slurry feed at a dry solids concentration of 10 wt%. It produces a processed effluent at a rate of 440 pounds per hour at a dry solids concentration of 50 wt% for feed to the evaporator/extruder and mixing with asphalt. This volume reduction is accomplished via the use of centrifugal forces to separate the liquid and solid phases of the resin slurry feed. Resin slurry feed is taken directly from the spent resin hopper recirculation for processing. Decant is returned to one of the resin sluice tanks.

c. Crystallizer (WS-EV-6)

The principal components of the crystallizer are located above floor elevation 55' of the WPB. The crystallizer is a low head, submerged tube, forced circulation package with a separate entrainment separator. The crystallizer has a working capacity of 1200 gallons; 600 gallons within the vapor body and another 600 gallons within its associated heater and recirculation piping. Its external two-pass horizontal heater utilizes the Plant Auxiliary Steam System. Steam is applied to the shell side of the heater where it condenses on the outside of the tubes and transfers heat to the liquor circulating inside the tubes. The steam condensate is then removed from the shell side of the heater and returned to the condensate storage tank. The liquor circulating through the tubes is not allowed to boil. After the liquor passes through the heater, it enters the vapor body where it releases water vapor to the entrainment separator, crystallizer condenser, and subcooler. The recirculation pump, vapor body, and heater are designed to process undissolved solids, with the capability to crystallize inorganic salts in typical PWR process waste concentrates.

d. Waste Feed Tanks (WS-TK-198A,B)

The two waste feed tanks are located at elevation 53' of the WPB. They each have a working capacity of 1,000 gallons for processing of radwaste originating from various plant evaporators and drain tanks. These tanks accept radwaste directly from the 6,000-gallon waste concentrates tank. When the waste feed tanks are filled to working level a sample is taken from their recirculation loop, their pH is adjusted, as necessary, and their contents are homogeneously mixed by a mixing eductor located inside the tank at the inlet nozzle. Their contents are then fed, normally on a batch basis, to the crystallizer system. Bottoms from the crystallizer are subsequently transferred to the concentrates bottoms tank prior to processing through the evaporator/extruder.

The contents of the waste feed tanks are kept heated in preparation for pre-concentration volume-reduction in the crystallizer vapor body, using electric strip heater elements wrapped around their girth. A dip pipe is provided in each tank for level indication. A permanently installed spray ball assembly is also included within the top of each tank to provide a remote decontamination capability after the processing of each batch of radwaste concentrates.

e. Concentrates Bottoms Tank (WS-TK-200)

The concentrates bottoms tank is located at elevation 53' of the WPB. It has a working capacity of 1,000 gallons. It receives bottoms directly from the crystallizer vapor body after the desired solids concentration has been attained. The tank is necessary for holdup prior to feed to the evaporator/extruder for final volume reduction and solidification via blending with asphalt. The holdup is required primarily because of the difference in processing rates of the evaporator/extruder and the crystallizer vapor body. It also allows final chemistry adjustments, as necessary, prior to processing through the evaporator/extruder.

The tank is kept electrically heated during processing. The tank is decontaminated remotely after each batch by using a built-in spray arrangement located in the top of the vessel. The contents of the tanks are kept homogeneously mixed by use of a mechanical agitator during feed to the evaporator/extruder.

f. Evaporator/Extruder (WS-EV-7)

The evaporator/extruder is located at elevation 42'-5" of the WPB. It combines volume reduction and solidification with asphalt (bitumen) of either spent resin slurries or waste concentrates. Both the radwastes and the asphalt (heated to 325°F) are fed simultaneously into the twin screw, steam heated, eight-section evaporator/extruder where the associated transport water and entrained moisture is evaporated and vented through the three steam domes during the blending process. Up to 99.5 percent of the associated free water is removed at the rate of 21 gallons per hour for a resin waste stream or at the rate of 32 gallons per hour from a concentrates waste stream.

Product discharge rate into shipping containers varies from about 9.25 to 27.5 gallons per hour, depending upon the speed of the screws and the feed stream solids concentration. Total residence time of material inside the evaporator/extruder is on the order of one minute and the total inventory of the unit when full of asphalt and waste is less than one gallon. The solids-to-asphalt weight ratio fed into the storage/shipping containers will vary up to one-to-one as determined by waste feed radioactivity concentration and weight percent solids loading. The evaporator/extruder is remotely controlled and has a variable speed motor drive. The unit can operate either continuously or in an on-off mode. If feed is stopped, it will continue to operate and self-clean within approximately one minute. Should power or steam failure occur allowing an asphalt mixture to harden inside of the evaporator/extruder, simple heating to the process temperature after repairs restores it to normal operation.

g. Asphalt Storage Tank (WS-TK-201)

The asphalt storage tank is located at elevation 20' of a separate asphalt storage building adjacent to the southwest corner of the WPB. It has a working capacity of 7000 gallons. This capacity is adequate to meet normal plant processing requirements for more than four months. The tank utilizes steam panels to permit heating using steam provided by the dedicated auxiliary boiler. The tank is designed to receive asphalt from a vendor's tank truck at a fill rate of 100 gpm, while simultaneously recirculating and straining the influx. The recirculation permits the rapid achievement of a homogeneous temperature distribution within the tank. Recirculation will be at a rate of 20 to 25 gpm. Fill strainers are designed to remove any foreign particles which could be present in the influx. Recirculation strainers are designed to remove foreign matter missed by the fill strainers.

h. Auxiliary Boiler Skid (WS-SKD-112)

The auxiliary boiler skid is located at elevation 53' of the WPB. The skid includes the electrically heated auxiliary boiler, the condensate return tank, two boiler feed pumps, blowdown tank and sample cooler plus associated valves and motors. The auxiliary boiler is designed to deliver steam at a rate of 5,000 pounds per hour at an operating pressure of 275 psig and an operating temperature of 410°F as needed to maintain asphalt within system piping and equipment in a molten condition for extended periods.

i. Chemical Feed Skid (WS-SKD-115)

The chemical feed skid is located at elevation 53' of the WPB. The skid includes the chemical feed tank and the chemical feed pump, plus associated valves and motors. The chemical feed tank is designed with a working capacity of 50 gallons and provides chemical feed to the auxiliary boiler.

j. Caustic Skid (WS-SKD-111)

The caustic skid is located at elevation 53' of the WPB. The skid includes the caustic day tank and the caustic metering pump, plus associated valves and motor. The caustic day tank is designed with a working capacity of 200 gallons and can supply caustic to the waste feed tanks, spent resin hopper, and the concentrates bottom tank.

k. Vent Hood Assembly (WS-MM-615)

The vent hood assembly is located at elevation 42'-5" of the WPB. The vent hood assembly is designed to fit over the container(s) to be filled, and exhaust any off-gassing of volatile vapors or steam carry-over at the evaporator/extruder discharge port or from the waste container being filled, to the WPB ventilation exhaust filter system. The vent hood assembly is equipped with redundant level probes and a camera for visual level inspection.

l. Compactor & Ancillary Equipment (WS-MM-722)

The compactor and its ancillary equipment have been permanently removed from elevation 25' of the WPB. Other methods of waste volume reduction, including use of a filter shear, may be utilized.

m. Material Handling

A 30-ton bridge crane, WS-CR-35, 55-gallon drum conveyors, a 7½-ton self-propelled transporter cart, a process aisle overhead monorail, and fork lifts are provided for in-plant movement of solid waste containers. These items, in conjunction with the use of closed circuit television monitors, shielding and remote operation capability provide as low as reasonably achievable radiation protection for plant or contractor personnel.

11.4.2.2 Operating Procedures

The solid wastes listed in Subsection 11.4.1.2 are handled according to their physical properties. The system is used intermittently and requires one full-time operator in attendance during steady-state system operation. Control of the system is remote-manual.

The entire volume reduction, solidification and drum handling systems are controlled remotely from two control panels in the shielded control room of the Waste Processing Building (WPB).

Waste solidification processing is controlled from a single panel which integrates all the process subsystems. A programmable controller provides for automatic operation after manual startup of the system. Interlocks and permissives prohibit inadvertent or improper operation of subsystems and also form the basis of the solidification process control program. Details of the process control program for the use of permanently installed equipment utilizing an asphalt binder are given in a separate process control document. Details of a process control program for the use of contracted mobile solidification or dewatering services have been placed on file for NRC review.

The overall Radwaste Management System is designed to provide maximum flexibility in the processing of the various waste streams. Spent resins are processed by dewatering, volume reduction and/or solidification. Evaporator bottoms and chemical wastes are processed by volume reduction and solidification. Both waste types are capable of being pumped to an alternate solidification station in the truck bay area for solidification, or in the case of resin dewatering, into a dry form.

a. Waste Concentrates Handling Subsystem

The waste concentrates tank receives concentrated nonrecyclable wastes from the liquid waste evaporator, the two boron recovery evaporators, the three steam generator blowdown evaporators, the chemical drain treatment tanks, the floor drain tanks and return from the waste feed tanks. Concentrates in the tank are heated by electric tracing to prevent solidification of concentrated material. A waste concentrates transfer pump is provided for recirculation of the waste concentrates tank and transfer of the waste concentrates to the waste feed tanks. The overflow from the waste feed tanks is directed back to the waste concentrates tank. The overflow from the waste concentrates tank is directed to a floor drain.

The pre-concentrated liquid waste solution is pumped from the waste feed tanks to the crystallizer to remove excess water more quickly with the final waste volume reduction and solidification achieved upon subsequent delivery to either the permanently installed evaporator/extruder using an asphalt binder or alternately upon delivery to contracted mobile equipment for binding with an approved binder. The radioactive waste in the form of a slurry, containing

35-50 percent total dissolved solids by weight, is pumped to either the permanently installed asphalt evaporator/extruder by the concentrates metering pump or to the alternate solidification station using alternate solidification concentrates feed pump.

b. Spent Resin Handling Subsystem

The purpose of the spent resin dewatering subsystem is to remove the spent resins from the various demineralizers in the Radioactive Liquid Cleanup Systems, to store the resin for a time sufficient to reduce the radiation levels to an acceptable level, and then to pump the spent resin to the spent resin hopper where it is prepared for solidification and offsite disposal or to the WPB truck bay or adjacent shielded storage area for processing, storage, and/or shipment offsite.

The four basic operations performed by the system are:

1. Removal of spent resin from the demineralizers in various Radioactive Liquid Waste Cleanup Systems in the Primary Auxiliary and Waste Processing Buildings by a sluicing operation, and storage of these resins in the spent resin sluice tanks.
2. Post-sluicing cleanup of the resin sluice piping to lower the radiation levels, by recirculating the sluice water through the pipes and then through the resin sluice filter.
3. Recirculating the liquid in the sluice tanks and transferring the contents from one tank to the other, to equalize bulk or dose.
4. Spent resins are normally transferred to the WPB truck bay or to the adjacent shielded storage area. Alternatively, the spent resins may be transferred from the sluice tanks by the spent resin transfer pump for dewatering via the spent resin hopper and/or the centrifuge or directly to the shielded storage area or truck bay. The final waste volume reduction will be accomplished upon delivery to either the permanently installed evaporator/extruder using an asphalt binder or alternately upon delivery to mobile equipment.

The spent resin hopper receives waste from the spent resin sluice tanks. The agitator in the hopper keeps solids and spent resin in suspension, and the spent resin dewatering pump and pump suction screen allow for removal of water from the resin slurry. The spent resin hopper is vented to the plant vent via the aerated vent header. Samples can be drawn from the recirculation line from the hopper for direct analysis of radionuclides, waste, and spent resin concentrations, as well as total activity level. Direct sampling also determines the process control requirements and container shield requirements.

The spent resin is pumped from the spent resin hopper in the form of a bead resin slurry by the resin recirculation and the resin centrifuge metering pumps.

c. <u>Liquid Waste Volume Reduction Subsystem</u>

The liquid waste volume reduction subsystem provides for the concentration of various liquid wastes from a nominal 5-12 wt% total dissolved solids to 35-50 wt% total dissolved solids. This concentration is accomplished by a 5-gpm forced circulation evaporator/crystallizer operating on plant auxiliary steam.

The prime function of the crystallizer is to quickly remove excess water from the liquid waste stream prior to feeding the asphalt evaporator/extruder, which has a comparatively low evaporation rate. The crystallizer also provides for a means of volume reduction independent of the evaporator/extruder operation and, in addition, allows for a reduced demand on the station's existing upstream evaporators.

Feed from the waste feed tanks enters the recirculation loop of the crystallizer and mixes with the recirculation flow prior to entering the heater. This stream then enters the vapor body where part of the flow flashes to steam. The flashed steam then enters the entrainment separator where it passes through distillation trays and a demister pad and then enters the condenser skid and ultimately returns to the plant liquid waste system or discharges to the Circulating Water System.

Bottoms from the crystallizer are pumped from the main recirculation loop to the crystallizer bottoms tank by the crystallizer drain pump.

System operation is dependent upon the amount of waste available for volume reduction. The system is capable of continuous operation with periodic bottoms blowdown. However, the crystallizer will normally be run on a batch basis, concentrating to the maximum extent practical and then shutting down.

The entire contents of the 6000-gallon waste concentrates storage tank can be volume reduced in the crystallizer and stored in the crystallizer bottoms tank prior to processing through the extruder/ evaporator.

d. Asphalt Volume Reduction and Solidification Subsystem

The evaporator/extruder is the main component of this subsystem and the heart of the entire volume reduction and solidification process. It is located in the WPB at elevation 42'-5". In the evaporator/extruder the waste stream (concentrates or resin) mixes with asphalt which is then heated by steam to evaporate the remaining water in the mix. The resulting matrix contains approximately 50 percent dried residual waste and 50 percent asphalt by weight and is deposited into either 55 gallon drums or 85 ft^3 containers. As the matrix cools from the operating temperature, solidification takes place due to the thermoplastic properties of asphalt. Water evaporated from the waste during volume reduction is condensed in the three evaporator/extruder steam domes and is gravity drained to the floor drain tanks.

Steam for this process is supplied by an electric auxiliary boiler at 410°F and 275 psig. Asphalt is supplied from the 7,000-gallon asphalt storage tank located in a separate building at grade just south of the WPB. Asphalt lines are steam traced using the high temperature steam from the solid waste auxiliary boiler. Plant auxiliary steam is provided as a backup system in case the solid waste system auxiliary boiler needs to be shut down for maintenance.

Normally, 55-gallon drums are used for disposal of solidified waste. However, the ability to utilize 85 ft^3 containers is provided. The following steps are required in order to align the system for usage with 85 ft^3 containers: (1) The drum capper station is relocated to the position required for functioning with 85 ft^3 containers. The cap stack unit is removed. (2) The 55-gallon drum turntable station is removed by unbolting the four flanged end connections. (3) The drum grab and rotator unit is changed out via the 30-ton overhead crane to enable use of the large container lifting rig. (4) A vent hood extension is added to properly interface the fill port to the large container. (5) The 85 ft^3 container will be loaded onto and off of the transfer cart via the overhead crane. (6) The waste solidification control panel settings are adjusted for 85 ft^3 container operation.

e. Material Handling Subsystem

The components of this subsystem are all located within the WPB at floor elevation 25', except for the radwaste crane which sits at elevation 73'. The empty drum conveyor is loaded from the solid waste control room with empty drums (6 minimum, 10 maximum) prior to the start of solidification operations. The drum hoist and grab lifts empty drums from the empty drum conveyor and places them on the turntable prior to fill operation.

The evaporator/extruder utilizes a one-step volume reduction and solidification process that operates continuously. The drum handling subsystem is designed to allow continuous flow of drums through the process area to take full advantage of the benefits that continuous system operation provides.

Operation of the evaporator/extruder will cause a drum to fill with the asphalt/waste mixture. After a drum has been filled, the turntable rotates an empty drum into the fill position and the filled drum into the pickup position on the turntable. Multiple fill passes on the turntable ensure maximum fill efficiency in each drum. The drum hoist and grab lifts the full drum and places it on the full drum conveyor. At this time an empty drum may be placed on the turntable if continuous operation beyond the six drum capacity of the turntable is anticipated. The drip pan mechanism is provided for product collection during drum indexing. The pan with drippings is deposited into the next empty drum after indexing.

All drum movement operations on the drum conveyors and turntable are manual start with automatic stop. All drum lifting and capping operations are manually controlled. Crane operation is manually controlled with specific lockouts to prevent crane travel with load in down position and to prevent collisions.

On the full drum conveyor the drum moves to the capping station where the capper is activated to pick up a cap from the cap stack and place it on the drum and crimp it in place. After capping, the swipe station turntable allows the drum to spin. A swipe may be taken, at this time, by the swipe manipulator to check for external contamination. The drum is then transferred to the end of the full drum conveyor where the overhead radwaste crane can pick it up and place it in the storage area.

All drum handling and process operations can be visually monitored via closed circuit television. A lead glass window on the fill aisle wall also provides direct visual observation of the capping and swipe operations.

As an alternative to filling drums, 85 ft^3 liners may be used. Liners will be located under the evaporator/extruder discharge by the transfer cart. The cart also moves liners to the capping and crane pickup positions. The use of liners or drums will be dictated by disposal economics.

f. <u>Dry Active Wastes (DAW) Volume Reduction Subsystem</u>

A subsystem for dry active waste is provided. The subsystem utilizes a 100 ft^3 box compactor with anti-spring back devices.

The subsystem handles typical trash collected at this operating plant with an average collected density of 7 to 10 lbs/ft^3 with a final waste density of approximately 30 to 40 lbs/ft^3.

This system combines reliable forty-year service life with maximum operator safety, compacting efficiency and ALARA exposure time.

The compactor exhaust system operates to prevent dust from escaping during trash handling and is complete with fan, fan motor, prefilter, absolute filter and connects to the duct in the filled drum storage area exhaust system.

g. Alternate Solidification

Waste concentrates or resin slurries prepared as described above will be delivered to the alternate solidification station, located in the truck bay area for processing via a mobile solidification or resin dewatering services contractor. Also provided will be flush water, vent lines, resin water return lines, isolation valves, and a control panel for the pumps and valves.

Two small panels in the truck bay supplement the main control and drum handling panels by providing functional control over waste flow to the alternate solidification station and control of crane movements in the truck bay area only.

11.4.2.3 System Controls, Protective Devices, and Instrumentation

The Solid Waste Management System has design provisions incorporated to reduce leakage, control the unplanned release of radioactive materials, and facilitate operation and maintenance in accordance with the guidelines of Regulatory Guide 1.143. Mechanical protective devices and/or instrumentation incorporated for the reduction of leakage potential and for control of the potential for uncontrolled releases of radioactive materials include: (a) a 100-micron filter array attached to the suction side of each of the two resin sluice pumps preventing resin movement except by the system; (b) system piping normally containing resin or concentrates has butt welds and five diameter bends; (c) a double mechanical seal for system pumps to prevent any leakage at the shaft; and (d) a vent hood assembly which is designed to collect off-gases, (e) seal water to most pumps is a closed system with individual seal water tanks. The remainders have a dead end seal water design.

Mechanical protective devices and/or instrumentation incorporated for the facilitation of operation and maintenance include: (a) capability to supply low pressure nitrogen to each of the two resin sluice tanks to prevent the possibility of the formation of an explosive hydrogen/oxygen mixture; (b) equipment within the system has high point vents and low point drain valves; (c) equipment layout is arranged so that radiation exposure during handling, volume reduction, or solidification operations is limited to less than 15 mrem/hr in operator occupied areas; (d) redundant disposal container level probes and television cameras provided for monitoring to minimize the potential for spills due to overflows during filling; (e) a 30-ton indexed radwaste crane which is remotely controlled from a shielded control station viewed through closed circuit television cameras; and (f) tank cubicles incorporate concrete curbs to further aid in controlling radioactive releases in the event of an overflow.

Radiation levels are monitored near the spent resin hopper cubicle and in the filling area by detectors permanently located in each of these areas. Local indication alarms are provided.

Contents of the hopper, the bottom tank, and the waste feed tanks are monitored for pH, with remote indication at the shielded waste solidification control panel.

Instrumentation is provided to monitor the pressure, level and temperature in various parts of the system to aid safe operation of the system from the main control panel.

Upon loss of power or air, the system reverts to the safe position.

In addition, due to the complexity of the Resin Sluice System and number of systems with which it interfaces, administrative procedures require a valve line-up inspection prior to any sluicing operation to insure that:

1. Operating systems under pressure do not become lined up to the sluice system.
2. Sluice water is not pumped into an operational system which is shut down and depressurized.
3. The sluice pump suction is not lined up with one sluice tank and the return flow to the other tank.

11.4.2.4 Maximum and Expected Processed Waste Volumes

The maximum and expected volumes of radioactive solid wastes available for offsite shipment annually, for each source, are presented in Table 11.4-9 considering the use of either an asphalt or a cement binder. The projected annual volumes are based on uniform one-to-one mixing ratio for a commercial asphalt with the maximum practical volume reduction of all wet solid plant wastes or a one-and-one-half-to-one mixing ratio using a cement binder without volume reduction. For spent demineralizer resins, the minimum expected applied volume reduction will be a factor of 1.85 and for evaporator bottoms and other, the minimum volume reduction will be 6.3. For compressible dry wastes, the minimum expected volume reduction will be 4. It is expected that, using an asphalt binder less than 22 containers with a 85 ft^3 capacity or 249 drums with a 55-gallon (7.35 ft^3) capacity will be needed for wet radwastes such as spent resins and evaporator bottoms and for encapsulation of other wastes such as spent filter cartridges, contaminated tools, etc. Less than seventy boxes with a 100 ft^3 capacity will be needed for all dry wastes such as swipes, rags, anti-contamination clothing, etc., on an annual basis.

It is expected that, using a cement binder, less than 118 containers with an 85 ft^3 capacity or 1360 drums with a 55-gallon capacity would be needed for wet radwaste and noncompressible wastes on an annual basis.

11.4.2.5 Packing

Spent resins, evaporator bottoms and miscellaneous chemical wastes are solidified in various-sized containers. The containers may be filled remote-manually from behind a shield wall using closed circuit television to avoid unnecessary exposure. Spent cartridge filters may be encapsulated or dewatered in large containers; dry low-level wastes are compacted to the maximum extent practical and shipped or stored in nominal 100 ft^3 boxes. Filled containers are shipped as required in appropriate overpacks and shields. Solid waste containers, shipping casks, and methods of packaging will meet applicable state and federal regulations, including 10 CFR 71.

11.4.2.6 Storage Facilities

Radioactive waste is stored in various locations in the Waste Process Building. The shielded storage area adjacent to the truck bay at elevation 25' is normally used to process and store resins and other packaged low level wastes prior to shipment offsite. Other areas in the WPB may be used to store radioactive material and wastes, as necessary in accordance with station procedures. The Unit 2 Cooling Tower is used to store DAW and other non-liquid radioactive material prior to re-use or shipment offsite. The Asphalt Storage Building is used to store radioactive material including low-level contaminated liquids. Additional storage capacity will be provided, as necessary, on a timely basis.

The shipment of solid radwaste does not disturb normal plant operations. Means of in-plant transport includes fork lifts, monorails and a bridge crane. Loaded trucks may stay overnight inside the enclosed radioactive material loading dock area, which is a restricted area with controlled access.

11.4.2.7 Shipment

Radwaste from the radioactive material storage areas will be loaded on trucks in the loading dock area of the Waste Processing Building or other locations as appropriate.

The containers are monitored for surface contamination, decontaminated if necessary, and released for transport to a licensed radioactive waste disposal site. The expected curie contents of these shipments are presented in Table 11.4-10. Wastes will be shipped in accordance with applicable NRC, Department of Transportation, and state regulations.

11.4.3 References

1. Kibbey, A.H. and Godbee, H.W., "A Critical Review of Solid Radioactive Waste Practices at Nuclear Power Plants," ORNL-4924, March 1974.
2. Kibbey, A.H., et al., "A Review of Solid Radioactive Waste Practices," NUREG/CR-0144, October 1978.
3. Wild, R.E., et al., "Data Base for Radioactive Waste Management/ Waste Source Options Report," NUREG/CR-1759, November 1981.

4. NWT Corporation, "Solid Radwaste Radionuclide Measurements," EPRI NP-2734, Project 1568-1, November 1982.

5. WasteChem (WPC), "Radwaste Volume Reduction and Solidification System," WPC-VRS-001, Rev. 1, May 1978.

6. WasteChem Topical Report, "10 CFR 61 Waste Form Conformance Program for Solidified Process Waste Products Produced by a WasteChem Corp. Volume Reduction and Solidification System," VRS-002, August 1987.

11.5 PROCESS AND EFFLUENT RADIOLOGICAL MONITORING AND SAMPLING SYSTEMS

11.5.1 Design Bases

The process and effluent radiological monitoring and sampling systems are designed to provide radiation measurements, records, alarms, and/or automatic line isolation required to handle, process, and/or release station liquid and gaseous radioactive effluents, in compliance with the requirements of 10 CFR Parts 20 and 50, General Design Criteria 60, 63 and 64, and Regulatory Guide 1.21. The systems are designed to be in general compliance with the guidelines of Regulatory Guides 4.15 and 1.97.

The systems are designed to continuously monitor and/or sample process and effluent streams wherever a potential for a significant release of radio-activity exists during normal operations, including anticipated operational occurrences, and during postulated accidents. For certain effluent streams for which the potential release of radioactivity is determined to be insignificant relative to the design objectives of 10 CFR 50 Appendix I, process monitoring/sampling and airborne sampling are utilized to conservatively assess the releases.

11.5.1.1 Performance Requirements

The functional performance requirements for the Radiation Monitoring Systems are to:

a. Warn of leakage from process systems containing radioactivity

b. Monitor the amount of radioactivity released in effluents

c. Isolate lines containing liquid and gaseous activity when activity levels reach a preset limit

d. Record the radioactivity present in various station systems and effluent streams

e. Provide a means for leakage detection

f. Provide information on failed fuel.

11.5.1.2 Design Provisions

The components of the process and effluent radiation monitoring and sampling systems are designed for the following environmental conditions:

a. Temperature:	An ambient temperature range of 40°F to 120°F
b. Humidity:	0 to 95 percent relative humidity
c. Pressure:	Components designed for normal atmospheric pressure.

Radiation monitors, utilizing G.M. tubes are of a nonsaturating design so that they register full scale if exposed to radiation levels up to 100 times full-scale indication.

Radiation monitoring equipment is designed and located so that radiation damage to electrical insulation and other materials will not affect their usefulness over the life of the plant. Where possible, electronic components beyond the detector are mounted near the detector in a background radiation level of less than 2.5 mR/hr.

Each radiation monitoring channel is designed so that it can be checked on a daily basis, tested monthly, and recalibrated at refueling intervals.

Access to each of the radiation monitoring channel alarm setpoints is under administrative control.

Process and effluent radiation monitors provide annunciation and indication in the main control room.

Process and effluent monitors continuously monitor radiation levels in the various process streams and effluent release points.

Process and effluent monitors provide instrument failure annunciation in the main control room.

All online process and effluent radiation monitors are implemented with the capability to replace or decontaminate these monitors without opening the process stream or losing the capability to isolate the effluent stream.

This system is non-Class 1E and nonsafety-related, with the exception of the monitors identified in Subsection 11.5.2.1n (Containment Online Purge). The containment online purge monitors are Class 1E, safety-related and supplied from Class 1E uninterruptible power supplies.

The resin sluicing monitors (RM-6560, -6561 and -6564) and the Storm Drainage System monitor (RM-6454) are the only monitors that do not interface with the RDMS host computer system (Subsection 11.5.2.1i)

11.5.2 <u>System Description</u>

Process and effluent radiological monitoring systems consist of multiple channels which monitor radiation levels in various plant operating systems. The digital computer-based Radiation Data Management System (RDMS) consists of local microprocessors for each channel interconnected by redundant communication loops to a redundant (two computers) host computer system. Either of the two computers can provide, by itself, the total computing capacity required for satisfactory operation of the RDMS. The host computer system, in turn, is connected to an operator display/control console in the control room, the health physics checkpoint, the RDMS computer room, the Main Plant Computer System (MPCS) computer room and the hot chemistry lab. The process and effluent radiation monitoring system instrument engineering diagram (Figure 11.5-1) shows an overview of the system, its components and location.

Table 11.5-1 lists the various processes and effluent radiation monitoring channels provided and their pertinent design information, such as detector type, ranges, reference isotopes which the detectors are keyed to, and sensitivities.

Ranges and sensitivities have been selected using the following bases:

a. Maximum calculated concentrations during normal operations, anticipated operational occurrences and postulated accidents

b. State-of-the-art limitations of the commercially available detectors

c. Minimum concentrations that must be detected to permit timely automatic or operator manual responses tabulated in Table 11.5-2 and to avoid exceeding Technical Specification limits.

Shielding is provided on all the monitors to reduce the effect of background radiation, so that the minimum sensitivities specified are met. Table 11.5-2 shows the automatic system response and the operator response to annunciated radioactivity level limits.

A modular assembly with a microprocessor is provided in a locally mounted cabinet for each channel. The assembly converts pulse rate from the detectors to engineering units suitable for indication and recording. The following functions and components are included:

a. Indication

Radiation level is indicated by digital readout. Units are microcuries per cubic centimeter, milliroentgens per hour, counts per minute or roentgens per hour.

b. Alarms

Alarms on (1) rising signal (the setpoint is adjustable to any point on the scale), and (2) loss of signal implying circuit failure are provided.

c. Functional Test and Calibration Requirements

Each radiation monitoring channel has the capability to expose the detector to a radiation check source by energizing a solenoid actuated device. This will cause an up-scale indication verifying the operability of the channel. The check source has a long half-life and an energy emission with the spectra of the radiation being monitored.

For both safety and nonsafety-related monitors, check sources can be activated locally from the RM-23s or the RM-80s. The RDMS Host Computer can initiate check source actuation for nonsafety-related monitors and is prevented from initiating check source actuation of safety-related monitors by a disabling switch at the RM-80s. On high background radiation the RDMS Host Computer will disable check source actuation from the workstations for non-safety related monitors.

Calibration test is accomplished by inputting a pre-calibrated pulse signal to the channel. Reading at the local meter will verify the calibration of the channel.

d. Indicating Lights

Indicating lights at the local radiation monitoring cabinets monitor individual channel high radiation alarms and circuit failures.

e. Power Supplies

Power supplies are mounted at the local radiation monitoring cabinets, and provide the voltages for the modular component circuitry, relays and alarm lights. The power supplies also supply high voltage for the detector. Internal battery backup is provided to prevent loss of stored information in the event of loss of AC power.

f. Fail-Safe

Fail-safe circuits in each monitoring channel indicate channel failure caused by signal or power failure.

11.5.2.1 Channel Descriptions

a. Waste Gas Processing Monitor - Channels 6502, 6503 and 6504

Radiogas monitors are located online at three points within the Radioactive Gaseous Waste System. Monitor 6502 is located up-stream of the carbon delay beds, 6503 is located downstream of the carbon delay beds, and 6504 is located downstream of the waste gas compressors. These monitors serve as indicators of carbon bed performance, with control room annunciation to alert station operators of abnormal operation or conditions. Remote indication and annunciation are provided on the control panel for the Radioactive Gaseous Waste System and in the control room.

A high radiation signal on 6504 terminates waste gas system discharges to the ventilation stack by automatic closure of the waste gas discharge valve.

b. Condenser Air Evacuator System Gas Monitor – Channel 6505

This channel monitors the discharge from the shell-side vacuum pump exhaust header of the condenser for gaseous radioactivity, which is indicative of a primary-to-secondary system leak. During normal plant operation, the gas discharge is routed to the Primary Auxiliary Building exhaust filter system. During startup operation (hogging), the gases removed by the evacuation system are discharged to the atmosphere via the Turbine Building vent. A beta scintillator is used to monitor the gaseous radioactivity level. Remote indication and annunciation are provided locally and in the control room.

c. Primary Component Cooling Liquid Monitors – Channels 6515 and 6516

These two channels continuously monitor trains A and B of the Primary Component Cooling System for radioactivity indicative of a leak from the Reactor Coolant System or one of the other radioactive systems which exchange with the Primary Component Cooling System. Indication and annunciation are provided locally and in the control room.

The gamma scintillation detectors are located in offline liquid samplers.

d. Waste Processing System Liquid Effluent Monitor-Channel 6509

All discharges from the station via the WL Test Tank discharge header are monitored by an online gamma scintillation detector. This includes all discharges from the Test Tank itself, as well as steam generator blowdown demineralizer regenerant solution from the waste holdup sump or the bottom of the demineralizer beds. (See Subsection 10.4.8.2.)

Automatic valve closure action is initiated by this monitor to prevent further release after a high-radiation level is indicated and alarmed. Control room and remote indication and annunciation are provided.

e. Steam Generator Blowdown Liquid Sample Monitor – Channels 6510, 6511, 6512, 6513 and 6519

These channels monitor the liquid phase of the secondary side of the steam generator for radioactivity concentrations, which would indicate a primary-to-secondary system leak, providing backup information to that of the condenser air evacuation system gas monitor. A sample from the bottom of each steam generator is continuously monitored by a scintillation counter mounted in line with an offline type sample assembly.

Monitor 6519 is an offline detector with pumping system which monitors the flash tank discharge.

High activity alarm indications are displayed at the detector location and in the control room.

In the event of a high activity alarm from any monitor, the isolation valve in the blowdown flash tank discharge closes.

f. Reactor Coolant Letdown Gross Activity Monitor – Channels 6520-1, 6520-2

The reactor coolant letdown monitoring system is in service whenever normal letdown is in service and has no automatic functions. This monitor provides indication of primary coolant radioactivity concentration over a wide range of operating conditions and assists in the detection of failed fuel.

The design utilizes an adjacent-to-line detector (Geiger Mueller type) positioned in a shield that provides a collimated view of the letdown line. The detector is placed far enough from the RCS to allow for sufficient N-16 decay. The monitor readout indication is in mR/hr and has a range of 10^{-1} to 10^{4} mR/hr. This detector has a minimum sensitivity of about $1x10^{0}$ μCi/cc in a 15 mR/hr background depending upon the isotope of interest. The detector exhibits good energy linearity for gamma rays between 210 keV and 1333 keV and provides acceptable response to the isotopes listed in Table 11.5-1. The upper detection limit of the monitor is about $1x10^{3}$ μCi/cc depending upon the isotope of interest thereby providing a range of 1 to 1000 μCi/cc.

g. <u>Resin Sluicing Operation Monitors - Channels 6560, 6561 and 6564</u>

Geiger-Mueller tubes clamped to the process pipe monitor the resin sluicing operation. They provide indication and alarm locally and at the waste management panel. They do not interface with the RDMS host computer system. The function of these detectors is to monitor filter failure and to indicate completion of sluicing operation.

h. <u>Main Steam Line Radiation Monitors - Channels 6481-1, 6481-2, 6482-1, and 6482-2</u>

Online gamma-sensitive detectors are located on each main steam line upstream of the safety relief valves. As required by NUREG-0737, these monitors provide a method of quantifying high-level releases of radioactive noble gasses after an accident. Control room indications and alarms are provided.

The monitors display steam line dose rates in "mr/hr." The noble gas release rate is calculated from the dose rate by using a procedure.

i. <u>Plant Vent Monitor - Channels 6528-1, 6528-2, 6528-3, and 6495</u>

The monitoring capability associated with the main plant vent is described in Subsection 12.3.4.

j. <u>Fuel Storage Building Exhaust Monitor – Channel 6562</u>

The monitoring capability associated with air exhaust is described in Subsection 12.3.4.

k. <u>Turbine Building Sump Liquid Radiation Monitor - Channel 6521</u>

This online gamma scintillation detector is located on the Turbine Building sump effluent line. At a pre-determined radioactive concentration, this monitor will alarm and automatically terminate the discharge and isolate the sump.

l. Containment Online Purge Monitor - Channels 6527A, 6527B

These detectors monitor the air exhausted via the containment purge. They utilize GM tubes sensitive to Xe-133. These detectors provide measurement of the activity of the containment purge and provide isolation on a high signal. The detectors and their associated microprocessors are Class 1E. Each monitor utilizes a two-out-of-two detector logic such that two detectors must be in alarm before the monitor initiates an isolation signal.

m. Auxiliary Condensate Monitor - Channel 6490

An offline, skid-mounted monitor draws a sample from the auxiliary condensate return line. In the event that the auxiliary steam should become contaminated, this monitor will automatically terminate the condensate return to the auxiliary steam boiler and isolate the return piping.

n. Storm Drains Monitor - Channel 6454

This is an offline, skid-mounted monitor which continuously samples the storm drainage. The monitor design is identical to the steam generator blowdown liquid sample monitor. An auxiliary pump provides the necessary sample flow. Indication and alarm are provided only locally. This monitor does not interface with the RDMS host computer system. On a high alarm a composite grab sample is automatically obtained. An automatic continuous operating composite sampler is also provided.

o. Water Treatment Liquid Effluent Radiation Monitor – Channel 6473

This online gamma scintillation detector is located on the Water Treatment effluent line that receives water from the Water Treatment Neutralization Tank, Condensate Polishing Low Conductivity Tank or Condensate Polishing Resin Regeneration Megarinse Waste. High activity alarm indications are displayed at the detector location and in the Main Control Room. At a pre-determined radioactive concentration, this monitor will alarm and provide a signal to CPS PLC (1-CPS-CP-563) to automatically terminate the discharge.

11.5.2.2 Alarm Setpoints

The alarm setpoints for the process and effluent radiation monitoring system are provided in Table 11.5-1.

In establishing the site boundary concentration, it is assumed for continuous releases that average annual meteorology exists for gaseous discharges and average annual circulating water flow exists for diluting liquid discharges. For intermittent or off-normal releases, short-term meteorology and actual dilution water flow is assumed for gaseous and liquid discharges, respectively.

11.5.2.3 Design Evaluation

a. The reactor coolant letdown gross activity monitor (RM6520) serves as a failed fuel advisory and provides a function independent of the discharge monitoring system. This monitor is not required for detection of fuel cladding breach (see Appendix 7A, Deviation No. 8). The liquid and gaseous waste discharge monitoring system is employed to maintain surveillance over the release of radioactivity, and is provided with the following features:

1. The check source is operated by command from the display/control console or by command at the remote cabinet.

2. If the reading falls off scale at any time, an indicator visible to the operator in the control room will alarm.

3. Power failure is indicated by its own indicator, and does not alarm as high radiation failure.

b. An evaluation of instrumentation function, relative to monitoring and for controlling release of radioactivity from various plant systems, is discussed below.

1. <u>Liquid and Gas Wastes</u>

For ruptures or leaks in the Waste Processing System, station area monitors and the vent stack monitor (see Subsection 12.3.4) will alarm on an increase in radiation level over a preset level. For cases where leaks are involved, the operator may control activity release by system isolation. For more severe postulated accident cases, such as rupture of a carbon delay bed, activity release is not controlled. The environmental consequences of the postulated accidents are based on no instrument action. For inadvertent releases relative to violation of administrative procedures, monitors provide means for limiting radioactivity release as well as alarming functions. The waste gas vent monitor will trip the flow control valve in the discharge line when the radiation level exceeds a preset level. Where liquid waste releases are involved, the waste processing system liquid discharge monitor trips shut a valve in the liquid waste discharge line when the radiation level in the discharge line exceeds a preset level.

2. <u>Liquid Waste Release Procedure</u>

The release of liquid waste is under administrative control. The normal procedure for discharging liquid waste is:

(a) A batch of waste is collected in one waste or recovery test tank.

(b) The tank is isolated.

(c) The tank contents are recirculated to mix the liquid.

(d) Samples are taken for analysis before release.

(e) If analysis indicates that release can be made within the terms of the operating license, the quantity of activity to be released is recorded on the basis of the liquid volume in the tank and its activity concentration. Release is made when it is determined that the release will be within the operating license.

(f) To release the liquid, an operator must unlock and open the last stop valve in the discharge line (which is normally locked shut); open a second valve, which trips shut automatically on high radiation signal from the monitor (6509); start a test tank pump and establish the normal flow rate using the flow indicator provided; and finally, close the recirculation valve. Liquid is now being discharged.

11.5.2.4 Sampling

Sampling provisions are installed at the locations shown in Table 11.5-3. The samples are drawn from lines or tanks, and transported to the radiochemistry laboratory where the samples are analyzed for radioactivity content.

The sampling program is defined by a series of procedures for obtaining and analyzing representative samples. The administrative and procedural controls are in accordance with Regulatory Guide 4.15 (Position C) and Regulatory Guide 1.21 (Position C). Prior to sampling, large tanks of liquid waste are well mixed to assure uniform distribution of particulate solids. Sample lines are flushed for a sufficient period of time prior to sample extraction in order to remove sediment deposits and air and gas pockets. Sample collection techniques which preclude losses of radionuclides are employed.

Effluent ventilation release points are monitored continuously. Particulate sampling is done isokinetically for major release points.

11.5.2.5 Laboratory Analytical Instrumentation and Capabilities

Samples of process and effluent gases and liquids are analyzed in the laboratory. Laboratory instrumentation includes the appropriate means of detection for alpha and beta analysis, and gamma isotopic analysis.

Sample volume and counting time are chosen to yield the required sensitivities. Corrections are made for sample-detector geometry, sample self-absorption, and other parameters as necessary to assure accuracy. Gross alpha analysis of all liquid effluent samples is performed by liquid scintillation or by direct counting of evaporated deposits.

Gross alpha analysis of air particulate filters is performed by direct counting of the filters.

Alpha isotopic analysis is performed using the silicon surface barrier detector with the multichannel analyzer system, or by liquid scintillation.

Gamma spectrometry is used for isotopic analysis of liquid, gaseous and airborne particulate and iodine samples. A high efficiency, high-resolution HpGe detector is available, in conjunction with a multichannel analyzer, for resolving complex gamma spectra.

Effluent tritium samples are collected by various methods and analyzed by liquid scintillate.

11.5.2.6 Calibration and Maintenance of Effluent Radiation Monitors

A primary calibration is performed on a one-time basis, using typical isotopes of interest to determine proper detector response. Further primary calibrations are not required since the geometry cannot be significantly altered within the sampler. Calibration of samplers is then performed based on a known correlation between the detector responses and multiple secondary standards.

Secondary standard calibrations are performed with radiation sources of known activity. This calibration confirms the channel sensitivity. The secondary standard calibration is performed by placing the secondary standards on the sensitive area of the detector and comparing detector response to the detector response at the time of primary calibration.

The radiation monitoring system channels will be status checked at least daily and calibrated periodically. If a monitor functionally tested quarterly provides a control function on release, it will be functionally tested prior to that release.

Calibration of the indicating channels is performed following any equipment maintenance which could result in reducing the accuracy of the instrument indication. It is also done any time use of the ion chamber precalibrated pulse test signal or the radioactive check source indicates instrument drift.

A burn-in test, operational test and isotopic calibration of the Complete Radiation Monitoring System are performed at the factory. Field calibration after system installation will be performed using calibration sources and their decay curves provided with the system. The sample chambers will be decontaminated in situ periodically and, if required, are easily replaceable.

11.5.3 Effluent Monitoring and Sampling

General Design Criterion 64 requires monitoring of effluent discharge paths. Compliance with requirements is discussed in Subsection 11.5.2.

Airborne radioactivity monitoring is discussed in Subsection 12.3.4.

11.5.4 Process Monitoring and Sampling

Means are provided to control and monitor the release of radioactivity to the environment in accordance with the requirements of General Design Criteria 60 and 63 as discussed in Subsection 11.5.2.

Appendix 11A DATA BASE FOR SEABROOK 10 CFR 50, APPENDIX I (REALISTIC) SOURCE TERM

The analysis and parameters described in this section are historical and are the basis of the facility's 10 CFR 50 Appendix I analysis, utilizing the assumptions and methodology of NUREG-0017 and a core power level of 3654 MWt. For the licensed core power level, the core power level for analysis purposes is 3659 MWt. Since 3659 MWt represents only an approximate 0.1% increase above 3654 MWt, use of the core power level of 3659 MWt in the Appendix I analysis discussed herein will have an insignificant impact on radiological releases.

Continued compliance with the annual dose limits to an individual in an unrestricted area set by 10 CFR 50 Appendix I and 40 CFR 190, resulting from gaseous and liquid effluents released to the environment following operation at the licensed core power level was also demonstrated by utilizing five (5) years (1998-2002) of effluent/dose impact data and using scaling factors to estimate the impact of operation at the licensed core power level.

It is noted that for an operating plant, the actual performance and operation of installed equipment, the reporting of actual offsite releases and doses, and compliance with the regulatory limits of 10 CFR 50 Appendix I, 10 CFR 20, and 40 CFR 190, are controlled by the Offsite Dose Calculation Manual.

The following information is presented to comply with Appendix B of Regulatory Guide 1.112. Ventilation system flow rates used in the gaseous dose analysis are those values in effect at the time of the analysis.

The analytical methods described in NUREG-0017 are extensively used in the source term calculation. Radioactive concentrations in the Primary and Secondary Coolant Systems are evaluated on the basis of a pressurized water reactor (PWR) with recirculating U-tube steam generators. Volatile treatment is applied to control secondary system chemistry. A more detailed description is presented in Section 11.1 of the Updated FSAR.

1. General

a. The maximum core thermal power (MWt) evaluated for safety considerations in the Updated FSAR. (Note: All of the following responses are adjusted to this power level.)

   Response 1a. Thermal power is 3654 MWt

b. The quantity of tritium released in liquid and gaseous effluents (Ci per year).

Response 1b. A total tritium release of 0.4 curies per MWt per year is recommended in NUREG-0017 for a PWR with moderate tritium control. Accordingly, 1462 curies of tritium are expected to be released per year. One-half of the total release is assumed to be through the liquid pathway and one-half through the gaseous pathway.

2. **Primary System**

a. The total mass (lb.) of coolant in the primary system, excluding the pressurizer and primary coolant purification system, at full power.

Response 2a. Total primary coolant mass is 5.05×10^5 lbs.

b. The average primary system letdown rate (gal/min) to the primary coolant purification system.

Response 2b. 80 gal per min (4.01×10^4 lbs/hr)

c. The average flow rate (gal/min) through the primary coolant purification system cation demineralizers. (Note: The letdown rate should include the fraction of time the cation demineralizers are in service.)

Response 2c. Letdown flow through the primary coolant purification system cation demineralizers is used intermittently only when additional purification of the reactor coolant is required. No credit for cation demineralizer cleanup is assumed in the source term calculations.

d. The average shim bleed flow (gal/min)

Response 2d. The shim bleed and other clean recyclable waste (e.g., Primary Drain System) are processed through the Boron Recovery System. A detailed description and operational procedure are presented in Section 3.5 of the Seabrook Station Environmental Report and Subsection 9.3.5 of the Updated FSAR. A schematic flow diagram is shown in Figure 11A-1. The shim bleed is diverted from the normal chemical and volume control flow path (purification letdown) after the stream has been degasified. It has two components (computed on an annualized average) which are:

1) Reactor coolant diverted for boron recovery in the amount of 116 lb/hr (0.23 gpm).

2) Reactor coolant diverted for tritium control in the amount of 194 lb/hr (0.38 gpm).

The shim bleed is then routed to the cesium removal ion exchangers of the Boron Recovery System where it is treated through filtration and evaporation. Provisions for demineralization are included as shown in Figure 11A-1. However, this is an optional pathway used for the recycle mode of operation and as such is not included when calculating plant releases. Equipment leakages and valve stem leak-offs are collected through the primary drain tank in an estimated amount of 300 gpd (0.21 gpm). The primary drain tank inventory is processed through the primary drain tank degasifier and routed to the Boron Recovery System, joining with the shim bleed.

The radioactivity level for shim bleed and primary leakages is the same as the reactor coolant. Flow patterns for these sources are intermittent in nature. A combined flow rate of 4.16×10^2 lbs/hr (0.82 gpm) is estimated on an annual average basis.

To control the tritium level within the Primary Coolant System, 200,000 gallons of reactor coolant is expected to be discharged annually through the Boron Recovery System. Therefore, the release fraction amounts to 46 percent of reactor coolant processed through the BRS annually.

System decontamination factors (DF) are conservatively assumed to be 10^3 for iodines and 10^4 for other nuclides due to evaporation and demineralization.

Holdup time is calculated to be a minimum of 5 days on the basis of the capacities of two boron waste storage tanks (225,000 gals each) and two recovery test tanks (18,000 gals each).

3. **Secondary System**

a. The number and type of steam generators, the type of chemistry used and the carry-over factor used in the evaluation for iodine and nonvolatiles.

Response 3a. Four vertical, recirculating, inverted U-tube steam generators per unit. Volatile chemistry will be used to control secondary side chemistry.

Carry over factors: 1 percent for iodines

0.1 percent for nonvolatiles

b. The total steam flow (lbs/hr) in the secondary system.

Response 3b. 1.514×10^7 lbs/hr.

c. The mass of liquid in each steam generator (lb.) at full power.

Response 3c. 9.55×10^4 lbs.

d. The primary-to-secondary leakage rate (lb/day) used in the evaluation.

Response 3d. 100 lb. per day

e. Description of the Steam Generator Blowdown and Blowdown Purification Systems. The average steam generator blowdown rate (lb/hr) is used in the evaluation.

Response 3e. When primary-to-secondary leakage as described in response 3d exists, the primary method to process radioactive secondary liquid from the steam generators is to direct steam blowdown flash tank bottoms cooler discharge to the floor drain tanks. If no secondary pressure is available, the steam blowdown and wet lay-up pumps can be used. From the floor drain tanks, processing through the installed vendor system (WL-SKD-135) to the waste test tanks is the preferred method (reference Subsection 11.2.2.1). The DF assumed for all radionuclides by this method of treatment is 100. Approximately 30 percent of the blowdown volume flashes to steam in the flash tank. This steam is vented to the number 3 feedwater heater, or main condenser if they are available. With the number 3 feedwater heater or main condenser not available, steam from the flash tank will be directed to the flash tank condenser/cooler and then pumped to the waste test tanks in the Liquid Waste System (see Section 11.2). Further processing by the Liquid Waste System is available, if required, prior to release to the environment via the plant Service and Circulating Water System.

Average steam generator blowdown rate of 75 gpm (3.75×10^4 lbs/hr) is assumed for the analysis. The Steam Generator Blowdown System is described in 10.4.8 of the Updated FSAR. A schematic flow diagram of the Steam Generator Blowdown System is shown in Figure 11A-2.

f. The fraction of the steam generator feedwater processed through the condensate demineralizers and the decontamination factors used in the evaluation for the Condensate Demineralizer System.

Response 3f. Not applicable; Seabrook does not utilize a Condensate Demineralizer System.

g. Condensate demineralizers:

(1) Average flow rate (lb/hr)

(2) Demineralizer type (deep bed or powdered resin)

(3) Number and size (ft^3) of demineralizers

(4) Regeneration or replacement frequency

(5) Indicate whether ultrasonic resin cleaning is used and the waste liquid volume associated with its use

(6) Regenerant (backwash) volume (gal/event) and activity

Response 3g. Not applicable; Seabrook does not utilize a Condensate Demineralizer System.

4. **Liquid Waste Processing Systems**

a. For each Liquid Waste Processing System (including the shim bleed, steam generator blowdown, and detergent waste processing systems), provide the following information in tabular form:

(1) Sources, flow rates (gal/day), and expected activities (fraction of primary coolant activity (PCA) for all inputs to each system).

(2) Capacities of all tanks (gal) and processing equipment (gal/day) considered in calculating holdup times.

(3) Decontamination factors for each processing step.

(4) The fraction of each processing stream expected to be discharged over the life of the plant.

(5) For demineralizer regeneration, the time between regenerations, regenerant volumes and activities, treatment of regenerants, and the fraction of regenerant discharged. Include parameters used in making these determinations.

(6) Liquid source term by radionuclide (in Ci/yr) for normal operation, including anticipated operational occurrences.

Response 4a. Nonrecyclable dirty wastes are collected and processed through the Liquid Waste System. A schematic flow diagram of the Liquid Waste System is shown in Figure 11A-3. Sources are determined according to NUREG-0017.

They are:

Fraction of Primary Coolant

Source	Flow Rate	Activity
Containment Building	40 gpd	1.0
Auxiliary Building Floor Drain	200 gpd	0.1
Laboratory Drains	400 gpd	0.002
Sampling Drains	15 gpd	1.0
Miscellaneous Sources	700 gpd	0.01

These sources and additional sources of liquid radwaste and processing parameters are presented in Table 11A-1.

Liquid source terms by radionuclide (in Ci/yr) for normal operation, including anticipated operational occurrences, are presented in Table 11A-2.

b. Provide piping and instrumentation diagrams and process flow diagrams for the Liquid Radwaste Systems and for all other systems influencing the source term calculations.

Response 4b. Piping and instrumentation diagrams and process flow diagrams for the Liquid Radwaste Systems are provided in Figure 11.2-1 and Figure 11.2-2 of the Updated FSAR.

5. Gaseous Waste Processing System

a. The volume (ft^3/yr) of gases stripped from the primary coolant.

Response 5a. 1.68×10^5 ft^3/yr.

b. A description of the process used to hold up gases stripped from the primary system during normal operations and reactor shutdown. If pressurized storage tanks are used, include a process flow diagram of the system indicating the capacities (ft^3), number, and design and operating storage pressures of the storage tanks.

Response 5b. Not applicable. The Seabrook Gaseous Waste Processing System is described under Item 5e below.

c. A description of the normal operation of the system, e.g., the number of tanks held in reserve for back-to-back shutdown, fill time for tanks. Indicate the minimum holdup time used in the evaluation and the basis for this number.

Response 5c. Not applicable. See Item 5e below.

d. If HEPA filters are used downstream of the pressurized storage tanks, the decontamination factor used in the evaluation.

Response 5d. Not applicable. See Item 5e below.

e. If a charcoal delay system is used, a description of this system indicating the minimum holdup times for each radionuclide considered in the evaluation. List all parameters including mass of charcoal (lb.), flow rate (ft^3/min), operating and dew point temperatures, and dynamic adsorption coefficients for Xe and Kr used in calculating holdup times.

Response 5e. Fission gases from the primary coolant are stripped through the letdown degasifier. Average letdown flow of 80 gpm (4.01×10^4 lbs/hr) is processed through the degasifier with a gas stripping fraction of 1. In addition to the above continuous process, two volumes of primary coolant are assumed to be degassed during cold shutdown. The reactor will operate in a base-load mode. Detailed descriptions are presented in Section 11.3 of the Updated FSAR.

Stripped gases from the primary coolant are processed through the Gaseous Waste Processing System (GWPS) during normal operation and shutdown. A detailed description of the GWPS and operational procedures are given in Section 3.5 of the Seabrook Station Environmental Report and Section 11.3 of the Updated FSAR. A schematic flow diagram is shown in Figure 11A-4. The GWPS consists of chillers, compressors, iodine guard beds, dryers, ambient carbon delay beds and filters. The Ambient Carbon Delay System includes five charcoal delay beds with 1680 lbs. of charcoal in each bed. Design flow rate through the adsorbers is 1.2 scfm. Normal expected flow is 0.8 scfm.

The minimum holdup time used for evaluation/dynamic adsorption coefficients:

Krypton isotopes: 85 hours/45.4 cc per gm. atm.

Xenon isotopes: 60 days/772.5 cc per gm. atm.

Operating and dew point temperatures are ambient (70°F) and -40°F, respectively.

f. Piping and instrumentation diagrams and process flow diagrams for the Gaseous Radwaste Systems and for other systems influencing the source term calculations.

Response 5f. Piping and instrumentation diagrams and process flow diagrams for the Gaseous Radwaste Systems are provided in Figure 11.3-1, Figure 11.3-2, Figure 11.3-3 and Figure 11.3-4 of the Updated FSAR.

6. Ventilation and Exhaust Systems

For each building that houses a steam generator blowdown system vent exhaust, a gaseous waste processing system vent, a main condenser air removal system, or a system that contains radioactive materials, provide the following:

a. Provisions to reduce radioactivity releases through the ventilation or exhaust systems.

Response 6a.1 Primary Auxiliary Building

The primary coolant leak rate to the Auxiliary Building is 160 lb/day. The temperature of the primary coolant in the letdown line as it enters the Auxiliary Building is 290°F. Release of 0.75 percent of the iodine is assumed.

The Auxiliary Building ventilation system pipes all air from potentially contaminated areas through charcoal filters at a flow rate of 36,000 cfm.

Response 6a.2 Waste Processing Building

The Waste Processing Building exhaust air is filtered by HEPA filters prior to release to the environment via the plant vent. No significant releases are anticipated from this building however, and it is not included as a source of gaseous release. Provisions are included in the Waste Processing Building ventilation system for the inclusion of carbon filters, if operational experience and releases indicate that they are required.

Response 6a.3 Turbine Building and Turbine Building Heater Bay Roof Vents

Turbine Building exhaust air is vented directly to the atmosphere, unfiltered, through roof vents.

Response 6a.4 Main Condenser Off-Gas System

Effluent from the main condenser during the normal mode of operation (holding mode) is routed through the Primary Auxiliary Building filter system which contains carbon filters to reduce potential iodine releases. Main condenser effluent during startup operations (hogging mode) is released directly to the atmosphere via the Turbine Building vents.

b. Decontamination factors assumed and the bases (include charcoal adsorbers, HEPA filters, and mechanical devices).

Response 6b. DF of 10 for iodine removal by charcoal adsorbers. DF of 100 for particulate removal by HEPA filtration.

Bases: NUREG-0017

c. Release rates for radioiodine, noble gases, and radioactive particulates and their bases.

Response 6c. See Table 11A-3

Bases: NUREG-0017 and PWR-Gale code:

d. Description of the release points, including height above grade, height above and location relative to adjacent structures, expected average temperature difference between gaseous effluents and ambient air, flow rate, exit velocity, and size and shape of flow orifice.

Response 6d. See Table 11A-4

e. For the Containment Building, the building free volume (ft^3) and a thorough description of the internal recirculation system (if provided), including the recirculation rate, charcoal bed depth, operating time assumed, and mixing efficiency. Indicate the expected purge and venting frequencies and duration and the continuous purge rate (if used).

Response 6e The containment free air volume used for the analysis is 2.704×10^6 ft^3.

The atmosphere inside the Containment is circulated through charcoal filters with a 4" bed depth and 90 percent efficiency for 16 hours prior to personnel entry or purge. A mixing efficiency of 70 percent is used. The recirculation flow is 4,000 cfm. A detailed description of the Containment internal recirculation system is given in Updated FSAR Subsection 9.4.5.

Experience with operating PWRs indicates a purge frequency of 4/year, during shutdown for a duration of 24 hours per purge. The purge flow of 15,000 cfm is filtered through 4" deep charcoal filter beds with iodine removal efficiency of 90 percent. Primary coolant leakage is assumed to be 1 percent per day for noble gases and 0.001 percent per day for iodine. An online purge system is available for use during power operation. The continuous purge rate used to evaluate plant releases is 1000 scfm, and is filtered through 4" deep charcoal filter beds with iodine removal efficiency of 90 percent.

SEABROOK STATION
UPDATED FINAL SAFETY ANALYSIS REPORT

CHAPTER 11
RADIOACTIVE WASTE MANAGEMENT

TABLES

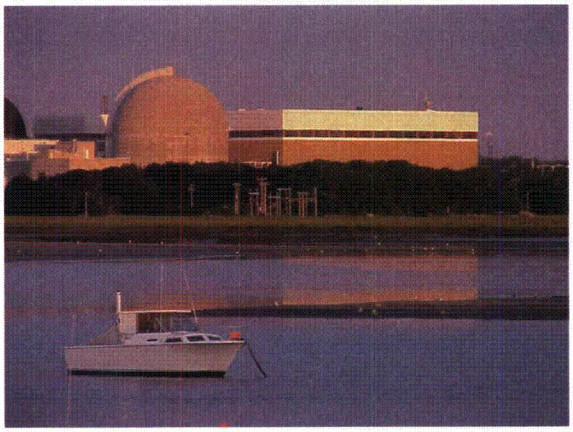


TABLE 11.1-1 REACTOR COOLANT RADIONUCLIDE CONCENTRATIONS

	Concentration (μCi/gm)		
Radionuclide	0.12% Clad Defects [Historical]	1% Clad Defects	0.25% Clad Defects [Historical]
H - 3	1.00E+00*	-	5.0E+00
N - 16	4.00E+01	-	-
I - 130	2.10E-03	5.6E-02	-
I - 131	2.70E-01	2.5E+00	6.3E-01
I - 132	1.00E-01	1.0E+00	2.3E-01
I - 133	3.99E-01	3.8E+00	1.0E+00
I - 134	4.70E-02	5.7E-01	1.5E-01
I - 135	1.90E-01	2.2E+00	5.5E-01
Kr - 83m	2.03E-02	3.7E-01	1.1E-01
Kr - 85m	8.61E-02	1.6E+00	4.3E-01
Kr - 85	2.03E-03	5.1E-01	3.3E-02
Kr - 87	6.14E-02	9.3E-01	3.3E-01
Kr - 88	1.78E-01	2.8E+00	8.5E-01
Kr - 89	5.82E-03	-	-
Xe - 131m	5.29E-03	4.6E-01	1.7E-02
Xe - 133m	3.83E-02	1.8E+00	1.4E-01
Xe - 133	1.59E+00	3.8E+00	6.3E+00
Xe - 135m	1.48E-02	4.7E-01	2.1E-01
Xe - 135	2.02E-01	7.5E+00	7.8E-01
Xe - 137	1.05E-02	-	4.3E-02
Xe - 138	4.99E-02	5.9E-01	1.8E-01
Br - 83	4.80E-03	7.3E-02	-
Rb - 86	8.50E-05	5.7E-01	-
Sr - 89	3.50E-04	2.5E-03	1.0E-03
Sr - 90	1.00E-05	1.6E-04	4.5E-05
Sr - 91	6.50E-04*	1.3E-03	7.8E-03

* $1.00E+00 = 1.00 \times 10^{0}$

	Concentration (μCi/gm)		
Radionuclide	0.12% Clad Defects [Historical]	1% Clad Defects	0.25% Clad Defects [Historical]
Y - 90	1.20E-06	2.4E-04	5.5E-05
Y - 91	6.40E-05	5.8E-03	1.5E-03
Y - 92	-	7.9E-04	2.5E-04
Zr - 95	6.00E-05	5.5E-04	1.7E-04
Nb - 95	5.00E-05	5.6E-04	1.7E-04
Mo - 99	8.4E-02	3.8E+00	8.3E-01
Tc - 99m	4.80E-02	2.3E+00	-
Te - 127m	2.80E-04	2.7E-03	-
Te - 127	8.50E-04	1.2E-02	-
Te - 129m	1.40E-03	9.3E-03	-
Te - 129	1.60E-03*	1.3E-02	-
Te - 131m	2.50E-03	2.2E-02	-
Te - 132	1.7E-02	2.5E-01	6.5E-02
Cs - 134	2.5E-02	6.9E+00	1.1E-01
Cs - 136	1.3E-02	1.6E+00	5.5E-02
Cs - 137	1.8E-02	4.1E+00	5.5E-01
Ba - 137m	1.6E-02	3.9E+00	-
Ba - 140	2.2E-04	3.4E-03	1.1E-03
La - 140	1.5E-04	1.1E-03	3.5E-04
Ce - 144	3.3E-05	4.0E-04	1.1E-04
Np - 239	1.2E-03	-	-
All Others	2.5E-01	-	-

Note: With the exception of the Primary Coolant System source term based on 1% failed fuel, the analysis and parameters described in this table are historical and are the basis of the 10CFR Part 50 Appendix I program. It is noted that for an operating plant, the actual performance and operation of installed equipment, the reporting of offsite releases and doses and compliance with the regulatory limits of 10CFR50 Appendix I, 10CFR20, and 40CFR190 are controlled by the Offsite Dose Calculation Manual.

* $6.50E\text{-}04 = 6.50 \times 10^{-4}$

	Concentration (μCi/gm)		
Radionuclide	0.12% Clad Defects [Historical]	1% Clad Defects	0.25% Clad Defects [Historical]
Sr-92	-	6.5E-04	-
Y-93	-	5.4E-04	-
Zr-97	-	3.6E-04	-
Ru-103	-	4.8E-04	-
Ru-105	-	1.1E-04	-
Ru-106	-	1.6E-04	-
Rh-105	-	2.6E-04	-
Ba-139	-	5.1E-04	-
La-141	-	2.5E-04	-
La-142	-	8.0E-05	-
Ce-141	-	5.4E-04	-
Ce-143	-	4.0E-04	-
Pr-143	-	4.9E-04	-
Cs-138	-	8.9E-01	-
Cs-134m	-	4.4E-02	-
Rb-88	-	2.8E+00	-
Rb-89	-	7.3E-02	-
Te-131	-	1.3E-02	-
Te-133	-	7.6E-03	-
Te-134	-	2.6E-02	-
Te-125m	-	5.8E-04	-
Te-133m	-	1.5E-02	-
Ba-141	-	1.2E-04	-
Rh-106	-	1.6E-04	-

	Concentration (μCi/gm)		
Radionuclide	0.12% Clad Defects [Historical]	1% Clad Defects	0.25% Clad Defects [Historical]
Rh-103m	-	4.7E-04	-
Tc-101	-	1.8E-02	-
La-143	-	1.3E-05	-
Nb-97	-	8.4E-05	-
Nb-95m	-	4.0E-06	-
Pr-144	-	4.0E-04	-
Pr-144m	-	7.2E-06	-
Y-94	-	1.8E-05	-
Y-95	-	1.1E-05	-
Y-91m	-	7.7E-04	-
Br-82	-	1.2E-02	-
Br-84	-	3.4E-02	-

Corrosion Products

Radionuclide	Concentration μCi/gm**
Mn -54	1.1E-03
Mn -56	-
Co -58	3.3E-03
Co -60	3.8E-04
Fe -59	2.2E-04
Cr -51	2.2E-03
Fe -55	8.6E-04

** Corrosion product activities based on values given in Table 2-12, NUREG-0017, April 1976

TABLE 11.1-2 PARAMETERS USED IN THE CALCULATION OF REACTOR COOLANT FISSION AND CORROSION PRODUCT ACTIVITIES

1.	Ultimate core thermal power, MWt		3,659
2.	Clad defects, as a percent of rated core thermal power being generated by rods with clad defects		1.0
3.	Reactor coolant liquid volume, lbm		492,200
4.	Reactor coolant full power average temperature, °F		571.0-589.1
5.	Purification flow rate (normal) gpm		120
6.	Effective cation demineralizer flow, gpm		7.5
7.	Fission product escape rate coefficients*		
	a.	Noble gas isotopes, sec^{-1}	$6.5x10^{-8}$
	b.	Br, Rb, I and Cs isotopes, sec^{-1}	$1.3x10^{-8}$
	c.	Te, Se, Tc, Sn and Sb isotopes, sec^{-1}	$1.0x10^{-9}$
	d.	Mo isotopes, sec^{-1}	$2.0x10^{-9}$
	e.	Sr and Ba isotopes, sec^{-1}	$1.0x10^{-11}$
	f.	Y, Zr, Nb, Ru, Rh, La, Ce, Pr, Nd and Pm	$1.6x10^{-12}$
8.	Mixed bed demineralizer decontamination factors:		
	a.	Noble gases and Cs, Y and Mo	1.0
	b.	All other isotopes including corrosion products	10.0
9.	Cation bed demineralizer decontamination factor for Cs,		10.0
10.	Degasifier noble gas stripping fractions:		
	a.	Kr83m	0.73
	b.	Kr85	$2.3x10^{-1}$
	c.	Kr85m	$2.90x10^{-1}$
	d.	Kr87	$6.00x10^{-1}$
	e.	Kr88	$4.30x10^{-1}$
	f.	Xe131m	$2.50x10^{-1}$
	g.	Xe133	$2.50x10^{-1}$
	h.	Xe133m	$2.60x10^{-1}$
	i.	Xe135	$2.80x10^{-1}$
	j.	Xe135m	$8.00x10^{-1}$
	k.	Xe138	1.0

* Escape rate coefficients are based on fuel defect tests performed at the Saxton reactor. Experience at two plants operating with fuel rod defects has verified the listed escape rate coefficients.

TABLE 11.1-3 PRINCIPAL PARAMETERS USED IN ESTIMATING REALISTIC RELEASES OF RADIOACTIVE MATERIAL IN EFFLUENTS FROM SEABROOK [historical]

Reactor Power level, megawatts thermal	3,654
Plant Capacity Factor	0.80
Operating Power Fission Product Source Term, percent clad defects	0.12
Primary System	
Mass of coolant, pounds	505,000
Letdown Rate to chemical and Volume Control System, lbs/hr	4.01×10^4
Shim bleed rate and primary system equipment leakage, lbs/hr	4.16×10^2
leakage rate to secondary system, pounds per day	100
leakage rate to auxiliary area, pounds per day	160
Frequency of degassing (cold shutdown), times per year	2
Secondary System	
Steam flow rate, pounds per hour	15.14×10^6
Mass of steam in each generator, pounds	5,700
Mass of liquid in each generator, pounds	95,500
Mass of secondary coolant, pounds	1.8×10^6
Rate of steam leakage to Turbine Building, pounds per hour	1,700
Containment Building Volume, cubic feet	2.715×10^6
Frequency of Containment Purges, times per year	4 (refueling and maintenance)
Continuous Ventilation Rate, ft^3/min	1,000
Turbine Building Leak Rate, gallons per day	7,200

Iodine Partition Factors	
Steam generator internal partition	0.01
Primary coolant leak to auxiliary area	0.0075
Condenser/vacuum pump (volatile species)	0.15
Iodine Decontamination Factor for Ventilation Systems Charcoal absorbers	10
Particulate Decontamination Factors for Ventilation System HEPA Filters	100

Liquid Waste Processing Systems

System	Input Flow Rate, gallons per day	Decontamination Factors Iodine	Cesium, Rubidium	Others
Miscellaneous Waste	1360	10^3	10^4	10^4
Equipment Drain	302	10^3	$2x10^3$	10^4
Turbine Building Sump Waste	7200	1	1	1
Boron Recovery	878	10^3	$2x10^3$	10^4
Steam Generator Blowdown (During a primary-to-secondary leak)	$1.1x10^5$	10^2	10^2	10^2

Note: The analysis and parameters described in this table are historical and are the basis of the 10 CFR Part50 Appendix I program. It is noted that for an operating plant, the actual performance and operation of installed equipment, the reporting of offsite releases and doses, and compliance with the regulatory limits of 10CFR50 Appendix I, 10CFR20, and 40CFR190 are controlled by the Offsite Dose Calculation Manual.

TABLE 11.1-4 STEAM GENERATOR SECONDARY SIDE EQUILIBRIUM RADIONUCLIDE CONCENTRATIONS [historical]

	Concentration (μCi/gm)	
Radionuclide Defects	Expected Values(1) 0.12% Clad Defects	Design Values(2) 0.25% Clad Defects
A. **Water**		
H-3	1.00E-03*	1.4E-03
N-16	1.00E-06	-
I-130	1.45E-07	-
I-131	3.33E-05	1.6E-04
I-132	1.04E-05	1.1E-05
I-133	3.51E-05	1.8E-04
I-134	6.12E-07	2.9E-06
I-135	1.01E-05	5.8E-05
Br-83	1.41E-07	-
Rb-86	1.02E-08	-
Sr-89	4.91E-08	3.1E-07
Sr-90	1.20E-09	9.4E-09
Sr-91	3.81E-08	1.0E-06
Y-90	6.66E-10	9.9E-09
Y-91	7.34E-09	4.1E-07
Y-92	-	1.6E-08
Zr-95	7.33E-09	4.5E-08
Nb-95	7.43E-09	4.9E-08
Mo-99	9.88E-06	2.0E-05
Tc-99m	2.24E-05	-
Te-127m	2.18E-08	-

* 1.00E-03 = 1.00×10^{-3}

	Concentration (μCi/gm)	
Radionuclide Defects	Expected Values(1) 0.12% Clad Defects	Design Values(2) 0.25% Clad Defects
Te-127	1.29E-07	-
Te-129m	1.49E-07	-
Te-129	6.28E-07	-
Te-131m	2.09E-07	-
Te-132	2.54E-06	1.6E-05
Cs-134	2.88E-06	2.8E-05
Cs-136	1.30E-06	1.4E-05
Cs-137	1.92E-06	1.3E-04
Ba-140	2.35E-08	2.4E-07
La-140	3.04E-08	7.0E-08
Ce-144	4.82E-09	2.9E-08
Mn-54	4.82E-08	2.6E-07
Mn-56	-	2.0E-06
Co-58	1.71E-06	8.8E-06
Co-60	2.16E-07	2.6E-07
Fe-59	1.23E-07	3.5E-08
Cr-51	2.00E-07	3.3E-07
Fe-55	1.68E-07	-
Np-239	1.03E-07	-
All others	1.1E-05	-
B. Steam		
H-3	1.00E-03	1.4E-03
N-16	1.00E-07	-
I-130	1.45E-09	-
I-131	3.33E-07	1.6E-06

Radionuclide Defects	Concentration (μCi/gm) Expected Values[1] 0.12% Clad Defects	Design Values[2] 0.25% Clad Defects
I-132	1.04E-07	1.1E-07
I-133	3.51E-07	1.8E-06
I-134	6.12E-09	2.9E-08
I-135	1.01E-07	5.8E-07
Kr-83m	5.56E-09	5.1E-08
Kr-85m	2.41E-08	2.0E-07
Kr-85	5.63E-10	1.5E-08
Kr-87	1.62E-08	1.5E-07
Kr-88	4.85E-08	3.9E-07
Kr-89	1.61E-09	-
Xe-131m	1.48E-09	7.8E-09
Xe-133m	1.07E-08	6.4E-08
Xe-133	4.37E-07	2.9E-06
Xe-135m	4.06E-09	9.7E-08
Xe-135	5.54E-08	3.6E-07
Xe-137	2.88E-09	2.0E-08
Xe-138	1.35E-08	8.3E-08
Br-83	1.41E-09	-
Rb-86	1.02E-11	-
Sr-89	4.91E-12	3.1E-10
Sr-90	1.20E-12	9.4E-12
Sr-91	3.81E-11	1.0E-09
Y-90	6.66E-13	9.9E-13
Y-91	7.34E-12	4.1E-11
Y-92	-	1.6E-12

	Concentration (μCi/gm)	
Radionuclide Defects	Expected Values(1) 0.12% Clad Defects	Design Values(2) 0.25% Clad Defects
Zr-95	7.33E-12	4.5E-11
Nb-95	7.43E-12	4.9E-11
Mo-99	9.88E-09	2.0E-08
Te-99m	2.24E-08	-
Te-127m	2.18E-11	-
Te-127	1.29E-10	-
Te-129m	1.49E-10	-
Te-129	6.28E-10	-
Te-131m	2.09E-10	-
Te-132	2.54E-09	1.6E-08
Cs-134	2.88E-09	-
Cs-136	1.30E-09	-
Cs-137	1.92E-09	1.3E-08
Ba-140	2.35E-11	2.4E-10
La-140	3.09E-11	7.0E-11
Ce-144	4.82E-12	2.9E-11
Mn-54	4.82E-11	2.6E-10
Mn-56	-	2.0E-09
Co-58	1.71E-09	8.8E-09
Co-60	2.16E-10	2.6E-10
Fe-59	1.23E-10	3.5E-11
Cr-51	2.00E-10	3.3E-10
Fe-55	1.68E-10	-
Np-239	1.03E-10	-
All Others	1.1E-08	-

Radionuclide Defects	Concentration (μCi/gm) Expected Values[1] 0.12% Clad Defects	Design Values[2] 0.25% Clad Defects

Notes

(1) Bases: 0.12% clad defects

100 lbs/day primary-to-secondary leak rate

75 gpm steam generator blow down rate

95,500 lbm per steam generator

3,654 MWt

0.1% moisture carryover for nonvolatiles

1.0% moisture carryover for halogens

(2) Bases: 0.25% clad defects

20 gal/day primary-to-secondary leak rate

50 gpm steam generator blowdown rate

97,000 lbm per steam generator

3,654 MWt

0.25% moisture carryover for nonvolatiles

1.0% moisture carryover for halogens

Note: The analysis and parameters described in this table are historical and are the basis of the 10CFR Part50 Appendix I program. It is noted that for an operating plant, the actual performance and operation of installed equipment, the reporting of offsite releases and doses, and compliance with the regulatory limits of 10CFR50 Appendix I, 10CFR20, and 40CFR190 are controlled by the Offsite Dose Calculation Manual.

TABLE 11.1-5 SECONDARY SYSTEM CONDENSATE RADIONUCLIDE CONCENTRATIONS [historical]

	Concentration (μCi/gm)	
Radionuclide	Expected Values[(1)] 0.12% Clad Defects	Design Values[(2)] 0.25% Clad Defects
H-3	1.00E-03*	-
N-16	1.00E-07	-
I-130	1.45E-09	-
I-131	3.33E-07	1.6E-06
I-132	1.04E-07	1.1E-07
I-133	3.51E-07	1.8E-06
I-134	6.12E-09	2.9E-08
I-135	1.01E-07	5.8E-07
Br-83	1.41E-09	-
Rb-86	1.02E-11	-
Sr-89	4.91E-11	7.8E-10
Sr-90	1.20E-12	2.4E-11
Sr-91	3.81E-11	2.6E-09
Y-90	6.66E-13	2.5E-12
Y-91	7.34E-12	1.0E-10
Y-92	-	-
Zr-95	7.33E-12	1.1E-10
Nb-95	7.43E-12	1.2E-10
Mo-99	9.88E-09	5.0E-08
Tc-99m	2.24E-08	-
Te-127m	2.18E-11	-
Te-127	1.29E-10	-

(1) Note 1 of Table 11.1-4 (Sheet 3 of 3)
(2) Note 2 of Table 11.1-4 (Sheet 3 of 3)
* 1.00E-03 = 1.00×10^{-3}

	Concentration (μCi/gm)	
Radionuclide	Expected Values[1] 0.12% Clad Defects	Design Values[2] 0.25% Clad Defects
Te-129m	1.49E-10	-
Te-129	6.28E-10	-
Te-131m	2.09E-10	-
Te-132	2.54E-09	4.00E-08
Cs-134	2.88E-09	-
Cs-136	1.30E-09	-
Cs-137	1.92E-09	3.3E-08
Ba-140	2.35E-11	7.5E-10
La-140	3.09E-11	1.8E-10
Ce-144	4.82E-12	7.3E-11
Mn-54	4.82E-11	6.5E-10
Mn-56	-	5.0E-09
Co-58	1.71E-09	2.2E-08
Co-60	2.16E-10	6.5E-10
Fe-59	1.23E-10	8.8E-11
Cr-51	2.00E-10	8.3E-10
Fe-55	1.68E-10	-
Np-239	1.03E-10	-

Note: The analysis and parameters described in this table are historical and are the basis of the 10CFR Part50 Appendix I program. It is noted that for an operating plant, the actual performance and operation of installed equipment, the reporting of offsite releases and doses, and compliance with the regulatory limits of 10CFR50 Appendix I, 10CFR20, and 40CFR190 are controlled by the Offsite Dose Calculation Manual.

TABLE 11.2-1 LIQUID WASTE SYSTEM COMPONENTS

Floor Drain Tank

Number	2
Capacity each	10,000 gal.
Material	304 SS
Design Pressure	1 psig
Operating Pressure	Atmospheric
Design Temperature	250°F
Operating Temperature	Ambient
Design Code	ASME Sect. VIII, Div. 1

Floor Drain Tank Pump

Number	2
Design Flow	50 gpm
Design TDH	190 ft
Material	316 SS
Design Pressure	150 psig
Design Temperature	200°F
Design Code	Manufacturer's Standards
Type	Centrifugal frame-mounted
Pump Seals	Double Mechanical

Floor Drain Filter

Number	2
Capacity	50 gpm
Material	304 SS
Retention for 25 micron particles	98%
Design Pressure	200 psig

Design Temperature	250°F
Design Code	ASME Sect. VIII, Div. 1
Type	Cartridge

Duplex Strainer

Number	1
Capacity (Design)	120 gpm
Perforation Size	$\frac{1}{16}$ inch
Design Pressure	150 psig
Design Temperature	200°F
Material	316 SS
Design Code	Manufacturer's Standards
Type	Cartridge

Evaporator, Evaporator Distillate Condenser,
Evaporator Distillate Cooler

Same data as on BRS (Subsection 9.3.5)

Evaporator Distillate Pump

Number	1
Design Flow	30 gpm
Design TDH	170 ft
Material	316 SS
Design Pressure	150 psig
Design Temperature	300°F
Design Code	Manufacturer's Standards
Type	Centrifugal frame-mounted
Pump Seals	Single Mechanical

Evaporator Bottoms Pump

Number	1
Design Flow	15 gpm
Design TDH	40 ft
Material	Goulds Alloy 20
Design Pressure	150 psig
Design Temperature	300°F
Design Code	Manufacturer's Standards
Type	Centrifugal frame-mounted
Pump Seals	Double Mechanical

Evaporator Bottoms Cooler

Number	1	
Design Code	ASME Sect. VIII, Div.1	
	Shell Side	Tube Side.
Design Temperature, °F	200	300
Design Pressure, psig	150	150
Operating Pressure, psig	85	50
Design Flow, gpm	47	15
Fluid	PCCW	Concentrate
Temperature in, °F	85	252
Temperature out, °F	105	180
Material	Carbon Steel	Incoloy 825

Waste Test Tanks

Number	2
Capacity	25,000 gal.
Design Pressure	1 psig

Operating Pressure	Atmospheric
Design Temperature	200°F
Operating Temperature	Ambient to 120°F
Material	304 SS
Design Code	ASME Sect. VIII, Div. 1

<u>Waste Test Tank Pumps</u>

Number	2
Material	316 SS
Design Flow	150 gpm
Design TDH	130 ft
Design Pressure	150 psig
Design Temperature	200°F
Design Code	Manufacturer's Standards
Type	Centrifugal frame-mounted
Pumps Seals	Single Mechanical

<u>Waste Demineralizer</u>

Number	1
Media Volume	75 ft^3
Material	304 SS
Design Pressure	150 psig
Design Temperature	300°F
Design Flow	150 gpm
Design Code	ASME Sect. VIII, Div. 1

<u>Waste Demineralizer Filter</u>

Number	1
Retention of 25 micron particles	98%
Material	304 SS

Design Pressure	150 psig
Design Temperature	200°F
Design Flow	150 gpm
Design Code	ASME Sect. VIII, Div. 1
Type	Cartridge

<u>Piping and Valves</u>

Material	316 SS, Al/Br, Cement-lined CS
Design Code	ANSI B31.1
Safety Class	NNS

<u>Liquid Waste Chemical Addition Pump</u>

Number	1
Design Flow	.75 gpm
Design TDH	40 psi
Material	316 SS
Design Pressure	300 psig
Design Temperature	130°F
Design Code	Manufacturer's Standards
Type	Positive Displacement, gear-type
Pump Seals	Magnetic coupling, no shaft seal

TABLE 11.2-2 BORON RECOVERY SYSTEM RELEASES [historical]

Radionuclide	Annual Release (Ci/Year)
I-131	1.57E-02*
I-132	4.00E-05
I-133	8.00E-05
Rb-86	1.00E-05
Sr-89	1.00E-05
Mo-99	9.00E-05
Tc-99m	8.00E-05
Te-127m	1.00E-05
Te-127	1.00E-05
Te-129m	2.00E-05
Te-129	1.00E-05
Te-132	4.00E-05
Cs-134	6.33E-03
Cs-136	7.60E-04
Cs-137	4.78E-03
Ba-137m	4.48E-03
Cr-51	3.00E-05
Mn-54	1.00E-05
Fe-55	4.00E-05
Fe-59	2.00E-05
Co-58	3.20E-04
Co-60	5.00E-05
All Others	1.00E-05
TOTAL (Except Tritium)	3.30E-02

Note: The information presented in the above table represents the inputs/assumptions/results utilized in the Seabrook 10 CFR 50 Appendix I analysis (which is part of the original License Application), and is retained here for historical purposes. For the licensed core power level, the core power level for analysis purposes is 3659 MWt. Updating the Appendix I analysis addressed herein using the core power level of 3659 MWt (which includes margin for power uncertainty) will have an insignificant impact on the calculated effluent releases/doses, since 3659 MWt represents only an approximate 0.1% increase from the core power of 3654 MWt utilized in the original analysis.

* 1.57E-02 = 1.57×10^{-2}

TABLE 11.2-3 NONRECYCLABLE RELEASES FROM LIQUID WASTE SYSTEM [historical]

Radionuclide	Annual Release (Ci/Year)
I-130	2.00E-05*
I-131	2.38E-02
I-132	5.20E-04
I-133	7.70E-03
I-135	8.40E-04
Mo-99	4.80E-04
Tc-99m	4.50E-04
Te-129m	2.00E-05
Te-129	1.00E-05
Te-131m	1.00E-05
Te-132	1.70E-04
Cs-134	2.90E-04
Cs-136	1.30E-04
Cs-137	2.10E-04
Ba-137m	1.90E-04
Cr-51	2.00E-05
Fe-55	2.00E-05
Fe-59	1.00E-05
Co-58	1.80E-04
Co-60	2.00E-05
Np-239	1.00E-05
All Others	1.00E-05
TOTAL (Except Tritium)	3.50E-02

Note: The information presented in the above table represents the inputs/assumptions/results utilized in the Seabrook 10 CFR 50 Appendix I analysis (which is part of the original License Application), and is retained here for historical purposes. For the licensed core power level, the core power level for analysis purposes is 3659 MWt. Updating the Appendix I analysis addressed herein using the core power level of 3659 MWt (which includes margin for power uncertainty) will have an insignificant impact on the calculated effluent releases/doses, since 3659 MWt represents only an approximate 0.1% increase from the core power of 3654 MWt utilized in the original analysis.

* 2.00E-05 = 2.00×10^{-5}

TABLE 11.2-4 STEAM GENERATOR BLOWDOWN SYSTEM RELEASES [HISTORICAL]**

Radionuclide	Annual Release (Ci/Year)
I-131	1.10E-3*
I-132	3.22E-3
I-133	3.23E-3
I-134	3.14E-3
I-135	5.27E-3
Mo-99	2.41E-4
Tc-99m	1.27E-4
Te-132	6.33E-5
Cs-134	3.65E-4
Cs-136	4.44E-5
Cs-137	4.86E-4
Ba-137m	9.68E-4
All Others	Negligible
TOTAL (Except Tritium)	1.83E-2

Note: The information presented in the above table represents the inputs/assumptions/results utilized in the Seabrook 10 CFR 50 Appendix I analysis (which is part of the original License Application), and is retained here for historical purposes. For the licensed core power level, the core power level for analysis purposes is 3659 MWt. Updating the Appendix I analysis addressed herein using the core power level of 3659 MWt (which includes margin for power uncertainty) will have an insignificant impact on the calculated effluent releases/doses, since 3659 MWt represents only an approximate 0.1% increase from the core power of 3654 MWt utilized in the original analysis.

** Bases:

- Primary coolant activity per USNRC PWR Gale Code, Rev. 1 (1986)
- 100 lbm/day primary-to-secondary leakage
- 75 gpm blowdown processed
- WL DF value of 100 for listed radionuclides

* 1.10E-3 = 1.10×10^{-3}

TABLE 11.2-5 NORMAL SECONDARY SYSTEM CONDENSATE LEAKAGE RELEASES [HISTORICAL]

Radionuclide	Annual Release (Ci/Year)
I-130	1.00E-05*
I-131	2.93E-03
I-132	1.90E-04
I-133	2.53E-03
I-135	5.40E-04
Mo-99	9.00E-05
Tc-99m	1.50E-04
Te-132	2.00E-05
Cs-134	3.00E-05
Cs-136	1.00E-05
Cs-137	2.00E-05
Ba-137m	2.00E-05
Co-58	2.00E-05
TOTAL (Except Tritium)	6.58E-03

Note: The information presented in the above table represents the inputs/assumptions/results utilized in the Seabrook 10 CFR 50 Appendix I analysis (which is part of the original License Application), and is retained here for historical purposes. For the licensed core power level, the core power level for analysis purposes is 3659 MWt. Updating the Appendix I analysis addressed herein using the core power level of 3659 MWt (which includes margin for power uncertainty) will have an insignificant impact on the calculated effluent releases/doses, since 3659 MWt represents only an approximate 0.1% increase from the core power of 3654 MWt utilized in the original analysis.

* $1.00E\text{-}05 = 1.00 \times 10^{-5}$

TABLE 11.2-6 SUMMARY OF NORMAL RADIOACTIVE LIQUID RELEASES [HISTORICAL]

Radionuclide	Annual Release (Ci/Year)
I-130	4.00E-05*
I-131	4.40E-02
I-132	1.30E-04
I-133	1.19E-02
I-134	3.00E-05
I-135	1.91E-03
Br-83	1.00E-05
Rb-86	1.00E-05
Sr-89	1.00E-05
Mo-99	7.10E-04
Tc-99m	8.00E-04
Te-127m	1.00E-05
Te-127	1.00E-05
Te-129m	4.00E-05
Te-129	3.00E-05
Te-131m	1.00E-05
Te-132	2.40E-04
Cs-134	6.66E-03
Cs-136	9.10E-04
Cs-137	5.02E-03
Ba-137m	4.74E-03
Cr-51	5.00E-05
Mn-54	1.00E-05
Fe-55	6.00E-05
Fe-59	3.00E-05
Co-58	5.20E-04
Co-60	8.00E-05
Np-239	1.00E-05
All Others	2.00E-05
Total (Except Tritium)	7.92E-02

* 4.00E-05 = 4.00×10^{-5}

Note: The information presented in the above table represents the inputs/assumptions/results utilized in the Seabrook 10 CFR 50 Appendix I analysis (which is part of the original License Application), and is retained here for historical purposes. For the licensed core power level, the core power level for analysis purposes is 3659 MWt. Updating the Appendix I analysis addressed herein using the core power level of 3659 MWt (which includes margin for power uncertainty) will have an insignificant impact on the calculated effluent releases/doses, since 3659 MWt represents only an approximate 0.1% increase from the core power of 3654 MWt utilized in the original analysis.

TABLE 11.2-7 RADIOACTIVE LIQUID RELEASES DUE TO ANTICIPATED OPERATIONAL OCCURRENCES [HISTORICAL]

Radionuclide	Annual Release (Ci/Year)
I-130	7.57E-05*
I-131	8.33E-02
I-132	2.46E-03
I-133	2.26E-02
I-134	5.68E-05
I-135	3.62E-03
Br-83	1.89E-05
Br-86	1.89E-05
Sr-89	1.89E-05
Mo-99	1.34E-03
Tc-99m	1.51E-03
Te-127m	1.89E-05
Te-127	1.89E-05
Te-129m	7.57E-05
Te-129	5.68E-05
Te-131m	1.89E-05
Te-132	4.54E-04
Cs-134	1.26E-02
Cs-136	1.72E-03
Cs-137	9.51E-03
Ba-137m	8.98E-03
Ba-140	1.00E-05
La-140	1.00E-05
Cr-51	2.84E-04
Mn-54	7.57E-05
Fe-55	3.60E-04
Fe-59	1.70E-04
Co-58	2.97E-03
Co-60	4.36E-04
Np-239	5.68E-05
All Others	3.79E-05
TOTAL (Except Tritium)	1.50E-01

* 7.57E-05 = 7.57x10-5

Note: The information presented in the above table represents the inputs/assumptions/results utilized in the Seabrook 10 CFR 50 Appendix I analysis (which is part of the original License Application), and is retained here for historical purposes. For the licensed core power level, the core power level for analysis purposes is 3659 MWt. Updating the Appendix I analysis addressed herein using the core power level of 3659 MWt (which includes margin for power uncertainty) will have an insignificant impact on the calculated effluent releases/doses, since 3659 MWt represents only an approximate 0.1% increase from the core power of 3654 MWt utilized in the original analysis.

TABLE 11.2-8 RADIONUCLIDE DISCHARGE CONCENTRATIONS NORMAL LIQUID RELEASES - INCLUDING ANTICIPATED OPERATIONAL OCCURRENCES [HISTORICAL]

Radio Nuclide	Total Annual Release (Ci/yr)	Discharge Concentration (μCi/ml)	(MPC)w (μCi/ml)	Fraction of (MPC)w
H-3	7.30E+02*	1.1E-06	3E-03	3.7E-04
I-130	1.2E-04	1.8E-13	3E-06	6.1E-08
I-131	1.3E-01	2.0E-10	3E-07	6.6E-04
I-132	3.9E-03	5.9E-12	8E-06	7.4E-07
I-133	3.6E-02	5.5E-11	1E-06	5.5E-05
I-134	1.0E-04	1.5E-13	2E-05	7.5E-09
I-135	5.8E-03	8.8E-12	4E-06	2.2E-06
Br-83	4.0E-05	6.1E-14	1E-07	6.1E-07
Rb-86	2.0E-05	3.1E-14	2E-05	1.5E-09
Rb-88	1.0E-05	1.5E-14	NA	NA
Sr-89	3.0E-05	4.6E-14	3E-06	1.5E-08
Mo-99	2.1E-03	3.2E-12	4E-05	8.0E-08
Tc-99m	2.4E-03	3.7E-12	3E-03	1.2E-09
Te-127m	3.0E-05	4.6E-14	5E-05	9.2E-10
Te-127	3.0E-05	4.6E-14	2E-04	2.3E-10
Te-129m	1.2E-04	1.8E-13	2E-05	9.2E-09
Te-129	9.0E-05	1.4E-13	8E-04	1.8E-10
Te-131m	3.0E-05	4.6E-14	4E-05	1.1E-09
Te-132	7.3E-04	1.1E-12	2E-05	5.6E-08
Cs-134	2.0E-02	3.0E-11	9E-06	3.4E-06
Cs-136	2.7E-03	4.1E-12	6E-05	6.9E-08
Cs-137	1.5E-02	2.3E-11	2E-05	1.2E-06

* 7.30E+02 = 7.30×10^2

Radio Nuclide	Total Annual Release (Ci/yr)	Discharge Concentration (μCi/ml)	(MPC)w (μCi/ml)	Fraction of (MPC)w
Ba-137m	1.4E-02	2.1E-11	1E-07	2.1E-04
Ba-140	1.0E-05	1.5E-14	2E-05	7.6E-10
La-140	1.0E-05	1.5E-14	2E-05	7.6E-10
Cr-51	1.5E-04	2.3E-13	2E-03	1.1E-10
Mn-54	4.0E-05	6.1E-14	1E-04	6.1E-10
Fe-55	1.9E-04	2.9E-13	8E-04	3.6E-10
Fe-59	9.0E-05	1.4E-13	5E-05	2.7E-10
Co-58	1.6E-03	2.4E-12	9E-05	2.7E-08
Co-60	2.3E-04	3.5E-13	3E-05	1.2E-08
Np-239	3.0E-05	4.6E-14	1E-04	4.6E-10
All Others	6.0E-05	9.2E-14	1E-07	9.2E-07
TOTAL (Except Tritium)	2.4E-01	3.6E-10	-	9.3E-04

Note: The information presented in the above table represents the inputs/assumptions/results utilized in the Seabrook 10 CFR 50 Appendix I analysis (which is part of the original License Application), and is retained here for historical purposes. For the licensed core power level, the core power level for analysis purposes is 3659 MWt. Updating the Appendix I analysis addressed herein using the core power level of 3659 MWt (which includes margin for power uncertainty) will have an insignificant impact on the calculated effluent releases/doses, since 3659 MWt represents only an approximate 0.1% increase from the core power of 3654 MWt utilized in the original analysis.

TABLE 11.2-9 ESTIMATED ANNUAL RADIOACTIVE LIQUID RELEASES - DESIGN BASES FUEL LEAKAGE (1% FAILED FUEL) [HISTORICAL]

Radionuclide	Total Annual Release (Ci/yr)	Discharge Concentration (μCi/ml)	(MPC)w (μCi/ml)	Fraction of (MPC)w
H-3	6.08E+03*	9.2E-06	3E-03	3.1E-03
I-130	8.33E-04	1.3E-12	3E-06	4.3E-07
I-131	1.08E+00	1.7E-09	3E-07	5.5E-03
I-132	1.92E-02	2.9E-11	8E-06	3.7E-06
I-133	2.67E-01	4.1E-10	1E-06	4.1E-04
I-135	3.50E-02	5.3E-11	4E-06	1.3E-05
Br-83	1.67E-04	2.6E-13	1E-07	2.6E-06
Rb-86	1.67E-04	2.6E-13	2E-05	1.3E-08
Sr-89	2.50E-04	3.8E-13	3E-06	1.3E-07
Mo-99	1.67E-02	2.6E-11	4E-05	6.3E-07
Tc-99m	1.75E-02	2.7E-11	3E-03	9.2E-08
Te-127m	2.50E-04	3.8E-13	5E-05	7.7E-09
Te-127	2.50E-04	3.8E-13	2E-04	1.9E-09
Te-129m	1.00E-03	1.5E-12	2E-05	7.7E-08
Te-129	6.67E-04	1.0E-12	8E-04	1.3E-09
Te-131m	2.50E-04	3.8E-13	4E-05	9.2E-09
Te-132	5.83E-03	9.2E-12	2E-05	4.4E-07
Cs-134	1.50E-01	2.3E-10	9E-06	2.6E-05
Cs-136	2.25E-02	3.4E-11	6E-05	5.8E-07
Cs-137	1.17E-01	1.8E-10	2E-05	9.2E-06
Ba-137m	1.08E-01	1.7E-10	1E-07	1.7E-03

* $6.08E+03 = 6.08 \times 10^3$

Radionuclide	Total Annual Release (Ci/yr)	Discharge Concentration (μCi/ml)	(MPC)w (μCi/ml)	Fraction of (MPC)w
Ba-140	8.33E-05	1.3E-13	2E-05	6.3E-09
La-140	8.33E-05	1.3E-13	2E-05	6.3E-09
Cr-51	1.25E-03	1.9E-12	2E-03	9.2E-10
Mn-54	2.50E-04	3.8E-13	1E-04	3.8E-09
Fe-55	1.50E-04	2.3E-12	8E-04	2.8E-09
Fe-59	7.50E-04	1.2E-12	5E-05	2.3E-09
Co-58	1.25E-02	1.9E-11	9E-05	2.1E-07
Co-60	1.92E-03	2.9E-12	3E-05	1.0E-07
Np-239	1.67E-04	2.6E-13	1E-04	2.6E-09
All Others	5.00E-04	7.7E-13	1E-07	7.7E-06
TOTAL (Except Tritium)	1.86E+00	2.8E-09	---	7.7E-03

Note: The information presented in the above table represents the inputs/assumptions/results utilized in the Seabrook 10 CFR 50 Appendix I analysis (which is part of the original License Application), and is retained here for historical purposes. For the licensed core power level, the core power level for analysis purposes is 3659 MWt. Updating the Appendix I analysis addressed herein using the core power level of 3659 MWt (which includes margin for power uncertainty) will have an insignificant impact on the calculated effluent releases/doses, since 3659 MWt represents only an approximate 0.1% increase from the core power of 3654 MWt utilized in the original analysis.

TABLE 11.2-10 PARAMETERS USED IN DOSE CALCULATION OF RADIOACTIVE LIQUID RELEASES [HISTORICAL]

Mixing Ratio (Dilution Factor)	8
Circulating Water Flow Rate (ft^3/sec)	735
Shoreline Width Factor	0.5
Environmental Transit Time (hours)	
Aquatic Foods	24
Shoreline Exposure	0
Usage Factors for Fish (kg/year)	
Adult	21.0
Teen	16.0
Child	6.9
Usage Factors for Other Seafoods (kg/year)	
Adult	5.0
Teen	3.8
Child	1.7
Usage Factors For Shoreline Recreation (hours/year)	
Adult (including recreational clam digging activities)	334
Teen	67
Child	14

Note: The information presented in the above table represents the inputs/assumptions/results utilized in the Seabrook 10 CFR 50 Appendix I analysis (which is part of the original License Application), and is retained here for historical purposes. For the licensed core power level, the core power level for analysis purposes is 3659 MWt. Updating the Appendix I analysis addressed herein using the core power level of 3659 MWt (which includes margin for power uncertainty) will have an insignificant impact on the calculated effluent releases/doses, since 3659 MWt represents only an approximate 0.1% increase from the core power of 3654 MWt utilized in the original analysis.

TABLE 11.2-11 MAXIMUM ANNUAL DOSES FROM ALL PATHWAYS DUE TO RADIOACTIVE LIQUID RELEASES [HISTORICAL]

Organ	Adult (mrem/yr)	Teen (mrem/yr)	Child (mrem/yr)
Total Body	2.5E-03*	1.1E-03	6.3E-04
Skin	1.5E-05	3.0E-04	6.3E-05
Bone	2.1E-03	1.1E-03	1.2E-03
Liver	2.8E-03	1.7E-03	1.3E-03
Kidney	3.6E-03	2.6E-03	2.1E-03
Lung	1.8E-03	6.6E-04	3.8E-04
Thyroid	2.4E-02	2.1E-02	2.2E-02
GI-LLI	7.2E-03	4.5E-03	1.6E-03

Note: The information presented in the above table represents the inputs/assumptions/results utilized in the Seabrook 10 CFR 50 Appendix I analysis (which is part of the original License Application), and is retained here for historical purposes. For the licensed core power level, the core power level for analysis purposes is 3659 MWt. Updating the Appendix I analysis addressed herein using the core power level of 3659 MWt (which includes margin for power uncertainty) will have an insignificant impact on the calculated effluent releases/doses, since 3659 MWt represents only an approximate 0.1% increase from the core power of 3654 MWt utilized in the original analysis.

---

* 2.5E-03 = 2.5×10^{-3}

Highest organ dose: 2.4 E-02 mrem/yr; fraction of Appendix I: 2.4×10^{-3}

TABLE 11.3-1 RADIOACTIVE GASEOUS WASTE SYSTEM COMPONENT DESIGN DATA

Iodine Guard Beds

Quantity	Two
Material	Stainless Steel, Type 304
Type	Vertical, activated carbon cartridge
Capacity	1.2 scfm, 2 psig
Design Pressure	350 psig
Safety Class	NNS*

Molecular Sieve Dryer

Quantity	One
Material	Stainless Steel, Type 304
Type	Vertical, three tower
Capacity	1.2 scfm, 2 psig
Gas Outlet Temperature	70°F max.
Outlet Moisture Content	-40°F dew point (min.)
Design Pressure	350 psig
Design Temperature	500°F
Regeneration flow	11.3 scfm
Safety Class	NNS*

Carbon Delay Beds

Quantity	Five
Material	Carbon Steel
Type	Vertical tower 6x8 mesh type MBQ carbon (1600 lb/vessel)

* Purchased as Safety Class 3

Capacity	1.2 scfm, 2 psig
Design Pressure	Full vacuum - 350 psig
Gas Temperature	70°F
Safety Class	NNS*

Particulate Filters

Quantity	Two
Material	Stainless Steel
Type	HEPA
Capacity	1.2 scfm, 1 psig
Design Pressure	350 psig
Safety Class	NNS

Compressors

Quantity	Three
Material	Stainless Steel
Type	Single-stage, diaphragm
Capacity	H_2 gas comp. - 1.2 scfm @150 psig Regen. comp. - 11.3 scfm @150 psig
Safety Class	NNS

Hydrogen Surge Tank

Quantity	One
Material	Carbon Steel
Type	Vertical
Capacity	44 ft^3
Design Pressure	300 psig
Safety Class	NNS

Drain Transfer Pump

Quantity	One
Type	Rotorgear
Flow Rate	1 gpm
Discharge Pressure	40 psig

Drain Transfer Pump Motor

Voltage	460 V
Frequency	60 Hz
Horsepower	0.5 H.P.
Speed	1725 rpm

TABLE 11.3-2 ANNUAL GASEOUS EFFLUENTS RELEASE (Ci/yr) [HISTORICAL]

Radionuclide	Containment Purge	PAB Venting	Turbine Venting	Main Condenser Off-Gas System	Gaseous Waste System	Total
H-3	3.7E+02*	3.7E+02	c	c	c	7.4E+02
C-14	1.0E+00	a	a	a	7.0E+00	8.0E+00
Ar-41	2.5E+01	a	a	a	a	2.5E+01
Kr-83m	a	a	a	a	a	a
Kr-85m	7.0E+00	2.0E+00	a	1.0E+00	a	1.0E+01
Kr-85	1.0E+00	a	a	a	2.6E+02	2.6E+02
Kr-87	2.0E+00	1.0E+00	a	a	a	3.0E+00
Kr-88	1.0E+01	4.0E+00	a	2.0E+00	a	1.6E+01
Kr-89	a	a	a	a	a	a
Xe-131m	3.0E+00	a	a	a	2.0E+01	2.3E+01
Xe-133m	1.6E+01	a	a	a	a	1.6E+01
Xe-133	8.6E+02	3.4E+01	a	2.1E+01	7.7E+01	9.9E+02
Xe-135m	a	a	a	a	a	a
Xe-135	3.1E+01	4.0E+00	a	3.0E+00	a	3.8E+01
Xe-137	a	a	a	a	a	a
Xe-138	a	1.0E+00	a	a	a	1.0E+00
I-130	b	b	b	b	b	b
I-131	1.5E-02	4.3E-03	1.8E-03	2.7E-02	b	4.8E-02
I-132	b	b	b	b	b	b
I-133	1.1E-02	6.3E-03	1.9E-03	4.0E-02	b	5.9E-02
I-134	b	b	b	b	b	b

Radionuclide	Containment Purge	PAB Venting	Turbine Venting	Main Condenser Off-Gas System	Gaseous Waste System	Total
I-135	b	b	b	b	b	b
Mn-54	2.2E-04	1.8E-04	c	c	4.5E-05	4.4E-04
Fe-59	7.4E-05	6.0E-05	c	c	1.5E-05	1.5E-04
Co-58	7.4E-04	6.0E-04	c	c	1.5E-04	1.5E-03
Co-60	3.3E-04	2.7E-04	c	c	7.0E-05	6.7E-04
Sr-89	1.7E-05	1.3E-05	c	c	3.3E-06	3.3E-05
Sr-90	2.9E-06	2.4E-06	c	c	6.0E-07	5.9E-06
Cs-134	2.2E-04	1.8E-04	c	c	4.5E-05	4.4E-04
CS-137	3.7E-04	3.0E-04	c	c	7.5E-05	7.4E-04

* 3.7E+02 = 3.7×10^2

a. Less than 1.0 Ci/yr for Noble Gases and C-14

b. Less than 0.0001 Ci/yr for Iodine.

c. Less than 1.0 percent of total for this nuclide.

Note: The information presented in the above table represents the inputs/assumptions/results utilized in the Seabrook 10 CFR 50 Appendix I analysis (which is part of the original License Application), and is retained here for historical purposes. For the licensed core power level, the core power level for analysis purposes is 3659 MWt. Updating the Appendix I analysis addressed herein using the core power level of 3659 MWt (which includes margin for power uncertainty) will have an insignificant impact on the calculated effluent releases/doses, since 3659 MWt represents only an approximate 0.1% increase from the core power of 3654 MWt utilized in the original analysis.

TABLE 11.3-3 ESTIMATED MAXIMUM DOSES FROM NOBLE GASEOUS RELEASES [HISTORICAL]*

	Gamma Air Dose		Beta Air Dose	
Location	Annual Dose Rate (mrad/yr)	Fraction of Appendix I	Annual Dose Rate (mrad/yr)	Fraction of Appendix I
Highest Site Boundary Dose Point	1.2E-02	1.2E-03	2.3E-02	1.2E-03

	Total Body Dose		Skin Dose	
	Annual Dose Rate (mrad/yr)	Fraction of Appendix I	Annual Dose Rate (mrad/yr)	Fraction of Appendix I
Highest Residential Dose Point	4.4E-03	8.8E-04	1.0E-02	6.7E-04

* These values are for a two unit facility. Single unit values will be less.

Notes:

1. The numerical design objective of 10 CFR 50, Appendix I are:
   - Gamma air dose 10 mrad/yr, Beta air dose – 20 mrad/yr,
   - Total body dose 5 mrem/yr, skin dose – 15 mrem/yr
   - Organ dose due to radioiodines and other particulate radionuclides – 15 mrem/yr
2. The information presented in the above table represents the inputs/assumptions/results utilized in the Seabrook 10 CFR 50 Appendix I analysis (which is part of the original License Application), and is retained here for historical purposes. For the licensed core power level, the core power level for analysis purposes is 3659 MWt. Updating the Appendix I analysis addressed herein using the core power level of 3659 MWt (which includes margin for power uncertainty) will have an insignificant impact on the calculated effluent releases/doses, since 3659 MWt represents only an approximate 0.1% increase from the core power of 3654 MWt utilized in the original analysis.

TABLE 11.3-4 ESTIMATED MAXIMUM ANNUAL DOSE RATES (mrem/yr) FROM RADIOIODINES AND OTHER RADIONUCLIDES DUE TO STACK AND TURBINE VENT RELEASES [HISTORICAL]*

Age Group	Bone	Liver	Kidney	Lung	GI-LLI	Thyroid	Whole Body	Skin
Adult	6.3E-02**	3.4E-02	3.4E-02	3.4E-02	3.4E-02	7.2E-02	3.4E-02	1.1E-03
Teen	9.8E-02	4.4E-02	4.4E-02	4.3E-02	4.3E-02	9.0E-02	4.4E-02	1.1E-03
Child	2.3E-01	8.0E-02	7.9E-02	7.8E-02	7.8E-02	1.6E-01	7.9E-02	1.1E-03
Infant	7.0E-02	2.9E-02	2.9E-02	2.8E-02	2.8E-02	1.5E-01	2.8E-02	1.1E-03

Note: The information presented in the above table represents the inputs/assumptions/results utilized in the Seabrook 10 CFR 50 Appendix I analysis (which is part of the original License Application), and is retained here for historical purposes. For the licensed core power level, the core power level for analysis purposes is 3659 MWt. Updating the Appendix I analysis addressed herein using the core power level of 3659 MWt (which includes margin for power uncertainty) will have an insignificant impact on the calculated effluent releases/doses, since 3659 MWt represents only an approximate 0.1% increase from the core power of 3654 MWt utilized in the original analysis.

---

* These values are for a two unit facility. Single unit values will be less.

** $6.3E\text{-}02 = 6.3 \times 10^{-2} = 0.063$

The highest organ doses are 0.23 mrem/yr (child bone) and 0.16 mrem/yr (child thyroid), which are respectively 0.015 and 0.011 fractions of Appendix I dose objectives.

TABLE 11.3-5 GASEOUS WASTE RELEASE SOURCES AND ASSUMPTIONS [HISTORICAL]

1. Containment Venting	
Reactor coolant leakage	
Noble gaseous leakage	1% of total reactor coolant daily
Iodine leakage	0.001% of total reactor coolant daily
Charcoal filter efficiency	90%
Containment free volume	2.715×10^6 ft^3
Containment vent	4 purges per year during shutdown, 1000 cfm online purge system available during power operation
2. Primary Auxiliary Building	
Reactor coolant leakage	160 lb/day
Noble gases released	100%
Iodine released	0.75%
Charcoal filter efficiency	90%
Venting mode	Instantaneous release through charcoal filter
3. Main Condenser	
Steam generator tube leak	100 lb/day
Carry over in the steam generator	
Noble gases	100%
Iodine	1%
Other nuclides	0.1%
Fraction of main steam which reaches the main condenser	65%

	Partition factor in the main condenser	
	Noble gas	1.0
	Iodine	0.15
	Nonvolatile	0.0
	Charcoal filter efficiency	90%
	Vacuum pump vent rate	60 cfm
4.	Turbine Building Leakage	
	Secondary steam leakage	1700 lb/hr
	Noble gases released	100% of the leakage
	Iodine released	100% of the leakage
5.	Waste Gas System Release	
	Continuous stripping plus two reactor volumes degassed per year. All go through Gaseous Waste System.	
	Delay time in the charcoal beds	60 days for Xe 85 hr for Kr

Note: The information presented in the above table represents the inputs/assumptions/results utilized in the Seabrook 10 CFR 50 Appendix I analysis (which is part of the original License Application), and is retained here for historical purposes. For the licensed core power level, the core power level for analysis purposes is 3659 MWt. Updating the Appendix I analysis addressed herein using the core power level of 3659 MWt (which includes margin for power uncertainty) will have an insignificant impact on the calculated effluent releases/doses, since 3659 MWt represents only an approximate 0.1% increase from the core power of 3654 MWt utilized in the original analysis.

TABLE 11.3-6 GASEOUS RELEASES WITH 0.5% FAILED FUEL [HISTORICAL]

Radionuclide	Total, Off-Normal Unit Release Rate (Ci/year)
Kr-85m	4.2E+01*
Kr-85	1.1E+03
Kr-87	1.3E+01
Kr-88	6.7E+01
Xe-131m	9.6E+01
Xe-133m	6.7E+01
Xe-133	4.1E+03
Xe-135	1.6E+02
Xe-138	4.2E+00
I-131	2.0E-01
I-133	2.5E-01

Note: The information presented in the above table represents the inputs/assumptions/results utilized in the Seabrook 10 CFR 50 Appendix I analysis (which is part of the original License Application), and is retained here for historical purposes. For the licensed core power level, the core power level for analysis purposes is 3659 MWt. Updating the Appendix I analysis addressed herein using the core power level of 3659 MWt (which includes margin for power uncertainty) will have an insignificant impact on the calculated effluent releases/doses, since 3659 MWt represents only an approximate 0.1% increase from the core power of 3654 MWt utilized in the original analysis.

* $4.2E+01 = 4.2 \times 10^{1}$

TABLE 11.3-7 GASEOUS RELEASES WITH 500 GAL/DAY STEAM GENERATOR TUBE LEAKAGE FOR 90 DAYS [HISTORICAL]

Radionuclide	Total, Off-Normal Unit Release Rate (Ci/year)
Kr-85m	1.6E+01*
Kr-85	2.6E+02
Kr-87	3.0E+00
Kr-88	2.8E+01
Xe-131m	2.3E+01
Xe-133m	1.6E+01
Xe-133	1.1E+03
Xe-135	5.6E+01
Xe-138	1.0E+00
I-131	2.2E-01
I-133	3.1E-01

Note: The information presented in the above table represents the inputs/assumptions/results utilized in the Seabrook 10 CFR 50 Appendix I analysis (which is part of the original License Application), and is retained here for historical purposes. For the licensed core power level, the core power level for analysis purposes is 3659 MWt. Updating the Appendix I analysis addressed herein using the core power level of 3659 MWt (which includes margin for power uncertainty) will have an insignificant impact on the calculated effluent releases/doses, since 3659 MWt represents only an approximate 0.1% increase from the core power of 3654 MWt utilized in the original analysis.

* 1.6E+01 = 1.6×10^1

TABLE 11.3-8 GASEOUS RELEASES WITH 1 GAL/MIN OF REACTOR COOLANT LEAKAGE TO CONTAINMENT FOR 12 DAYS, FOLLOWED BY A CONTAINMENT PURGE [HISTORICAL]

Radionuclide	Total, Off-Normal Unit Release Rate (Ci/year)(1)
Kr-85m	1.0E+01*
Kr-85	2.6E+02
Kr-87	3.0E+00
Kr-88	1.6E+01
Xe-131m	2.3E+01
Xe-133m	1.7E+01
Xe-133	1.0E+03
Xe-135	3.9E+01
Xe-138	1.0E+00
I-131	5.6E-02
I-133	6.1E-02

Note: The information presented in the above table represents the inputs/assumptions/results utilized in the Seabrook 10 CFR 50 Appendix I analysis (which is part of the original License Application), and is retained here for historical purposes. For the licensed core power level, the core power level for analysis purposes is 3659 MWt. Updating the Appendix I analysis addressed herein using the core power level of 3659 MWt (which includes margin for power uncertainty) will have an insignificant impact on the calculated effluent releases/doses, since 3659 MWt represents only an approximate 0.1% increase from the core power of 3654 MWt utilized in the original analysis.

---

(1) Total release from all sources with 1 gpm of reactor coolant leakage in the Containment for 12 days prior to purge and the following assumptions:

a. Iodine partition factor = 0.0075

b. Carbon filter efficiency = 90%

* 1.0E+01 = 1.0×10^1

TABLE 11.3-9 GASEOUS RELEASES WITH 200 GAL/DAY REACTOR COOLANT LEAKAGE TO PRIMARY AUXILIARY BUILDING FOR 90 DAYS [HISTORICAL]

Radionuclide	Total, Off-Normal Unit Release Rate (Ci/year)
Kr-85m	1.4E+01*
Kr-85	2.6E+02
Kr-87	5.2E+00
Kr-88	2.5E+01
Xe-131m	2.3E+01
Xe-133m	1.6E+01
Xe-133	1.1E+03
Xe-135	4.7E+01
Xe-138	3.2E+00
I-131	5.8E-02
I-133	7.3E-02

Note: The information presented in the above table represents the inputs/assumptions/results utilized in the Seabrook 10 CFR 50 Appendix I analysis (which is part of the original License Application), and is retained here for historical purposes. For the licensed core power level, the core power level for analysis purposes is 3659 MWt. Updating the Appendix I analysis addressed herein using the core power level of 3659 MWt (which includes margin for power uncertainty) will have an insignificant impact on the calculated effluent releases/doses, since 3659 MWt represents only an approximate 0.1% increase from the core power of 3654 MWt utilized in the original analysis.

* 1.4E+01 = $1.4x10^1$

TABLE 11.4-1 PERMANENTLY INSTALLED EQUIPMENT NEEDED FOR PROCESSING WET WASTE VIA ALTERNATE MOBILE SOLIDIFICATION SYSTEM

Component	Quantity
Tanks:	
Spent resin hopper	1
Waste concentrates	1
Spent resin sluice	2
Waste feed	2
Pumps:	
Waste feed recirculation	2
Resin dewatering	1
Waste concentrates transfer	1
Spent resin transfer	1
Spent resin sluice	2
Spent resin recirculation	1
Resin centrifuge metering	1
Alternate solidification concentrates feed	1
Filters:	
Spent resin sluice	1
Panels:	
Crane control station	1
Crane power	1
Alternate solidification	1
Station feed control	1
Metering pumps SCR	1
Others:	
30 ton bridge crane	1
7½ ton transporter cart	1
Spent filter transfer cask	1

TABLE 11.4-2 DESIGN VOLUMETRIC INPUTS TO SOLID WASTE MANAGEMENT SYSTEM

Sources	Design Annual Volume (cf/yr)*	Expected Annual** Volume (cf/yr)
Dry active wastes		
(a) Noncompactible trash	8,800	4,400
(b) Compactible trash	20,400	10,200
Spent Demineralizer Resins	4,300	1,400
Evaporator Bottoms (at 12 w/o solids concentration) and Others, including filter cartridges	12,600	4,300
Totals	46,100	20,300

* The system design processing rate for evaporator bottoms and spent resins, using permanently installed equipment, is sufficient to process all of the design level quantities within 1664 hours of operation per year. This is equivalent to a 19% annual usage factor.

** These values are for a two unit facility. Single unit values will be less.

TABLE 11.4-3 Design Activity Inputs To Solid Waste Management System [HISTORICAL]

Sources		Expected Annual Activity (Ci/yr)*
Dry Active Wastes		7.2+01**
Spent Demineralizer Resins		3.6+03
Evaporator Bottoms & Other (at 12 w/o solids concentration)		3.1+02
Filter Cartridges		2.3+01
	Total	4.0+03

Note: The information presented in the above table represents the inputs/assumptions/results utilized in the Seabrook 10 CFR 50 Appendix I analysis (which is part of the original License Application), and is retained here for historical purposes. For the licensed core power level, the core power level for analysis purposes is 3659 MWt. Updating the Appendix I analysis addressed herein using the core power level of 3659 MWt (which includes margin for power uncertainty) will have an insignificant impact on the calculated effluent releases/doses, since 3659 MWt represents only an approximate 0.1% increase from the core power of 3654 MWt utilized in the original analysis.

* These values are for a two unit facility. Single unit values will be less.

** $7.2{+}01 = 7.2 \times 10^1$

TABLE 11.4-4 BASES AND ASSUMPTIONS FOR DETERMINING SOLID WASTES ACTIVITIES [HISTORICAL]

Volume Reduction Factors	
Compactible dry active waste	4 to 10
Bead resins in asphalt	1.85 minimum
Evaporator concentrates in asphalt	6.3 minimum
Radwaste in cement	1.0
Waste to Binder Mix Ratios, by Volume	
Asphalt	1 to 1 minimum
Cement	1.5 to 1 minimum
Waste Container Fill Fractions	
Bead resins in asphalt	91% average
Evaporator concentrates in asphalt	93% average
Radwaste in cement	93% average
Expected Failed Fuel Fraction	0.125% failed fuel
Decay of Filled Containers Awaiting Shipout	
Dry active wastes	none
Bead resins	30 days minimum
Evaporator concentrates	30 days minimum
Densities	
Noncompactible trash	10.0 lbs/cf
Compactible trash	37.4 lbs/cf
Encapsulated and solidified filter	37.4 lbs/cf
Depleted bead resins	56.8 lbs/cf
Evaporator concentrates	62.4 lbs/cf

Miscellaneous

Plant availability is eighty percent.

Each nonregenerable demineralizer is changed annually.

A maximum of thirty filter cartridges are expected to be shipped.

Time needed to fill each 55 gallon drum of waste for shipout, using asphalt binder, is 2 hours minimum; 6 hours maximum.

Time needed to fill each 85 cubic foot container of waste for shipout, using cement binder, is nominally 45 minutes.

Note: The information presented in the above table represents the inputs/assumptions/results utilized in the original License Application to determine solid waste activities and is retained here for historical purposes.

TABLE 11.4-5 EXPECTED ACTIVITY OF DRY ACTIVE WASTES AFTER PROCESSING [HISTORICAL]**

Noncompactible Trash

Isotope	ìCi/cc	Isotope	ìCi/cc
I-131	7.03-04*	Fe-59	8.44-03
Cs-134	2.32-02	Cr-51	8.59-02
Cs-137	3.83-02	Zn-65	1.05-03
Mn-54	7.28-02	Others	4.23-03
Co-58	1.41-02		
Co-60	1.04-01	Total	3.53-01

Compactible Trash

Isotope	ìCi/cc	Isotope	ìCi/cc
I-131	3.05-05	Fe-59	3.67-04
Cs-134	1.01-03	Cr-51	3.73-03
Cs-137	1.66-03	Zn-65	4.58-05
Mn-54	3.16-03	Other	1.84-04
Co-58	6.12-04		
Co-60	4.50-03	Total	1.53-02

Note: The information presented in the above table represents the inputs/assumptions/results utilized in the Seabrook 10 CFR 50 Appendix I analysis (which is part of the original License Application), and is retained here for historical purposes. For the licensed core power level, the core power level for analysis purposes is 3659 MWt. Updating the Appendix I analysis addressed herein using the core power level of 3659 MWt (which includes margin for power uncertainty) will have an insignificant impact on the calculated effluent releases/doses, since 3659 MWt represents only an approximate 0.1% increase from the core power of 3654 MWt utilized in the original analysis.

** based on Table 11.4-4

* 7.03-04 = 7.03×10^{-4}

TABLE 11.4-6 EXPECTED ACTIVITY OF SPENT DEMINERALIZER RESINS [HISTORICAL]**

Isotope	ìCi/cc	Isotope	ìCi/cc
I-131	1.6+0*	Te-132	5.5-4
Sr-89	2.7-1	Ba-140	1.9+0
Sr-90	1.7-1	Ce-144	3.6+1
Y-91	1.2-2	Mn-54	1.5+1
Zr-95	1.0+0	Co-58	1.4+1
Nb-95	4.4-1	Co-60	7.2+0
Mo-99	5.4-4	Fe-59	4.8-1
Cs-134	1.8+1	Cr-51	2.4+0
Cs-136	1.9-1		
Cs-137	2.4+1	Total	1.2+2

Processed Using Cement Binder Via Alternate Mobile Equipment

Isotope	ìCi/cc	Isotope	ìCi/cc
I-131	5.2-1	Te-132	1.8-4
Sr-89	8.7-2	Ba-140	6.1-1
Sr-90	5.6-2	Ce-144	1.2+1
Y-91	3.9-3	Mn-54	4.9+0
Zr-95	3.3-1	Co-58	4.4+0
Nb-95	1.4-1	Co-60	2.3+0
Mo-99	1.7-4	Fe-59	1.6-1
Cs-134	6.0+0	Cr-51	7.6-1
Cs-136	6.2-2		
Cs-137	7.8+0	Total	4.0+1

Note: The information presented in the above table represents the inputs/assumptions/results utilized in the Seabrook 10 CFR 50 Appendix I analysis (which is part of the original License Application), and is retained here for historical purposes. For the licensed core power level, the core power level for analysis purposes is 3659 MWt. Updating the Appendix I analysis addressed herein using the core power level of 3659 MWt (which includes margin for power uncertainty) will have an insignificant impact on the calculated effluent releases/doses, since 3659 MWt represents only an approximate 0.1% increase from the core power of 3654 MWt utilized in the original analysis.

** based on Table 11.4-4

* 1.6+01 = 1.6×10^{0}

TABLE 11.4-7 EXPECTED ACTIVITY OF EVAPORATOR BOTTOMS AND OTHER [HISTORICAL]**

Evap Btms & Chemical Wastes Processed Using Asphalt Binder

Isotope	ìCi/cc	Isotope	ìCi/cc
I-131	7.1-1*	Te-132	6.3-4
Sr-89	1.6-2	Ba-140	2.5-3
Sr-90	1.6-3	Ce-144	3.2-3
Y-91	2.4-2	Mn-54	4.7-2
Zr-95	3.0-3	Co-58	1.7-1
Nb-95	2.9-3	Co-60	5.3-2
Mo-99	2.2-3	Fe-59	3.0-2
Cs-134	6.8-1	Cr-51	1.6-2
Cs-136	1.8-1		
Cs-137	8.7-1	Total	3.3+0

Evap Btms & Chemical Wastes Processed Using Cement Binder Via Alternate Mobile Equipment

Isotope	ìCi/cc	Isotope	ìCi/cc
I-131	6.8-2	Te-132	6.0-5
Sr-89	1.5-3	Ba-140	2.4-4
Sr-90	1.5-4	Ce-144	3.1-4
Y-91	2.3-3	Mn-54	4.5-3
Zr-95	2.9-4	Co-58	1.7-2
Nb-95	2.8-4	Co-60	5.1-3
Mo-99	2.1-4	Fe-59	2.8-3
Cs-134	6.4-2	Cr-51	1.5-3
Cs-136	1.7-2		
Cs-137	8.3-2	Total	2.4-1

** based on Table 11.4-4

* $7.1\text{-}1 = 7.1 \times 10^{-1}$

Note: The information presented in the above table represents the inputs/assumptions/results utilized in the Seabrook 10 CFR 50 Appendix I analysis (which is part of the original License Application), and is retained here for historical purposes. For the licensed core power level, the core power level for analysis purposes is 3659 MWt. Updating the Appendix I analysis addressed herein using the core power level of 3659 MWt (which includes margin for power uncertainty) will have an insignificant impact on the calculated effluent releases/doses, since 3659 MWt represents only an approximate 0.1% increase from the core power of 3654 MWt utilized in the original analysis.

TABLE 11.4-8 SOLID WASTE MANAGEMENT SYSTEM EQUIPMENT PARAMETERS

Tanks

Waste Concentrates Tank (1-WS-TK-76)

Capacity	6750 gallons – maximum 6000 gallons - working
Material	Incoloy 825
Design/Operating Pressure	15 psig/atmos
Design/Operating Temperature	250°F/180°F
Design Code	ASME VIII - Div. 1
Minimum Holdup Time	2.0 Hrs.

Waste Feed Tanks (1-WS-TK-198A & B)

Capacity (each)	1320 gallons – maximum 1000 gallons - working
Material	Incoloy 825
Design/Operating Pressure	14.7 psig/atmos
Design/Operating Temperature	265°F/180°F
Design Code	ASME VIII Div. 1 - Nonstamped
Minimum Holdup Time	23.8 Hrs.

Bottoms Collection Tank
(1-WS-TK-200)

Capacity	1300 gallons – maximum 1000 gallons - working
Material	Incoloy 825
Design/Operating Pressure	5 psig/0.5 psig
Design/Operating Temperature	250°F/200°F
Design Code	ASME VIII - Div. 1
Minimum Holdup Time	23.8 Hrs.

Caustic Day Tank (1-WS-TK-199)

Capacity	250 gallons – maximum 200 gallons - working
Material	(Plastic)
Design/Operating Pressure	(Atmos/atmos)
Design/Operating Temperature	200°F/150°F
Design Code	Mfr. Std.
Safety Class	NNS

Resin Hopper (1-WS-TK-81)

Capacity	934 gallons
Design Operating Pressure	Atmos./atmos.
Material	Incoloy 825
Design Operating Temperature	200°F/200°F
Design Code	ASME VIII
Minimum Holdup Time	31.1 Hrs.

Crystallizer Distillate Tank
(1-WS-TK-210)

Capacity	30 gallons
Material	304L Stainless Steel
Design/Operating Pressure	30 psig/Atmos.
Design Operating Temperature	220°F/130°F
Design Code	ASME VIII

Asphalt Storage Tank (1-WS-TK-201)

Capacity	7400 gallons – maximum 7000 gallons - working
Material	ASTM A285
Design/Operating Pressure	Atmos./atmos.
Design/Operating Temp.	425°F/325°F
Design Code	API 650

Auxiliary Boiler Condensate Return Tank
(1-WS-TK-202)

Capacity	40 gallons
Material	ASTM A36
Operating Pressure	Atmos.
Operating Temperature	210°F
Design Code	Mfr. Standards

Spent Resin Sluice Tanks
(RS-RK-79A,B)

Capacity	9500 gal.
Material	304 SS
Design/Operating Press	150/5 psig
Design/Operating Temp.	150°F
Design Code	ASME VIII

Pumps

Concentrates Transfer Pump
(1-WS-P-354)

Type	Centrifugal
Design Flow	50 gpm

Waste Feed Recirculation Pumps
(1-WS-P-332A & B)

Type	Centrifugal
Design Flow	50 gpm

Caustic Metering Pump (1-WS-P-333)

Type	Progressive Cavity
Design Flow	2 gpm

Alternate Station Concentrates Feed Pump (1-WS-P-346)

Type	Progressive Cavity
Design Flow	18 gpm

<u>Spent Resin Transfer Pump</u>
(1-WS-P-331)

Type	Progressive Cavity
Design Flow	100 gpm

<u>Spent Resin Dewatering Pump</u>
(1-WS-P-353)

Type	In-line Centrifugal
Design Flow	10 gpm

<u>Spent Resin Recirculation Pump</u>
(1-WS-P-342)

Type	Progressive Cavity
Design Flow	50 gpm

<u>Resin Centrifuge Metering Pump</u>
(1-WS-P-338)

Type	Progressive Cavity
Design Flow	1-10 gpm

<u>Crystallizer Recirculation Pump</u>
(1-WS-P-334)

Type	Centrifugal
Design Flow	1800 gpm

<u>Crystallizer Distillate Pump</u>
(1-WS-P-335)

Type	Centrifugal
Design Flow	5 gpm

Crystallizer Reflux Pump (1-WS-P-352)

Type	Diaphragm
Design Flow	1 gpm

Crystallizer Drain Pump (1-WS-P-347)

Type	Centrifugal
Design Flow	1 gpm

Asphalt Recirculation Pump
(1-WS-P-339)

Type	Progressive Cavity
Design Flow	20 gpm

Asphalt Metering Pump (1-WS-P-340)

Type	Progressive Cavity
Design Flow	3-30 gpm

Auxiliary Boiler Feed Pumps
(1-WS-P-341 A & B)

Type	Centrifugal
Design Flow	9 gpm

Spent Resin Sluice Pumps
(WS-P-13 A, B)

Type	Centrifugal
Design Flow	250 gpm

Heat Exchangers, Coolers, Heaters, etc.

Crystallizer Condenser (1-WS-E-158)

Design Code	ASME VIII, Div. 1 TEMA R	
	Shell Side	Tube Side
Fluid	Process Vapor	Cooling Water
Design Flow	(2685 #/hr)	(150 gpm)
Design Temp	250°F	200°F
Operating Temperature	213°F	125°F
Design Pressure	30 psig	150 psig
Operating Pressure	Atmos.	60 psig
Material	304 S/S	Carbon Steel

Crystallizer Subcooler (1-WS-E-160)

Design Code	ASME VIII, Div. 1 TEMA R	
	Shell Side	Tube Side
Fluid	Distillate	Cooling Water
Design Flow	(2185 #/hr)	(150 gpm)
Design Temperature	250°F	200°F
Operating Temperature	212°F	85°F
Design Pressure	150 psig	150 psig
Operating Pressure	100 psig	100 psig
Material	304 S/S	Carbon Steel

Crystallizer Heater (1-WS-E-156)

Design Code	ASME VIII, Div. 1 TEMA R	
	Shell Side	Tube Side
Fluid	Steam	Concentrates
Design Flow	(3500 #/hr)	(1200 gpm)
Design Temperature	300°F	300°F
Operating Temperature	298°F	220°F
Design Pressure	150 psig	50 psig
Operating Pressure	50 psig	15 psig
Material	Carbon Steel	Inconel 625

Crystallizer Vapor Body (1-WS-EV-6)

Material	Inconel 625
Capacity	600 gallons - vapor body working 1200 gallons -vapor body, heater, recirculation pipe flooded
Design Code	ASME VIII, Div. 1
Design/Operating Pressure	30 psig/15 psia
Design/Operating Temperature	300°F/222°F

Entrainment Separator (1-WS-E-157)

Capacity	2695 #/hr.
Volume	20 gallons – working 300 gallons - flooded
Design Code	ASME VIII, Div. 1
Design Pressure	Full vacuum to 30 psig
Operating Pressure	Atmos.
Design Temperature	270°F
Operating Temperature	213°F

Auxiliary Boiler Vessel (1-WS-E-159)

Capacity - Steam	5000 #/Hr.
Operating Pressure	275 psig
Operating Temperature	410°F
Design Code	ASME VIII, Div. 1

Other Components

Resin Centrifuge (1-WS-MM-611)

Type	Horizontal
Capacity	4.7 gpm
Operating Pressure	Atmos.
Operating Temperature	Ambient
Design Code	Mfr. Std.

Evaporator/Extruder

Capacity Range	9.25 to 27.5 gph
Operating Pressure	Atmos.
Operating Temp.	280°F
Design Code	DIN

Shipping Containers

Types	55 gallon drums, 100 ft^3LSA boxes, and 85 ft^3 liners with associated shield

Resin Filter (RS-F-19)

Type	Backflushable, wedgewire
Design Code	ASME VIII, Div. 1

TABLE 11.4-9 DESIGN SHIPPING VOLUMES

	Processed Using Asphalt Binder Via Permanently Installed Equipment		Processed Using Cement Binder Via Alternate Mobile Equipment	
	Design Annual Volume (Ft^3)	Expected* Annual Volume (Ft^3)	Design Annual Volume (ft^3)	Expected* Annual Volume (ft^3)
Sources				
Dry Active Wastes				
a) Noncompactible Trash	8,800	4,400	8,800	4,400
b) Compactible Trash	5,100	2,600	5,100	2,600
Spent Demineralizer Resins	2,300	770	7,200	2,400
Evaporator Bottoms & Other	1,900	630	20,000	6,600
Encapsulated and Solidified Filter Cartridges	600	300	600	300
Totals	18,700	8,700	41,700	16,300

* These values are for a two unit facility. Single unit values will be less.

TABLE 11.4-10 EXPECTED ANNUAL ACTIVITY AVAILABLE FOR OFFSITE SHIPMENT (Ci/yr) [HISTORICAL]*

Isotope	Dry Active Wastes	Spent Demin Resins(1)	Evap Bottoms(1)	Total
I-131	1.4-1**	3.5+1	1.3+1	4.7+1
Sr-89	-------	5.9+0	2.8-1	6.1+0
Sr-90	-------	3.8+0	2.8-2	3.8+0
Sr-91	-------	2.7-1	4.4-1	7.1-1
Zr-95	-------	2.2+1	5.4-2	2.2+1
Nb-95	-------	9.5+0	5.3-2	9.5+0
Mo-99	-------	1.2-2	4.0-2	6.2-2
Cs-134	4.8+0	4.0+2	1.2+1	4.2+2
Cs-136	---	4.2+0	3.2+0	7.9+0
Cs-137	7.8+0	5.3+2	1.6+1	5.5+2
Te-132	-------	1.2-2	1.1-2	2.3-2
Ba-140	-------	4.1+2	4.5-2	4.1+2
Ce-144	-------	7.8+2	5.8-2	7.8+2
Mn-54	1.5+1	3.3+2	8.5-1	3.4+2
Co-58	2.9+0	3.0+2	3.1+0	3.1+2
Co-60	2.1+1	1.6+2	9.5-1	1.8+2
Fe-59	1.7+0	1.0+1	5.3-1	1.2+1
Cr-51	1.8+1	5.1+1	2.8-1	7.0+1
Others	1.1+0	------	------	1.0+0
Totals	7.2+1	2.7+3	5.2+1	2.7+3

Note: The information presented in the above table represents the inputs/assumptions/results utilized in the Seabrook 10 CFR 50 Appendix I analysis (which is part of the original License Application), and is retained here for historical purposes. For the licensed core power level, the core power level for analysis purposes is 3659 MWt. Updating the Appendix I analysis addressed herein using the core power level of 3659 MWt (which includes margin for power uncertainty) will have an insignificant impact on the calculated effluent releases/doses, since 3659 MWt represents only an approximate 0.1% increase from the core power of 3654 MWt utilized in the original analysis.

* These values are for a two-unit facility. Single unit values will be less.

(1) Thirty days decay minimum prior to shipout.

** $1.4\text{-}1 = 1.4\times10^{-1}$

TABLE 11.5-1 PROCESS AND EFFLUENT RADIATION MONITORS

Instrument Tag No. Re-	Description	Detector Type	Det Back Grd mr/hr	Range Low-High (µci/cc)	(Note 5) Alarm Set Point (µci/cc)	Reference Isotope	Detector Qty	Safety Class	Energy* Level	Loop Diag. 1-NHY	P&ID 1-NHY
6454	Storm Drains	Gamma Scint	0.5	10^{-6} 10^{-2}		Co^{58},I^{131},CS^{137}	1	Non 1E	Note 2	506765	SD-20404
6502	Waste Gas Inlet to Carbon Delay Beds	Gamma Scint	15.0	10^{-2} 10^{+2}		Xe^{133}	1	Non 1E	Note 1	506897	20772
6503	Waste Gas Compressor Inlet	Gamma Scint	15.0	10^{-3} 10^{+1}		Kr^{85}	1	"	Note 1	506898	20770
6504	H_2 Gas Compressor Disch.	Gamma Scint	15.0	10^{-3} 10^{+1}		Kr^{85}	1	"	Note 1	506899	20773
6505	Condenser Air Evac	Beta Scint	0.5	Note 3		Xe^{133}	1	"	Note 1	506055	20774
6515,6516	Primary Component Cooling Water	Gamma Scint	2.5	10^{-7} 10^{-3}		Co^{58},I^{131},CS^{137}	2	"	Note 2	506190, 506194	20211, 20205
6509	Liquid Waste Test Tk Disch to CWS	Gamma Scint	2.5	10^{-6} 10^{-2}		Co^{58},I^{131},CS^{137}	1	"	Note 2	506927	20831
6510, 6511, 6512, 6513	Steam Gen Blowdown Sample Loops 1,2,3,4	Gamma Scint	2.5	10^{-6} 10^{-2}		Co^{58},I^{131},CS^{137}	4	"	Note 2	506815	20521
6519	Steam Gen Blowdown Flash Tank Drain	Gamma Scint	2.5	10^{-7} 10^{-3}		Co^{58},I^{131},CS^{137}	1	"	Note 2	506734	20626
6520	Reactor Coolant Gross Activity Monitor	GM	15	10^{-1} 10^{+4} mr/hr		Co^{58},I^{131},CS^{137}		"	Note 2	506269	20722

* See Table 11.5-1 (Sheet 3) for notes.

Instrument Tag No. Re-	Description	Detector Type	Det Back Grd mr/hr	Range Low-High (µci/cc)	(Note 5) Alarm Set Point (µci/cc)	Reference Isotope	Detector Qty	Safety Class	Energy* Level	Loop Diag. 1-NHY	P&ID 1-NHY
6481-1, 6482-1 6481-2, 6482-2	Main Steam Line Monitor	Gamma Scint	2.5	10^{-0} 10^{+5}mr/hr		Xe^{133}		"	-	506551 -2, -3, -4	20580 20581
6490	Aux Steam Cond	Gamma Scint	0.5	$10^{-7}10^{-3}$		Co^{58}, I^{131}, CS^{137}	1	Non 1E	Note 2	507165	20908
6521	Turb. Bldg. Sump Liq. Monitor	Gamma Scint	2.5	10^{-6} 10^{-2}		Co^{58}, I^{131}, CS^{137}	1	Non 1E	Note 2	506713, 506716	20195
6527A1, A2 B1, B2	COP Monitors	GM	2.5	10^{1} -10^{6}cpm		Xe^{133}	4	1E	Note 1	506211	20504
6560	Resin Sluice Line	GM	100	Note 4		Xe^{133}	1	Non 1E		506694	20252
6561	Resin Transfer Line	GM	100	Note 4		Xe^{133}	1	Non 1E		506694	20735
6564	Sluice Pump Line	GM	100	Note 4		Xe^{133}	1	Non 1E		586692	20252
6473	Water Treatment Liquid Effluent Radiation Monitor	Gamma Scint	2.5	10^{-6} 10^{-2}		Co^{58}, I^{131}, CS^{137}	1	Non 1E	Note 2	506976	20040

Note	Isotopes	Max. Beta Energy (Mev)	Predominant Gamma Energy (Mev)
1	Xe^{133}	0.346	0.081
	Xe^{135}	0.92	0.249
	Kr^{85}	0.67	0.514
	Kr^{85m}	0.82	0.150
2	I^{131}	0.606	0.364
	I^{133}	1.27	0.53
	Cs^{134}	0.662	0.604
	Cs^{137}	0.514	0.662
	Co^{58}	0.474	0.81
	Co^{60}	0.314	1.17, 1.33

3 Condenser Air Evacuation Monitor to have output in counts per min (cpm) (10^1 to 10^6).

4 Monitors 6560, 6561, 6564 have output in mr/hr (10° to 10^5).

5 Radiation monitoring setpoints are varied during operation to follow station operating conditions. Setpoints are maintained within the bounds established in the Technical Specifications. The methodology for establishing the setpoints is found in the Station Offsite Dose Calculation Manual (ODCM) and/or station operating procedures.

TABLE 11.5-2 AUTOMATIC SYSTEM AND OPERATOR RESPONSES TO ANNUNCIATED RADIOACTIVITY LEVEL LIMITS

Monitor	Automatic Response to Radioactivity Level Limit	Operator Response to Radioactivity Level Limit
Waste Gas Processing (6504)	Closure of waste gas discharge valve	Request sampling and laboratory analysis.
Condenser Air Evacuation System	No automatic response	Request sampling and laboratory analysis and/or observe steam generator liquid samplers.
Component Cooling Liquid	No automatic response	Request sampling and laboratory analysis.
Waste Processing System Liquid	Close valve in effluent line	Terminate discharge of liquid effluents – Effluent request sampling and laboratory analysis of effluent.
Steam Generator	Isolation valve in the	Request sampling and Liquid Samples blowdown flash tank laboratory analysis discharge closes (observe Tech. Spec. 3/4.7.1.4 limit on secondary activity)
Reactor Coolant Gross Activity	No automatic response	Request sampling and laboratory analysis of reactor coolant samples to detect failed fuel (observe Tech. Spec. 3/4.4.8 limit on reactor coolant activity)
Turbine Bldg. Sump	Lock-out of sump pump	Request sampling and Liquid operation laboratory analysis (observe Tech. Spec. 3/4.3.3.9 limit on secondary activity)
Containment Online Purge	Close isolation valves on 8" atmospheric purge line	Request sampling and laboratory analysis of containment air and sump water.

Monitor	Automatic Response to Radioactivity Level Limit	Operator Response to Radioactivity Level Limit
Auxiliary Condensate Return	Terminate Condensate return and isolate return Piping	(later)
Water Treatment Liquid Effluent	Close valve in effluent line	Request sampling and laboratory analysis – check operation of system.

TABLE 11.5-3 RADIOLOGICAL SAMPLING MONITORING

Sampling Location	Basis for Selection of Location	Expected Process Flowrate	Sample Composition	Expected Concentrations ($\mu Ci/cm^3$)	Types Effluent Releases
Plant Vent	Determination of identity and quantity of radionuclides being released, calibration check of PVS monitor airborne radioactive	312,185 cfm (max) 272,185 cfm (normal)	Ventilation exhaust air	I-131: $1.18x10^{-11}$ Xe-133: $2x10^{-7}$ Cs-137: $1.82x10^{-13}$	Continuous releases of fission and activation gases and tritium; releases of airborne radioactive particulates; releases of iodines
Condenser Air Evacuation	Determination of identity and quantity of radionuclides being released; calibration check of the condenser air evacuation monitor	10-20 cfm Condenser in leakage gases and t	Condenser gases	Xe-133: $1.6x10^{-5}$	Continuous releases of fission and activation gases and tritium; release of airborne radioactive iodine
Turbine Gland Sealing System Exhaust	Determination of identity and quantity of radionuclides being released	<300 cfm	Condenser gases	I-131: $3x10^{-7}$	Continuous releases of fission and activation gases and tritium; release of airborne radioactive iodine

Sampling Location	Basis for Selection of Location	Expected Process Flowrate	Sample Composition	Expected Concentrations ($\mu Ci/cm^3$)	Types Effluent Releases
Steam Generator Blowdown Flash Tank Drain Outlet calibration	Determination of identity and quantity of radionuclides released	200 gpm	Steam generator blowdown liquid	Co-58: $1.7x10^{-6}$ I-131: $3.3x10^{-5}$ Cs-134: $2.88x10^{-6}$	Continuous releases of liquids
Liquid Waste Test Tanks (pump discharge) waste demin. filter outlet and Recovery Test Tank Pump Discharge	Determination of identity and quantity of radio nuclides released; set trip point of discharge monitor	Variable, up to 150 gpm	Processed radwaste	Co-58: $1x10^{-6}$ I-131: $1x10^{-5}$ Cs-134: $4.1x10^{-9}$	Batch releases of liquids
Waste Evaporator Bottoms	Determination of identity and quantity of radio nuclides in concentrate	Variable	Concentrated liquid radwaste	Cs-134: $1x10^{-3}$ Cs-137: $1x10^{-3}$ Co-58: $1x10^{-1}$	Batch releases of concentrates
Hydrogen Surge Tank	Determination of identity and quantity of radionuclides stored in tank	Variable, depending on concentration of radionuclides	H_2' N_2 noble gases	Kr-85: 2.86 Xe-133: $6.97x10^{-2}$	Batch releases of fission product gases

Sampling Location	Basis for Selection of Location	Expected Process Flowrate	Sample Composition	Expected Concentrations ($\mu Ci/cm^3$)	Types Effluent Releases
Waste Gas System	Determination of identity and quantity of radio-nuclides upstream and downstream of carbon delay beds and downstream of waste gas compressors; calibration check of waste gas monitors, efficiency check of delay beds	0.4 cfm	H_2· N_2 noble gases	Xe-133: 1.92×10^{-2} Kr-85: 2.88 (input to system)	Continuous releases of fission gases and tritium
Turbine Building Sump	Determination that the sump is free of contamination	-	H_2O	$<1 \times 10^{-9}$ (gross β - γ)	Continuous
Containment Purge	Determination of identity and quantity of radionuclides being released	1000 cfm	Ventilation exhaust air	Xe-133: 3.8×10^{-4}	Continuous

Sampling Location	Basis for Selection of Location	Expected Process Flowrate	Sample Composition	Expected Concentrations ($\mu Ci/cm^3$)	Types Effluent Releases
PAB Ventilation System	Determination of identity and quantity of radionuclides being released	43,200 cfm (clean filter condition) 39,225 cfm (dirty filter conditions)	Ventilation exhaust air	I-131: $2.9x10^{-11}$ Xe-133: $3.9x10^{-7}$ CS-137: $1.2x10^{-12}$	Continuous
Fuel Storage Building	Determination of identity and quantity of radionuclides being released and present in building atmosphere	34,000 cfm	Ventilation exhaust air	$2.7x10^{-6}$ (gross β - γ)	Continuous
Waste Process Building	Determination of identity and quantity of radionuclides in building atmosphere	151,620 cfm	Ventilation exhaust air	I-131: $3.2x10^{-13}$ CS-137: $1.6x10^{-13}$	Continuous
Water Treatment System	Determination of identity and quantity of radionuclides being released	350 gal/day	H_2O	$<1x10^{-9}$ (gross β - γ)	Continuous

Sampling Location	Basis for Selection of Location	Expected Process Flowrate	Sample Composition	Expected Concentrations ($\mu Ci/cm^3$)	Types Effluent Releases
Turbine Gland Steam Condenser	Determination of identity and quantity of radionuclides being released	--	Air with high moisture content	I-131: $2.6x10^{-10}$	Continuous*
Evaporator System Distillate Cooler Vent	Determination of identity and quantity of radionuclides being processed	----	Evaporator gases	I-131: $5.5x10^{-8}$ Xe-133: $5.8x10^{-10}$	Intermittent
Pressurizer and BRS Vent System	Determination of identity and quantity of radionuclides being released	----	Gases	I-131: $6.1x10^{-4}$ Xe-133: 1.6	Intermittent
Component Cooling Water System	Determination if cross-contamination with reactor coolant has occurred	8,986 gpm max	Demineralized water	Within the statistical background	None during normal operation

*Only with a (primary to secondary) steam generator tube leak.

Note: Sampling frequencies and sensitivities shall be as specified in the Offsite Dose Calculation Manual (ODCM).

Sampling Location	Basis for Selection of Location	Expected Process Flowrate	Sample Composition	Expected Concentrations ($\mu Ci/cm^3$)	Types Effluent Releases
Service Water System	Determination if cross-contamination with reactor coolant has occurred	10,500 gpm per train	Sea water containing sodium chloride	Within the statistical background	Continuous but this system is not normally radioactive

TABLE 11A-1 LIQUID WASTE PROCESSING SYSTEMS – SOURCES AND PROCESSING PARAMETERS

		Decontamination Factors				Holdup Times (days)		
System	Input Flow Rate (gal. per day)	Iodine	Cesium, Rubidium	Others	Fraction of Primary Coolant Activity (PCA)	Processing & Collection	Discharge	Fraction Discharged
Miscellaneous Waste	1,360	10^3	10^4	10^4	0.061	5.9	0.23	1.0
Equipment Drain	302	10^3	$2x10^3$	10^4	1.0	23.0	5.03	0.464
Turbine Building Sump Waste and Water Treatment Liquid Effluent that includes Condensate Polishing System	7,550	1.0	1.0	1.0	[a]	0.0	0.0	1.0
Boron Recovery System (Includes Shim Bleed)	878	10^3	$2x10^3$	10^4	1.0	152.8	5.03	0.560
Steam Generator Blowdown	108,000	10^3	10^4	10^4	-	0.0	0.0	0.7

[a] Activity levels are based on secondary side main steam inventories.

TABLE 11A-2 RADIONUCLIDE DISCHARGE CONCENTRATIONS NORMAL LIQUID RELEASES - INCLUDING ANTICIPATED OPERATIONAL OCCURRENCES

	Total Annual Release (Ci/yr)	Discharge Concentration (μCi/ml)	(MpC)w (μCi/ml)	Fraction of Nuclide (MPC)w
H-3	7.30E+02*	1.1E-06	3E-03	3.7E-04
I-130	1.2E-04	1.8E-13	3E-06	6.1E-08
I-131	1.3E-01	2.0E-10	3E-07	6.6E-04
I-132	3.9E-03	5.9E-12	8E-06	7.4E-07
I-133	3.6E-02	5.5E-11	1E-06	5.5E-05
I-134	1.0E-04	1.5E-13	2E-05	7.5E-09
I-135	5.8E-03	8.8E-12	4E-06	2.2E-06
Br-83	4.0E-05	6.1E-14	1E-07	6.1E-07
Rb-86	2.0E-05	3.1E-14	2E-05	1.5E-09
Sr-89	3.0E-05	4.6E-14	3E-06	1.5E-08
Mo-99	2.1E-03	3.2E-12	4E-05	8.0E-08
Tc-99m	2.4E-03	3.7E-12	3E-03	1.2E-09
Te-127m	3.0E-05	4.6E-14	5E-05	9.2E-10
Te-127	3.0E-05	4.6E-14	2E-04	2.3E-10
Te-129m	1.2E-04	1.8E-13	2E-05	9.2E-09
Te-129	9.0E-05	1.4E-13	8E-04	1.8E-10
Te-131m	3.0E-05	4.6E-14	4E-05	1.1E-09
Te-132	7.3E-04	1.1E-12	2E-05	5.6E-08
Cs-134	2.0E-02	3.0E-11	9E-06	3.4E-06
Cs-136	2.7E-03	4.1E-12	6E-05	6.9E-08
Cs-137	1.5E-02	2.3E-11	2E-05	1.2E-06

	Total Annual Release (Ci/yr)	Discharge Concentration (µCi/ml)	(MpC)w (µCi/ml)	Fraction of Nuclide (MPC)w
Ba-137m	1.4E-02	2.1E-11	1E-07	2.1E-04
Ba-140	1.0E-05	1.5E-14	2E-05	7.6E-10
La-140	1.0E-05	1.5E-14	2E-05	7.6E-10
Cr-51	1.5E-04	2.3E-13	2E-03	1.1E-10
Mn-54	4.0E-05	6.1E-14	1E-04	6.1E-10
Fe-55	1.9E-05	2.9E-14	8E-04	3.6E-11
Fe-59	9.0E-05	1.4E-13	5E-05	2.7E-10
Co-58	1.6E-03	2.4E-12	9E-05	2.7E-08
Co-60	2.3E-04	3.5E-13	3E-05	1.2E-08
Np-239	3.0E-05	4.6E-14	1E-04	4.6E-10
All Others	6.0E-05	9.2E-14	1E-07	9.2E-07
Total (Except Tritium)	2.4E-01	3.6E-10	-	9.3E-04

* 7.30E+02 = 7.30×10^2

TABLE 11A-3 ANNUAL GASEOUS EFFLUENTS RELEASE (Ci/yr)

Radionuclide	Containment Purge	PAB Venting	Turbine Venting	Main Condenser Off-Gas System	Gaseous Waste System	Total
H-3	3.7E+02*	3.7E+02	c	c	c	7.4E+02
C-14	1.0E+00	a	a	a	7.0E+00	8.0E+00
Ar-41	2.5E+01	a	a	a	a	2.5E+01
Kr-83m	a	a	a	a	a	a
Kr-85m	7.0E+00	2.0E+00	a	1.0E+00	a	1.0E+01
Kr-85	1.0E+00	a	a	a	2.6E+02	2.6E+02
Kr-87	2.0E+00	1.0E+00	a	a	a	3.0E+00
Kr-88	1.0E+01	4.0E+00	a	2.0E+00	a	1.6E+01
Kr-89	a	a	a	a	a	a
Xe-131m	3.0E+00	a	a	a	2.0E+01	2.3E+01
Xe-133m	1.6E+01	a	a	a	a	1.6E+01
Xe-133	8.6E+02	3.4E+01	a	2.1E+01	7.7E+01	9.9E+02
Xe-135m	a	a	a	a	a	a
Xe-135	3.1E+01	4.0E+00	a	3.0E+00	a	3.8E+01
Xe-137	a	a	a	a	a	a
Xe-138	a	1.0E+00	a	a	a	1.0E+00

* 3.7E+02 = 3.7×10^2

[a] Less than 1.0 Ci/yr for Noble Gases and C-14

[b] Less than 0.0001 Ci/yr for Iodine

[c] Less than 1.0 percent of total for this nuclide

Radionuclide	Containment Purge	PAB Venting	Turbine Venting	Main Condenser Off-Gas System	Gaseous Waste System	Total
I-130	b	b	b	b	b	b
I-131	1.5E-02	4.3E-03	1.8E-03	2.7E-02	b	4.8E-02
I-132	b	b	b	b	b	b
I-133	1.1E-02	6.3E-03	1.9E-03	4.0E-02	b	5.9E-02
I-134	b	b	b	b	b	b
I-135	b	b	b	b	b	b
Mn-54	2.2E-04	1.8E-04	c	c	4.5E-05	4.4E-04
Fe-59	7.4E-05	6.0E-05	c	c	1.5E-05	1.5E-04
Co-58	7.4E-04	6.0E-04	c	c	1.5E-04	1.5E-03
Co-60	3.3E-04	2.7E-04	c	c	7.0E-05	6.7E-04
Sr-89	1.7E-05	1.3E-05	c	c	3.3E-06	3.3E-05
Sr-90	2.9E-06	2.4E-06	c	c	6.0E-07	5.9E-06
Cs-134	2.2E-04	1.8E-04	c	c	4.5E-05	4.4E-04
CS-137	3.7E-04	3.0E-04	c	c	7.5E-05	7.4E-04

[a] Less than 1.0 Ci/yr for Noble Gases and C-14

[b] Less than 0.0001 Ci/yr for Iodine

[c] Less than 1.0 percent of total for this nuclide

TABLE 11A-4 VENT RELEASE INFORMATION FOR GASEOUS RELEASES

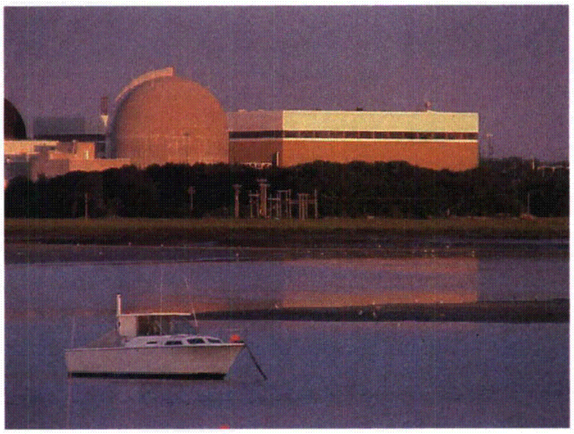
	PLANT VENT*	TURBINE BUILDING ROOF VENTS (10)	T.B. HEATER BAY ROOF VENTS (10)
Height above grade (ft)	185	151	100
Height above adjacent structures	5.5	0	0
Exit temperature (°F) Summer/Winter	104°/50°	145°/100°	145°/100°
Exit flow rate (cfm)(max)	312,185	50,000/40,000	
Exit velocity (ft/min)	1,950	1,470	1,290
Vent size and shape	9'x12' diameter stack open to the environment(exit gas is deflected down towards the roof)	10 mushroom type vents 5.5.ft diameter (exit gas is deflected down toward the roof)	10 mushroom type vents5 ft diameter (exit gas is deflected down toward the roof)
Deflectors or diffusers:	No	Yes	Yes

* Two samples are drawn from the plant vent stack; one is routed through the wide range radiation detectors and the other through a portable air sampler and released to the atmosphere at the PAB roof. Since these two samples are from a monitored release path and their flow rates are negligible (L3 scfm each), these release points are not separately described.

SEABROOK STATION
UPDATED FINAL SAFETY ANALYSIS REPORT

CHAPTER 11
RADIOACTIVE WASTE MANAGEMENT

FIGURES



See PID-1-WL-B20828

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Liquid Waste System Overview	
		Figure 11.2-1

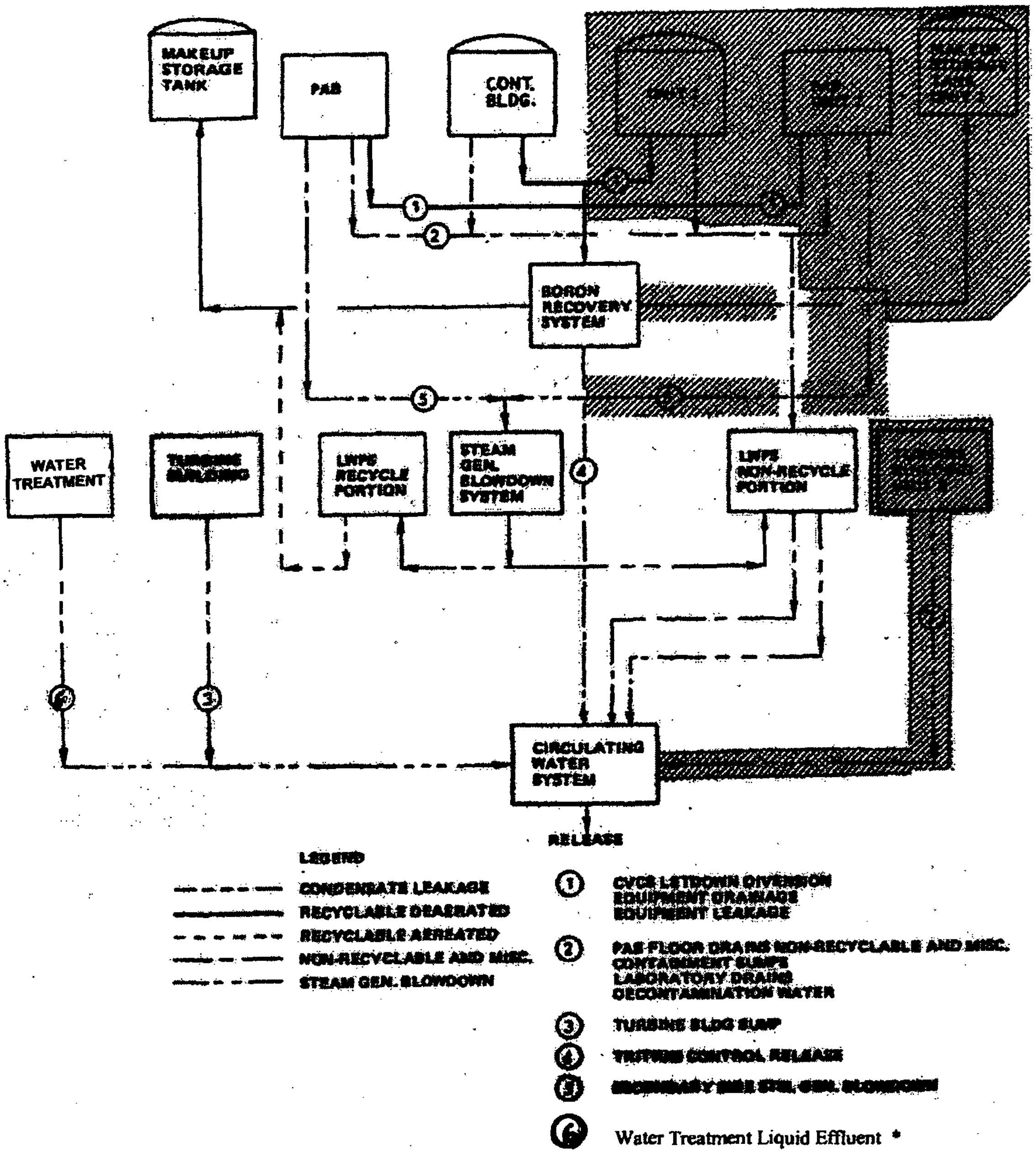
See PID-1-WL-B20829

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Liquid Waste System Storage and Filtration Detail	
		Figure 11.2-2

See PID-1-WL-B20830

See PID-1-WL-B20831

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Liquid Waste System Demineralization and Testing Detail	
		Figure 11.2-4



* includes Condensate Polishing System

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Radioactive Liquid Release Points	
		Figure 11.2-5

See PID-1-WG-B20768

See PID-1-WG-B20770

See PID-1-WG-B20771

See PID-1-WG-B20772

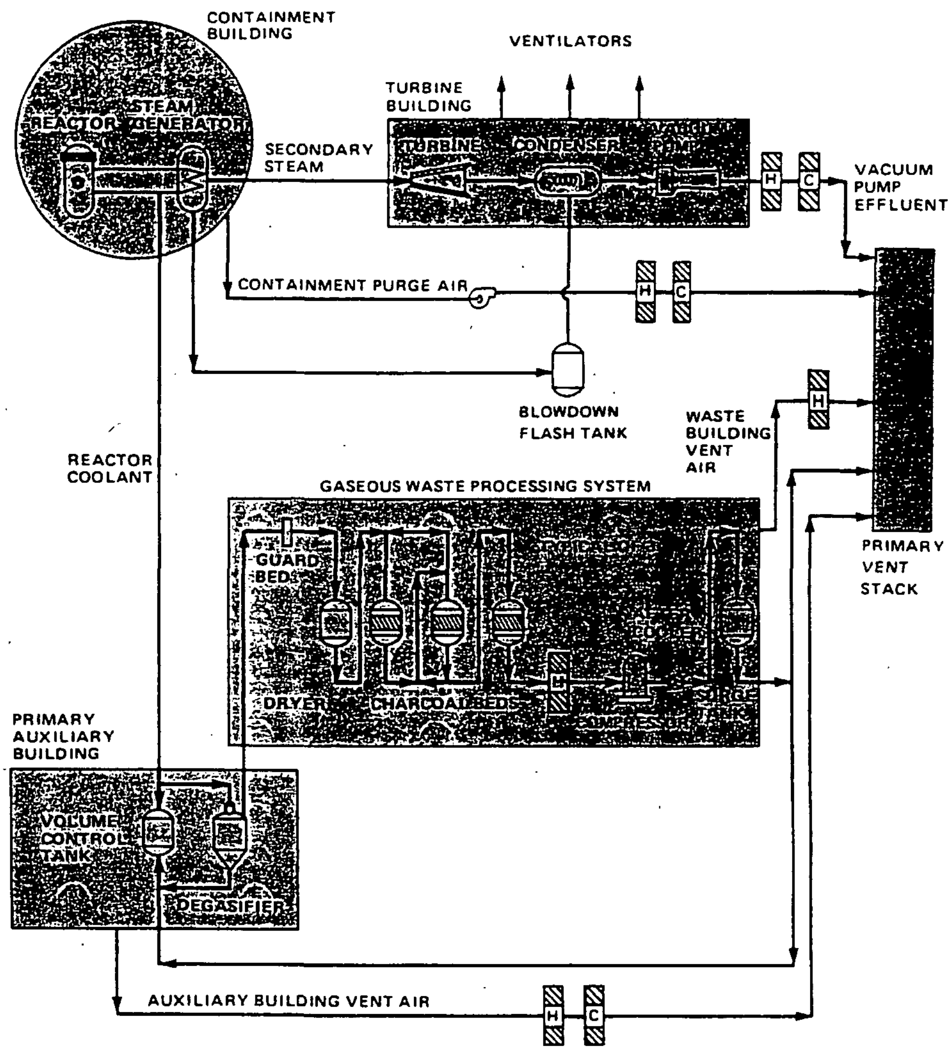
See PID-1-WG-B20773

See PID-1-NG-B20132

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Nitrogen Gas Overview	
		Figure 11.3-3

See PID-1-NG-B20135

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Nitrogen Gas Detail	
		Figure 11.3-4



LEGEND

H – HEPA FILTER
C – CHARCOAL FILTER

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Sources of Gaseous Waste	
		Figure 11.3-5

See PID-1-WS-B20733

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Solid Waste System Overview	
		Figure 11.4-1 Sh. 1 of 2

See PID-1-WS-B20734

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Solid Waste System Overview	
		Figure 11.4-1 Sh. 2 of 2

See PID-1-WS-B20735

See PID-1-WS-B20736

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Solid Waste System Waste Feed and Bottom Detail	
		Figure 11.4-3

See PID-1-WS-B20737

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Solid Waste System Asphalt and Steam Detail	
		Figure 11.4-4

See PID-1-WS-B20738

See PID-1-WS-B20739

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Solid Waste System Crystallizer Detail	
		Figure 11.4-6

See PID-1-WS-B20740

See PID-1-WS-B20741

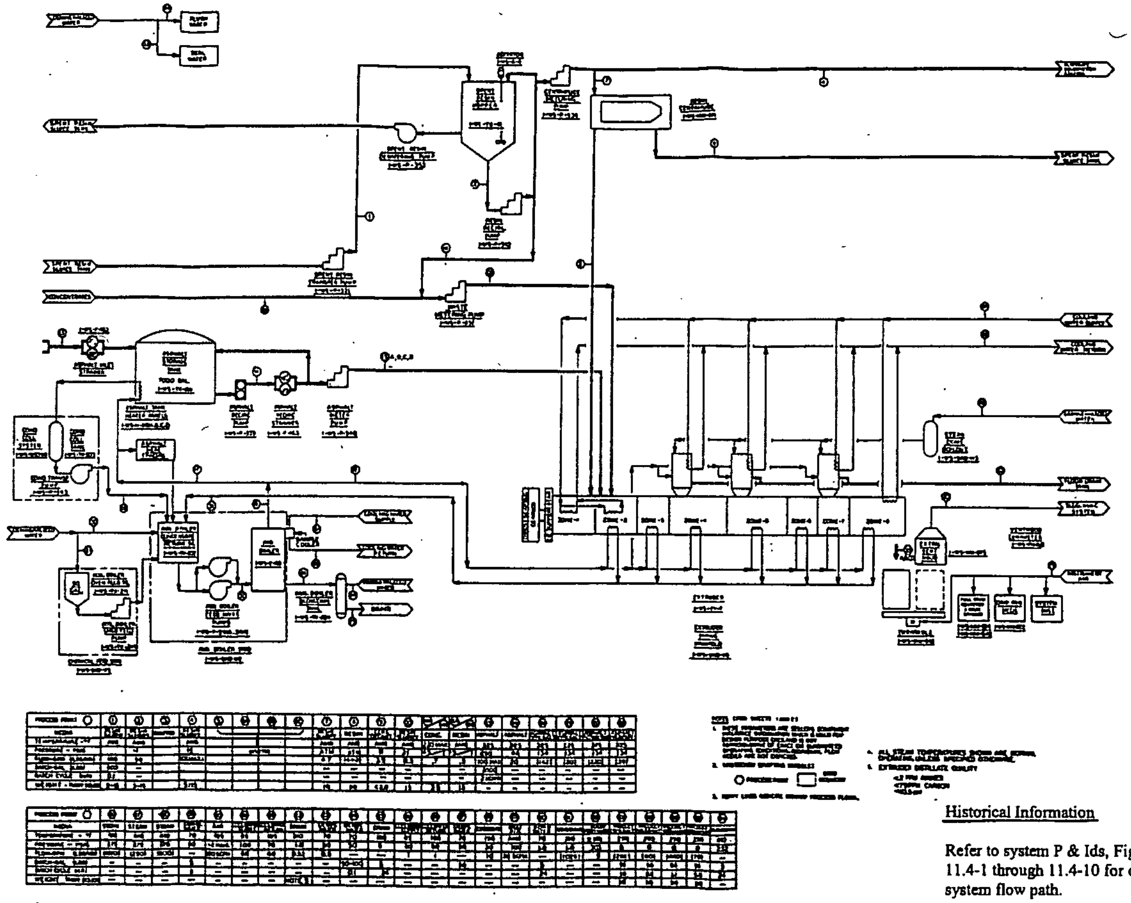
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Solid Waste System Caustic and Material Handling Detail	
		Figure 11.4-8

See PID-1-WS-B20742

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Solid Waste System Pump Seal Water Detail	
		Figure 11.4-9

See PID-1-RS-B20252

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Spent Resin Sluicing System	
		Figure 11.4-10



Historical Information

Refer to system P & Ids, Figures 11.4-1 through 11.4-10 for current system flow path.

SEABROOK STATION
UPDATED FINAL SAFETY
ANALYSIS REPORT

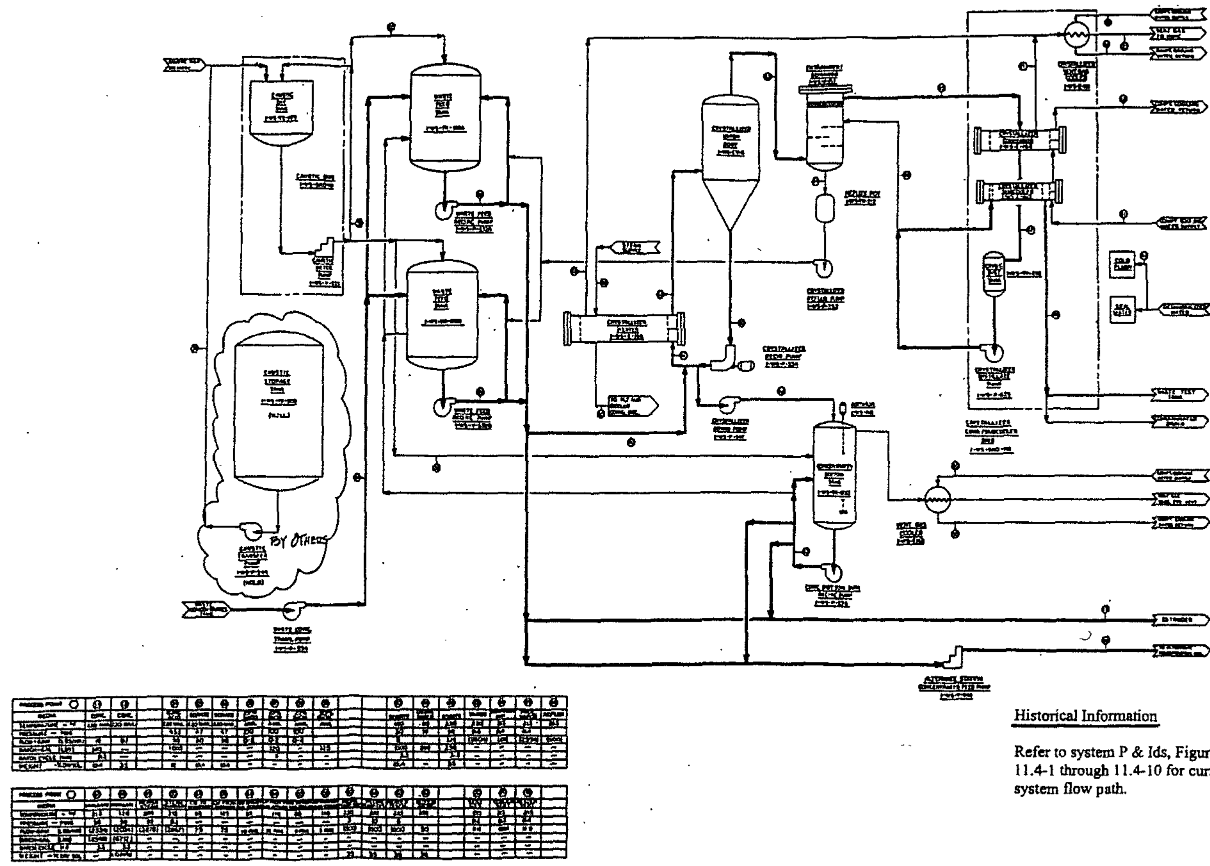
Solid Waste System Process Flow Diagram

Figure 11.4-11 Sh. 1 of 2

SEABROOK STATION
UPDATED FINAL SAFETY
ANALYSIS REPORT

Solid Waste System Process Flow Diagram

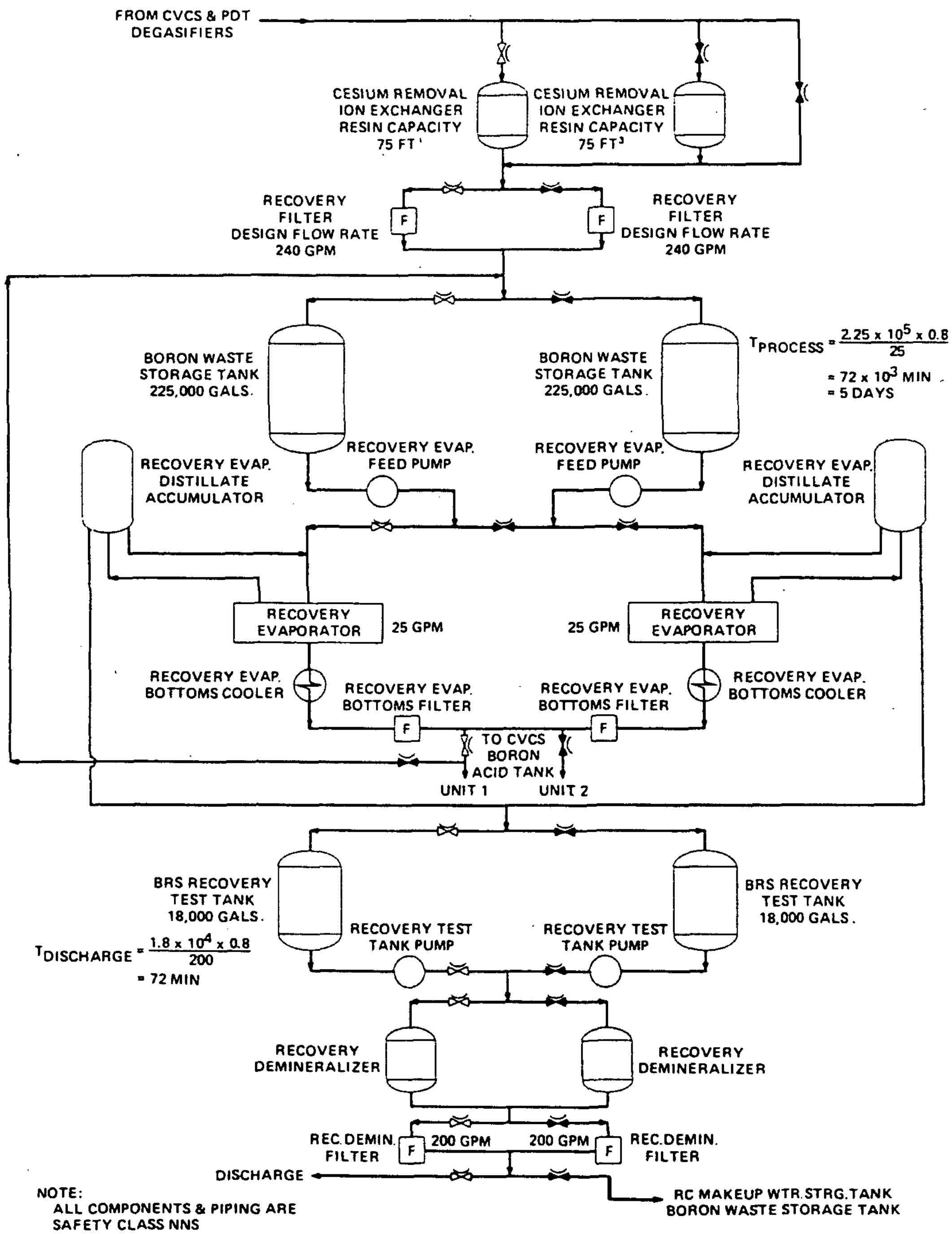
Figure 11.4-11 Sh. 2 of 2



Historical Information

Refer to system P & Ids, Figures 11.4-1 through 11.4-10 for current system flow path.

See 1-NHY-500015

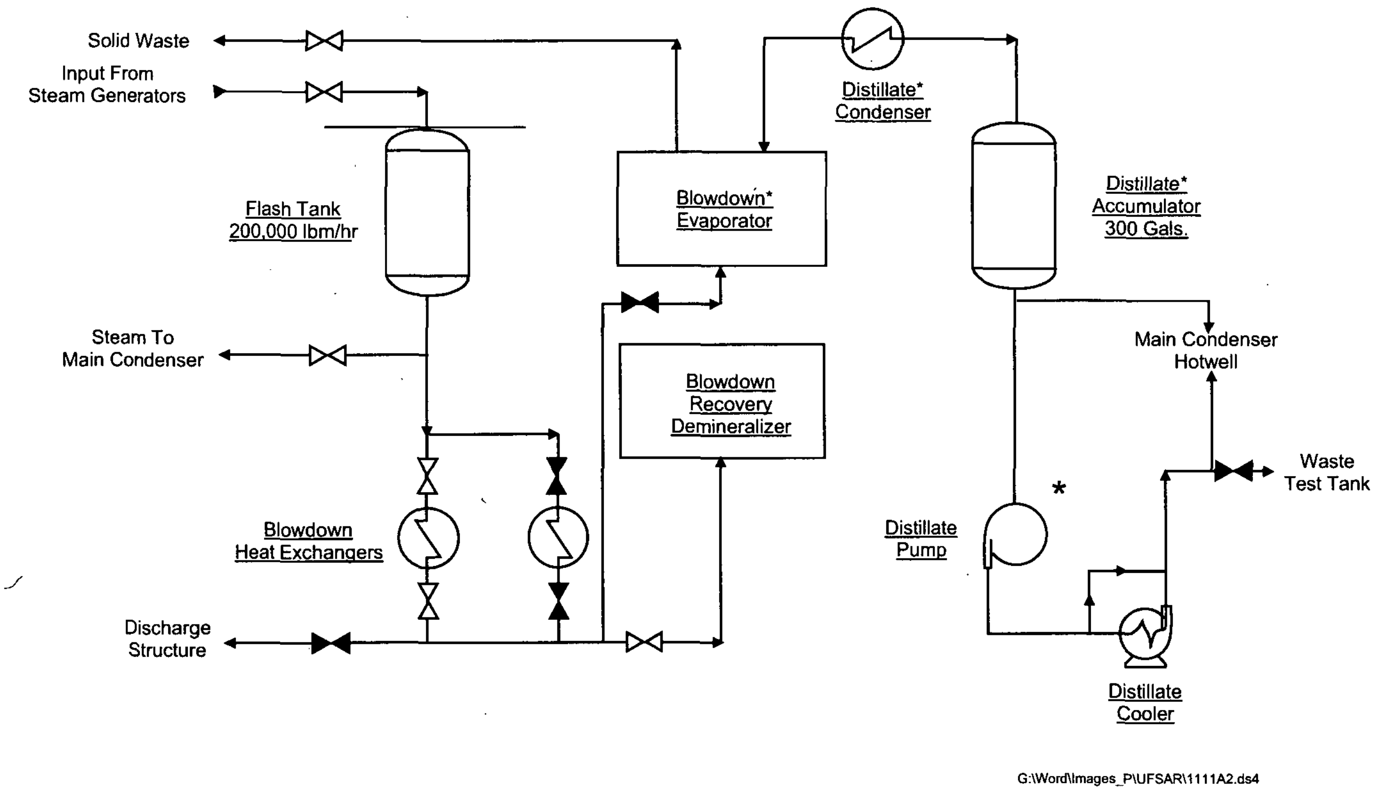


Seabrook Station Updated Final Safety Analysis Report	Boron Recovery System Schematic
	Figure 11A-1

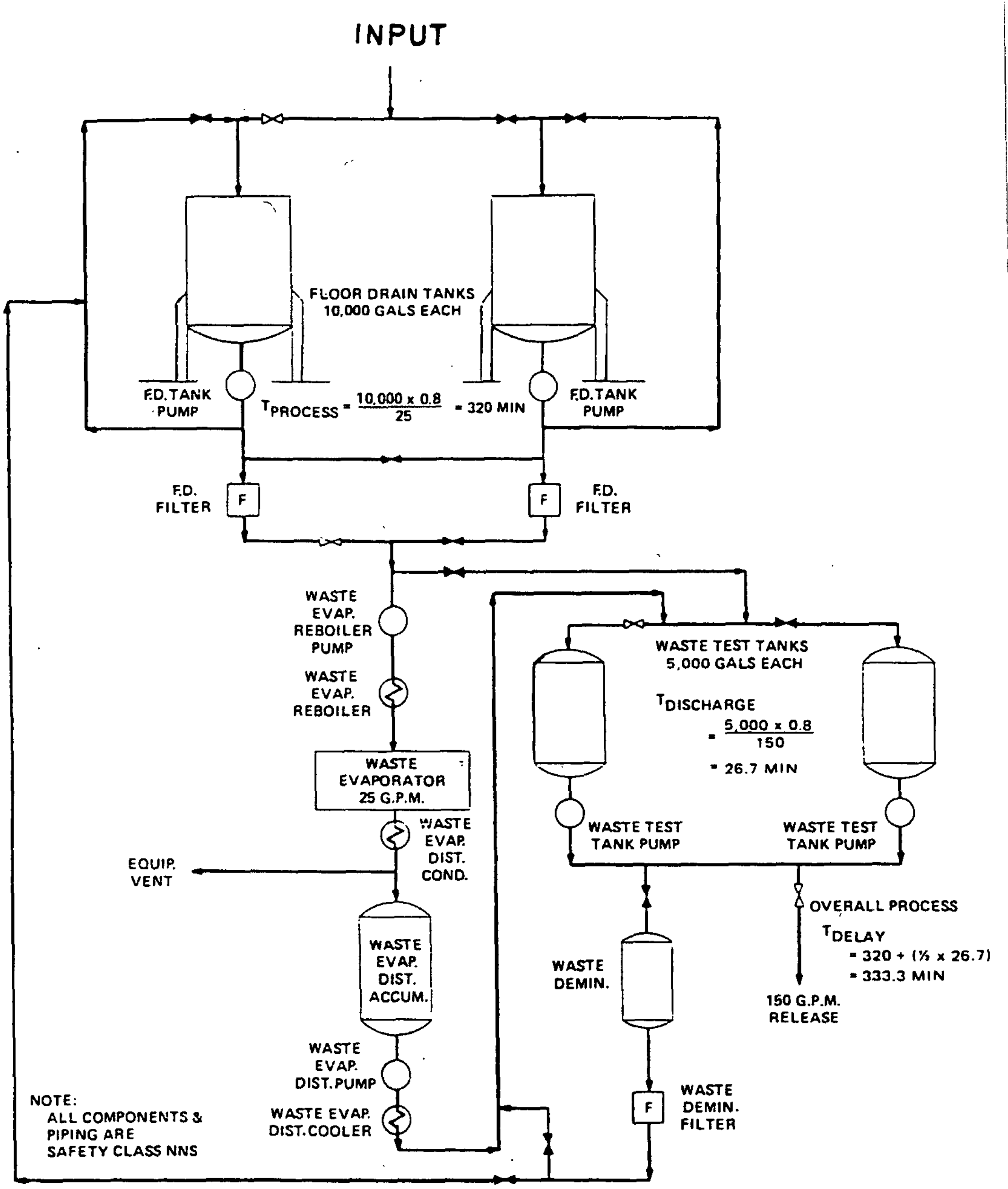
SEABROOK STATION
UPDATED FINAL SAFETY
ANALYSIS REPORT

Steam Generator Blowdown System Schematic

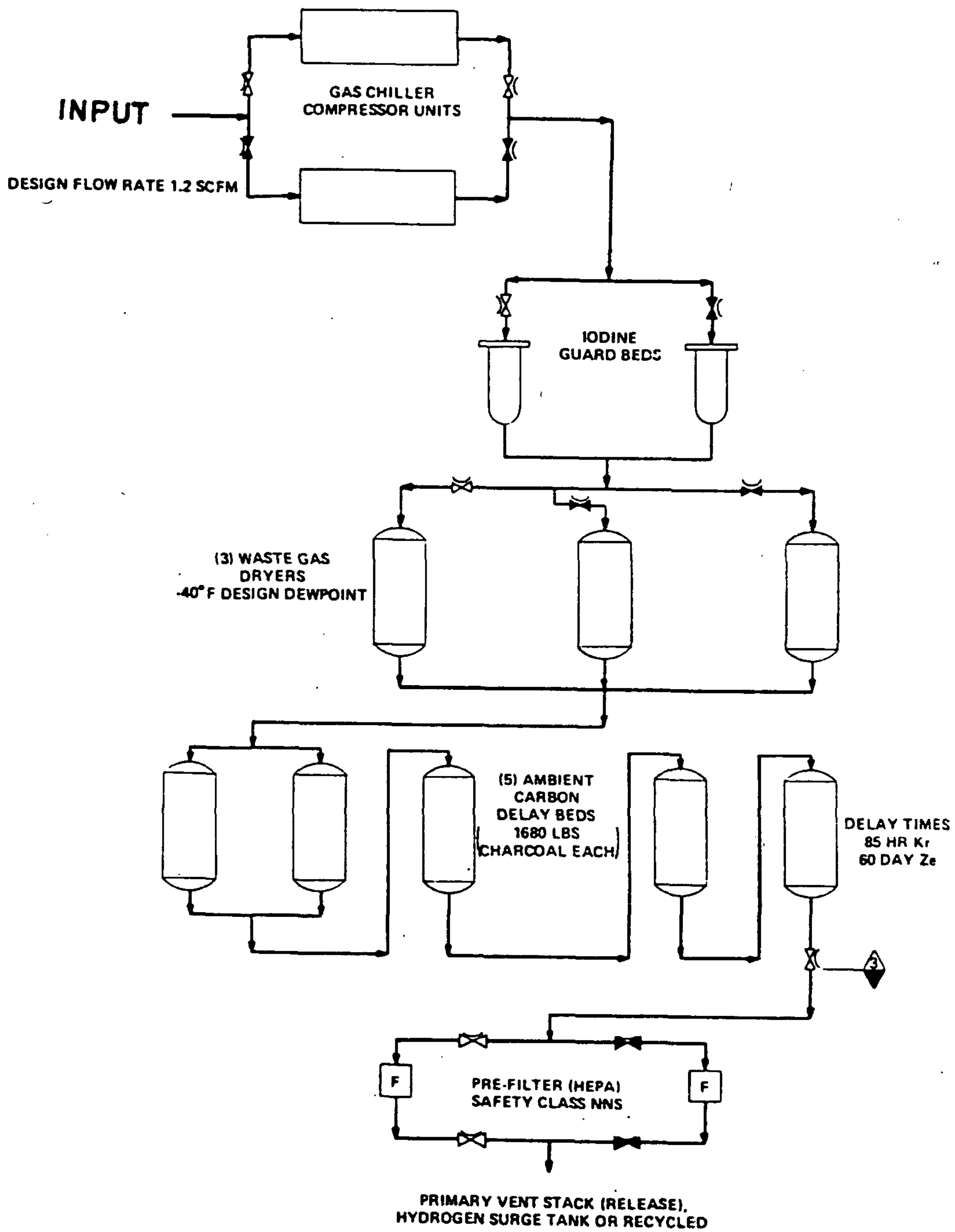
Figure 11A-2



* Evaporator Subsystem Not
Immediately Available For Use

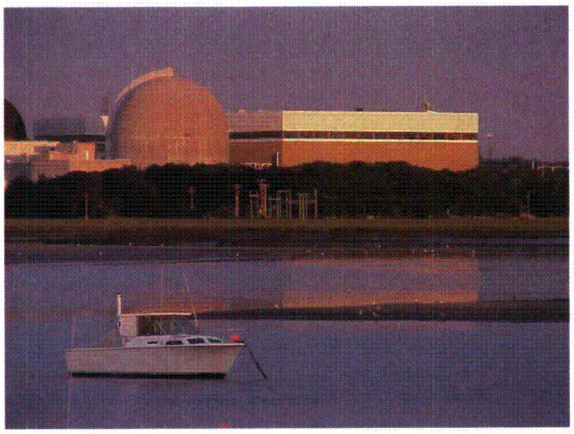


SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Liquid Waste System Schematic	
		Figure 11A-3


INPUT
DESIGN FLOW RATE 1.2 SCFM
GAS CHILLER COMPRESSOR UNITS
IODINE GUARD BEDS
(3) WASTE GAS DRYERS -40°F DESIGN DEWPOINT
(5) AMBIENT CARBON DELAY BEDS 1680 LBS (CHARCOAL EACH)
DELAY TIMES 85 HR Kr 60 DAY Ze
3
F
PRE-FILTER (HEPA) SAFETY CLASS NNS
F
PRIMARY VENT STACK (RELEASE), HYDROGEN SURGE TANK OR RECYCLED

SEABROOK STATION
UPDATED FINAL SAFETY ANALYSIS REPORT

CHAPTER 12
RADIATION PROTECTION



12.1 ENSURING THAT OCCUPATIONAL RADIATION EXPOSURES ARE AS LOW AS IS REASONABLY ACHIEVABLE (ALARA)

12.1.1 Policy Considerations

It is the policy of FPL Energy Seabrook to maintain occupational radiation exposure during Seabrook Station operations as low as is reasonably achievable (ALARA). This policy and philosophy of keeping exposure to radiation ALARA is derived from the fact that ionizing radiation is biologically damaging and that the amount of damage is related to, among other things, the magnitude of the dose received.

The ALARA philosophy, as embodied in Regulatory Guide 8.8, Revision 3 and 8.10, Revision 1, is implemented by employing sound radiation protection practices and techniques as delineated in the Radiation Protection Manual and Health Physics procedures. The implementation of the ALARA philosophy includes all phases of station operations from startup to eventual decommissioning. What is "reasonably achievable," for exposure reduction, is a judgment which all Seabrook Station management personnel are required to make. However, it is also the responsibility of station employees to make judgments regarding their radiation exposure during the performance of their assigned tasks in a radiologically controlled area.

The basis for these judgments should include an assessment of the state of technology and the economics of improvements in relation to all of the benefits from these improvements.

12.1.1.1 Overall ALARA Policy Responsibilities

The Station Director has the overall responsibility and authority for implementing the ALARA philosophy. He delegates this responsibility and authority to the Health Physics Department Supervisor. The Health Physics Department Supervisor ensures that the ALARA philosophy receives proper attention and that adequate resources are made available. He also reviews the progress in the area of exposure reduction, ensures that corrective actions are taken when necessary, and provides overall direction and coordination of the ALARA policy. This may include the following:

a. Participation in design reviews for facilities and equipment that can effect potential radiation exposures

b. Identification of situations that have potential for causing significant exposures to radiation

c. Implementation of an exposure control program (ascertain which jobs should be closely controlled for exposure purposes)

d. Development of procedures and methods for keeping radiation exposures of station personnel ALARA

e. Review of selected job procedures to maintain exposures ALARA

f. Participation in the development and implementation of training programs related to work in radiation areas or involving radioactive materials

g. Establishment of a radiation exposure surveillance program to maintain data on exposures of station personnel by job function and/or specific jobs

h. Performance of trend analyses of station radiological data such as contamination and radiation levels

i. On-the-job inspection of selected tasks in progress to review effectiveness of the ALARA policy

j. Documentation of ALARA efforts as determined necessary

k. Supervising, training and qualifying the radiation protection staff of the station

l. Ensuring that adequate radiation protection coverage is provided for station personnel during all working hours.

A Health Physics professional assists the Health Physics Department Supervisor in performing and coordinating the above ALARA functions and activities. Additionally, health physics personnel have the responsibility for providing exposure reduction guidance to workers during routine support interfaces. Subsection 12.5.1 provides a description of the Health Physics Department organization and responsibilities with regard to radiation protection.

12.1.1.2 Direct ALARA Policy Responsibility

Direct responsibility for implementation of the ALARA philosophy rests with each member of the Seabrook Station management organization. As managers, they are directly responsible for a defined area of the overall station operation and understand this responsibility includes minimizing radiation exposures for both their own personal and other station personnel. In keeping with the overall station goals of providing maximum availability, highest efficiency and the best working environment possible, management will ensure that station personnel perform all radiologically controlled area tasks in accordance with general and/or job-specific procedures and the ALARA philosophy.

This is accomplished through careful consideration of the following general guidelines:

a. When required, develop clear, easily understood, job specific procedures

b. Ensure that work being performed by procedure is accomplished on an orderly and timely basis

c. Plan and schedule work to ensure that it does not interfere with other work in progress in the same area and that individuals not required for the work are not assigned

d. Ensure that workers are familiar with the work assignments, requirements, procedures, and locations

e. Modify methods, techniques and procedures on a timely basis to account for changes in technology, modified equipment and/or identification of inadequacies

f. Provide training to personnel as necessary to improve skills and minimize dependence on certain individuals for performance of specific tasks

g. Investigate and specify the use of exposure-reducing techniques wherever practical and reasonable

h. Coordinate activities and procedures with the Health Physics Department to insure adequate review.

12.1.1.3 Employee Responsibilities

Radiation workers, whether permanent or temporary, are informed of their responsibilities for maintaining ALARA exposure for both themselves and fellow workers. This includes the responsibilities to notify supervisors of procedural changes that should be considered to reduce exposure and to report to radiation protection personnel potential radiological hazards that may result in unnecessary exposure. Employees are expected to actively participate in training programs developed for their specific disciplines and assigned major exposure tasks.

12.1.1.4 Training

To ensure all personnel fully understand the necessity of minimizing their exposure as well as their fellow workers' exposure, a radiation safety training program is provided. The Manager of Nuclear Training is responsible for the conduct of the training program and will coordinate inputs by the Health Physics Department Supervisor and other department supervisors regarding content and concepts. As a minimum, radiation workers are given a basic course in radiation protection, station requirements and federal regulations to understand the requirements for entering a radiologically controlled area.

Additionally, detailed radiation protection instructions are provided to those individuals whose duties require working with radioactive materials, entering radiologically controlled areas (RCA), or directing the activities of personnel who work with radioactive materials or enter radiologically controlled areas. This instruction emphasizes exposure reduction techniques and the ALARA philosophy.

The above training programs and other training programs that are conducted at Seabrook Station are outlined in Section 13.2. All instruction, whether the subject matter pertains to radiation safety, plant systems or craft skills, is intended to result in a training program that promotes the ALARA philosophy through improved workmanship and reliability.

12.1.2 Design Considerations

This subsection deals with the station layout, equipment location, and equipment maintainability, as applicable to the ALARA concept. The objectives of the design considerations, which are consistent with Regulatory Guide 8.8, Section C.3, are to reduce the number of personnel needed to perform work in a radiation area associated with maintenance activities, reduce maintenance time, and minimize radiological conditions. The designers have employed experience from past designs and operating plants to help reduce exposure from components and work on components. Reviews have been conducted by personnel with experience in radiation protection at operating power plants.

12.1.2.1 Reduction of Work Force Exposure to Radiation

There are three basic design considerations which allow a reduction of the work force in a radiation area. The design considerations are:

a. Reliability

b. Location

c. Maintainability.

Each of the above items can be subdivided many times, but emphasis will be placed on generalized designs and considerations.

Emphasis has been given to the reliability of equipment. Increasing the reliability of equipment, by design, decreases the need for inspection, maintenance, and preventive maintenance. When redundancy is used to increase reliability, the maintenance duration and location may not change, but maintenance can usually be performed when radiological conditions are more conducive to ALARA.

Equipment location is used where practical to eliminate exposure by placing equipment in nonradiologically controlled areas. When equipment is placed in the radiologically controlled area, then, when practical, the equipment is located away or shielded from substantially higher radiation areas. Additional reductions in exposure are made possible by locating equipment so that a minimum number of personnel are necessary to perform maintenance; i.e., adequate working and removal space are provided.

The work force necessary to perform maintenance on equipment is dependent on the ease of repairing the equipment. Selection of some equipment and systems is partially based on the ease of repair.

12.1.2.2 Work Time Reduction in the RCA

The time necessary to perform tasks is reduced where practical. The time reduction is accomplished by advanced technology and good general designs, such as equipment location, ventilation, and remote tools.

The use of advanced technology allows inspection and repair work to be performed faster. For example, advanced technology is responsible for an increased speed in performing in-service inspection (ISI) of steam generators and steam generator tube plugging. Improved design of valves seals allows less and faster maintenance. There are many examples of the use of advanced technology in the Seabrook design helping to reduce the plant man-rem figure. The above two examples were chosen due to the high exposure rate associated with the work.

Good quality general designs have been used to reduce the man-rem expenditure. The reduction in man-rem exposure is due to parts of equipment fitting together properly without the need for modifications, along with other components and systems such as the ventilation system described in Subsection 12.3.3. This is extremely effective in reducing exposure due to removal and replacement of insulation for repair and ISI work.

12.1.2.3 Reduction of Radiological Conditions

The facility general design has been reviewed by experienced personnel to ensure that systems, equipment (described in Subsection 12.3.1), ventilation (described in Subsection 12.3.3), and shielding design (described in Subsection 12.3.2) provides acceptably low exposures. The following design features have been incorporated in the RCA where reasonably possible:

a. Shielding of pipe chases and equipment containing radioactive materials (permanent and provisions for temporary)

b. Placement of reach rods for remote control

c. Placement of switches for remote control of equipment

d. Use of low nickel and low cobalt alloys to reduce cobalt problems

e. Leak detectors to reduce the amount of leakage and contamination by rapid detection and, therefore, appropriate action

f. Remote control panels where appropriate, such as for rad-waste processing and fuel transferring

g. Design of pipes for low or no crud trap problems

h. Floor drains for moving radioactive liquids to the Waste Processing System

i. Penetrations are stepped, shielded, or are out of line of site with the source when possible

j. Radiation Monitoring System to detect changes in radiation levels (discussed in Subsection 12.3.4)

k. Limiting personnel access to areas by barricading and/or locking, (discussed in Subsection 12.5.3.3)

l. Provisions for flushing and purging of contaminated systems.

There are many methods of reducing the exposure magnitude which do not require permanent station modification. The following is a partial list of methods available to reduce the exposure rate to workers:

a. Prevention of contamination

b. Decontamination of equipment

c. Removal of components to lower radiation zone for work

d. Before transferring highly contaminated equipment the equipment should be decontaminated, or the contamination should be prevented from spreading by placing the equipment in a container.

e. Use of portable shielding

f. Provisions for air and water filtration.

12.1.2.4 ALARA Design Changes

Periodic review of plant design and equipment for ALARA considerations has resulted in the following changes:

a. Restricted access to the RHR vaults via ladders in the containment building spray heat exchanger room

b. Restricted access to the waste gas regenerative compressor room, waste gas dryer columns, and waste gas valve room in the Waste Processing Building

c. Rearrangement of the primary sample heat exchanger and sink to reduce shine

d. Rearrangement of the evaporator equipment to reduce exposure levels in adjacent walkways.

12.1.2.5 Management of Radiation Protection Design Review - Construction Phase

The Seabrook Station ALARA program for construction, design changes, and reviewing field run piping was the joint responsibility of Westinghouse Electric Corporation (Westinghouse), United Engineers and Constructors Inc., (UE&C), Yankee Atomic Electric Company - Nuclear Services Division (YAEC), and New Hampshire Yankee (NHY).

Westinghouse was responsible for the design, fabrication and delivery of the Nuclear Steam Supply System, related auxiliary systems and the nuclear fuel. Technical direction for the installation of this equipment and technical assistance throughout the preoperational testing, initial core loading and testing programs were further responsibilities of Westinghouse.

United Engineers and Constructors (UE&C) was responsible for the engineering, design and construction management of the station. Included in their scope were the supply and installation of the balance of plant systems, components, and structures so that a complete and integrated facility was assured.

The radiation design review performed by UE&C was the responsibility of the Chief Power Engineer. The Chief Power Engineer ensured that an overall design program was implemented to help maintain occupational radiation exposures As Low As Reasonably Achievable (ALARA) during operation of the facility. He provided appropriate guidance to Chief Discipline Engineers regarding ALARA design implementation and verified implementation.

The chief discipline engineers (electrical, mechanical, instrumentation, etc.) provided for incorporation of ALARA considerations. This was accomplished by providing guidance to engineers responsible for the design of Seabrook Station. The Chief Engineers or designers reviewed the various systems to ensure provided guidance was used.

The Quality Assurance Manual defined contractor responsibilities as follows:

> "Each contractor shall maintain design control measures as required by ANSI N45.2.11. These design measures shall be applied to areas such as the following: ... accessibility for in-service inspection, maintenance and repair..."

The YNSD Project Office established appropriate reviewers as determined by the Quality Assurance Manual and Section 17.1 of the Seabrook Station Updated FSAR. Project policies indicate primary and secondary reviewers of UE&C specifications; Westinghouse specifications; UE&C system descriptions;

Westinghouse system descriptions; Updated FSAR chapters, sections, and subsections, engineering changes and general arrangement drawings of the Containment, Fuel Storage and Primary Auxiliary Building. Project policies also indicated the type of documentation required for reviews. The documentation was in the form of Engineering Review Reports, memoranda or other reports.

a. Design Reviewers

YNSD Radiation Protection personnel did not always possess the necessary expertise to perform complete ALARA reviews. The YNSD Radiation Protection Group relied on engineers with the expertise to determine equipment compatibility, accessibility (ladders, platforms, laydown space), operability and maintenance history (low-maintenance). Individuals performing reviews were usually Engineer grade or higher (B.S. degree or equivalent and 3 years professional experience). Individuals within departments who performed the reviews were chosen based on their general knowledge of the system and equipment. The departments within YNSD who performed reviews are listed below with some of their responsibilities:

1. Plant Engineering Department (Instrumentation and Control Group, Electrical Engineering Group, Mechanical Engineering Group and Systems Engineering Group)

   - Supported the Project Office in general and detailed technical review and guidance for plant concept, design construction and licensing in the field of Fluid Systems, Instrumentation and Control, Electrical Engineering, Mechanical Engineering, Systems Engineering, Materials Engineering and Structural Engineering.

   - Coordinated electrical and control design between the architect-engineer and nuclear steam system supplier.

   - Reviewed conceptual design and detailed engineering of all assigned primary and secondary fluid systems, included types of components selected, modes of operation and physical arrangements.

   - Reviewed all electrical and control equipment specifications, logic and wiring diagrams. These reviews included transformers, motors and switchgear, plant control devices, nuclear instrumentation and reactor control and protection system equipment.

2. Nuclear Engineering Department (PWR Transient Analysis Group, Reactor Physics Group and LOCA Analysis Group)

   - Reviewed all reactor physics design work performed by the reactor supplier to assure adherence to the design criteria and the use of adequate methods and assumptions.

   - Reviewed and/or participated in the design of instrumentation for core monitoring.

   - Verified that operational requirements were given adequate consideration and were appropriately factored into the design.

   - Assisted in the development of Technical Specifications and station operating procedures for accident conditions.

   - Reviewed the various station anticipated transients and accidents to ensure conformance with all applicable criteria.

   - Reviewed and analyzed data from the station to verify the reactor physics design.

3. Fuel Management Department (Nuclear Materials Group, Economic Analysis Group and Core Components Group)

   - Reviewed and approved mechanical designs and specifications for nuclear fuel assemblies and components.

   - Reviewed specifications, procedures, purchase orders and drawings for proper definition of quality assurance requirements for nuclear fuel assemblies.

4. Environmental Engineering Department (Radiological Engineering Group, Radiation Protection Group, Environmental Sciences Group and Environmental Laboratory)

   - Established functional requirements of engineered safeguard systems and evaluation of their performance.

   - Participated in the design and review of solid, liquid and gaseous radioactive waste treatment systems.

- Participated in the establishment of system design requirements for plant process and area Radiation Monitoring Systems.
- Participated in the establishment of Station Radiological Equipment and Facilities.

12.1.2.6 Independent Reviews

The ongoing interactions between NHY, YAEC, Westinghouse and UE&C engineers and radiation protection personnel resulted in continuous cross-checking or "independent" reviews of each organization's design, construction and operational activities. Proposals to modify or establish designs received appropriate levels of review by these diverse organizations.

Additionally, these organizations recognized that professional contractor organizations are available for use, as necessary, to provide assistance in specialized areas of radiation protection.

Contractor organizations have been used to provide specialized evaluations in such cases as the analysis of the proposed removable shielding for the reactor vessel annulus and the assessment of the radiological impact of the spent resin transfer system design. Such special evaluations significantly contributed to design review efforts directed toward ensuring that occupational exposures are maintained ALARA.

12.1.2.7 Field Reviews

The later example, cited in Subsection 12.1.2.6, is an instance of the onsite ALARA reviews conducted during the construction phase under the auspices of the Seabrook Station Health Physics Department. These evaluations were performed on systems, components and areas in accordance with station ALARA policies.

These evaluations were used to identify potential beneficial modifications, as well as to provide background information for use during operations. Expertise and assistance was obtained, as necessary, from other applicable station departments.

Coordination of major actions and the final decisions were the responsibility of appropriate station and corporate management. YNSD and, when necessary, UE&C were party to these station activities.

Day-to-day aspects of this ALARA effort were conducted by station Health Physics Technicians supervised by Health Physics Working Foremen under the cognizance of health physics supervisors. A Health Physicist - ALARA also participated in the daily efforts as well as coordinating long-term ALARA-related activities. Minimum experience and educational qualifications for Health Physics Department personnel are described in Subsection 12.5.1.

The onsite ALARA field review program, in concert with ALARA design considerations addressed by Westinghouse, UE&C, YNSD and selected professional contractors, as discussed in previous sections, is used by Seabrook management to ensure that the construction and operation of Seabrook Station results in occupational radiation exposures that are as low as reasonably achievable.

12.1.3 Operational Considerations

Operational considerations concern the station personnel efforts of ensuring the application of the ALARA philosophy to both day-to-day routines, and periodic tasks such as special maintenance, in-service inspections and refueling. In all cases, these efforts are consistent with the established radiation protection program.

12.1.3.1 General Operational Concept

As described in Subsection 12.1.1, the responsibilities for maintaining exposures ALARA does not rest with just one or several individuals, but rests with all personnel involved with planning, supervising or implementing work in radiological areas. The health physics department is charged with the overall responsibility and likewise, will be the most influential and informed department for specifying and ensuring the use of exposure-reduction techniques and methods by station personnel wherever and whenever practicable.

When a task is scheduled to be performed on any system that contains, collects, stores or transports radioactive liquids, gases and/or solids, the specific current and/or anticipated radiological conditions involved with that task will be determined. Health Physics factors the task complexity, difficulty, actual and/or projected radiological conditions, and a knowledge of plant systems into the specification of radiation protection and control requirements for the performance of the job. In addition, health physics provides input into the job planning and preparations so the job can be performed in a radiologically safe manner and with the least practical exposure. Where possible, the total exposure expenditure for the job (collective dose of all those personnel planned to be involved) is estimated. Such estimates permit manpower planning by management, but more importantly, they aid in the identification of those jobs that would likely result in a substantial total exposure and accordingly, where a significant dose reduction could be realized. Means to reduce exposure which could then be employed include, among others, the following:

a. Preoperational briefing by health physics personnel

b. Development of special procedures

c. Specific training on a mock-up of a component, equipment or structure

d. Development and use of special tools

e. Use of localized ventilation

f. Decontamination of equipment/areas

g. Draining, flushing or filling of pipes or components

h. Use of contamination containing devices (i.e., liquid collection, drop cloths, containments or tents)

i. Removal of equipment to areas of lower dose rates

j. Prefabrication of complicated equipment or structure

k. Radiological surveillance during a job to identify changing conditions

l. Provision for periodic supervision

m. Contracting specialist services.

In most instances, the collective exposure from using any of the above techniques or others and from the actual performance of a job must be considered to ensure actions are feasible and will not result in an increase of total exposure (although this may be necessary in special circumstances). Additionally, post-operation debriefings may be used, as appropriate, to identify what problems were encountered and to determine how the job techniques may be modified to reduce exposure in the future.

12.1.3.2 <u>Procedure Development</u>

Each department supervisor is responsible for developing department procedures that ensure compliance with applicable regulations. These procedures, developed as discussed in Section 13.5, also provide instructions to personnel for the performance of routine and special tasks that may include work in a radiologically controlled area of the station. These tasks include, among others, refueling operations, radioactive waste handling, in-service inspections, process sampling and surveillance, instrument calibrations and maintenance. Radiation protection supervisory personnel review procedures involving work with radioactive materials or work to be performed in radiation or high radiation areas, as defined in the Radiation Protection Manual.

Jobs with a significant radiological risk (e.g., above a projected dose threshold) undergo a documented planning and review process. This planning includes a review of procedures and techniques, work site provisions, worker preparation and training, and radiological considerations which can minimize the radiological risk to the worker. Personnel with relevant expertise are given prime responsibility for planning such jobs and documenting their preparations, and Health Physics supervisory personnel oversee and approve the final preparations.

12.1.4 Decommissioning Considerations

Seabrook Station relies on preparations in several areas to ensure that occupational radiation exposures are maintained as low as reasonably achievable (ALARA) during plant decommissioning.

As discussed in the following subsection, Seabrook incorporates design features that offer significant exposure reduction during decommissioning. These design features are merely the basis for the performance of an ALARA-oriented radiation protection program during plant operations, as well as effective ALARA preparations during decommissioning planning.

An important aspect of the decommissioning procedure is the use of specific ALARA practices tailored to deal with the particular decommissioning method employed. Delineation of these specific ALARA practices (including engineering design modifications) takes place during decommissioning planning, after information concerning the specific decommissioning method becomes available. Consistent with the guidance provided by 10 CFR 20 and Regulatory Guide 8.8, Revision 3, the specific practices implemented will be based on "as assessment of the state of technology and economic considerations" prevalent at the time of decommissioning. The state of technology and the economics that will prevail 40 years in the future are unknown factors and, therefore, performance of a cost benefit analysis is precluded at this time.

The commitment to formulate and implement the ALARA philosophy during decommissioning includes an acknowledgement and understanding that the process of preparing for eventual decommissioning with occupational exposure as low as reasonably achievable is ongoing in nature.

12.1.4.1 Design Features Contributing to ALARA during Decommissioning

Many basic ALARA design features incorporated into Seabrook Station for operations, maintenance and refueling will enhance exposure reduction during those phases and also during decommissioning, regardless of the specific decommissioning procedure. In effect, this is a generic, ALARA approach to operations and decommissioning.

Specific design features that will be used to maintain occupational radiation exposures ALARA include:

a. A separate building exists for waste processing and disposal that ensures availability of waste processing facilities while other systems and components are being maintained or dismantled.

b. The main hoist of the polar crane has been derated to 302 tons. The crane must be refurbished before it is capable of removing the reactor vessel or the steam generators. There is adequate capacity as is to lift the pressurizer. These components can be removed with minimal displacement of permanent concrete shielding to afford its maximum effectiveness.

c. Seabrook containments are equipped with 27-foot diameter equipment hatches that facilitate removal of large equipment intact.

Generic design features used to maintain occupational radiation exposure ALARA include:

a. <u>Location of Equipment</u>

As stated in Updated FSAR Subsection 12.1.2.1, paragraph (4), "Equipment location is used, where practical, to eliminate exposure by placing equipment in nonradiation control areas." This philosophy is embodied in the segregation of areas with radioactive systems and components. The design philosophy "to minimize the extent of areas housing radioactive equipment and piping through efficient arrangement of equipment and systems" as stated in Updated FSAR Subsection 12.3.1b.

b. <u>Equipment Accessibility and Removability</u>

Updated FSAR Subsection 12.1.2.1 indicates that equipment is designed and located to maximize accessibility to facilitate rapid, efficient work. Equipment is also designed and placed to enhance removal operations and thus, minimize exposure time.

c. <u>Plant Layout</u>

Updated FSAR Subsection 12.3.1.3a explains that "plant layout includes optimal location of radioactive components."

d. Flush/Drain Connections

The provision of flush and drain connections on many systems and components enables extensive chemical decontamination prior to operating phase maintenance and later, decommissioning.

e. Corrosion Control

Internal accumulation of radioactive material is limited through effective corrosion-control methods. Careful selection of plant materials and an aggressive chemistry control program greatly reduce source terms that must be dealt with during operating phase maintenance and decommissioning.

12.2 RADIATION SOURCES

The primary source of neutron and gamma radiation is the reactor core and Primary Coolant System. The secondary sources are contained in, or emanate from, the auxiliary systems and components not included as part of the Nuclear Steam Supply System. Radioactivity in these systems generally originates from:

- Fission products that have escaped into the coolant and are either deposited or carried into other systems, and
- Corrosion products activated while passing through the reactor core.

The basis for calculation of radiation source terms for shielding purposes for the various plant systems for normal operation is the concentrations of fission and corrosion products in the reactor coolant during operation at one percent failed fuel. The bases for calculation of radioisotope concentrations in the reactor coolant, including assumed failed fuel percentage and the applicability of ANSI N237 and Regulatory Guide 1.112, are discussed in Subsection 11.1.1. Source terms for accident conditions are addressed in Chapter 15.

The radiation sources and associated input parameters, assumptions, and methodology described in Section 12.2 represent those used to establish original shielding design. The radiation sources utilized for the design of the primary shield were developed by the reactor vendor and are based on a standard four loop Pressurized Water Reactor. The radiation sources in the spent fuel and the N-16 source were also developed by the reactor vendor, and are based on a reactor power of 3565 MWt. The radiation sources in the primary coolant system and the auxiliary systems are based on a reactor power of 3654 MWt, a one-year fuel cycle, and 1 percent fuel element defects.

The impact on plant shielding requirements was evaluated for an analyzed reactor power of 3659 MWt and operation with an 18-month fuel cycle. This represents a minor change from the original design basis. The conservative analytical techniques used to establish the original shielding requirements, and the Station Technical Specification which will restrict the reactor coolant activity to levels significantly less than the 1 percent fuel defects, ensure that operation at the licensed core power level will have no significant impact on shielding requirements and safe plant operation.

With the exception of Table 12.2-11, Table 12.2-27 and Table 12.2-29, data contained in Table 12.2-1, Table 12.2-2, Table 12.2-3, Table 12.2-4, Table 12.2-5, Table 12.2-6, Table 12.2-7, Table 12.2-8, Table 12.2-9, Table 12.2-10, Table 12.2-11, Table 12.2-12, Table 12.2-13, Table 12.2-14, Table 12.2-15, Table 12.2-16, Table 12.2-17, Table 12.2-18, Table 12.2-19, Table 12.2-20, Table 12.2-21, Table 12.2-22, Table 12.2-23, Table 12.2-24, Table 12.2-25, Table 12.2-26, Table 12.2-27, Table 12.2-28, Table 12.2-29, Table 12.2-30, Table 12.2-31, Table 12.2-32, Table 12.2-33, Table 12.2-34, Table 12.2-35, Table 12.2-36, and Table 12.2-37 are independent of one or two unit operation. The data contained in Table 12.2-11, Table 12.2-27, and Table 12.2-29 are based on a two-unit operation with the relevant assumptions provided in the appropriate sections.

12.2.1 Contained Sources

The sources contained in the equipment of the Primary, Auxiliary and Radioactive Waste Management Systems and all other major sources of radiation during normal operation are described in this chapter. The components of these systems are represented by cylinders approximating their actual geometry in the computer code used to calculate shielding requirements. The location of all equipment containing radioactive sources is shown in Figure 12.3-1, Figure 12.3-2, Figure 12.3-3, Figure 12.3-4, Figure 12.3-5, Figure 12.3-6, Figure 12.3-7, Figure 12.3-8, Figure 12.3-9, Figure 12.3-10, Figure 12.3-11, Figure 12.3-12, Figure 12.3-13, Figure 12.3-14, Figure 12.3-15 and Figure 12.3-16.

12.2.1.1 Source Terms for Reactor Core and Spent Fuel Pool

a. Source Terms for the Reactor Core

The major contribution of neutron radiation levels external to the reactor biological shield is due to fast neutron leakage from the reactor cavity during power operation. Reactor shielding design is based on a reactor vendor-supplied source term, the result of a quadrature analysis using the DOT III-W code. The source term consists of fifty-three angular fluxes at 216 mesh intervals for thirteen neutron energy groups at the reactor vessel surface. Neutron and gamma flux spectrums are given in Table 12.2-1 and Table 12.2-2.

b. Source Terms for Spent Fuel Pool Shielding

Shielding design source terms for the high density spent fuel pool are based on the "worst-case" of a full core of 193 fuel assemblies with 100 hours decay, and are given in Table 12.2-3.

12.2.1.2 Source Terms for Spent Fuel Transfer

Shielding for spent fuel transfer is based on a source term supplied by the reactor vendor, shown in Table 12.2-3.

The source strength given represents an average fuel assembly with four days decay after an irradiation time of 3.1 years. The fuel assembly is approximately 152 inches long and 8.5 inches square in cross section.

The bounding source term used to evaluate minimum water depth requirements to maintain dose rates below 2.5 mrem during fuel movement in the spent fuel pool is shown in Table 12.2-3. The source strength given represents a peak fuel assembly with 80 hour decay after an irradiation time of 69,000 Mwd/Mtu. The fuel is approximately 152 inches long and 8.5 inches square in cross section.

12.2.1.3 Sources in the Reactor Coolant System (RCS)

a. Reactor Coolant System Nitrogen-16

The N-16 activity of the coolant is the controlling radiation source in the design of the RCS secondary shielding, and is given in Table 12.2-4 as a function of transport time in a reactor coolant loop.

b. Deposited Corrosion Products

The most significant radiation sources encountered during normal maintenance and inspection of most plant equipment (pumps, heat exchangers, tanks, valves, and other out-of-core primary equipment) are deposits from the reactor coolant, such as activated corrosion products and some fission products.

The activity of the deposits is predominantly due to Co-58 and Co-60. It is estimated that 50 to 90 percent of personnel radiation exposure can be attributed to Co-58 and Co-60, each of which contributes about equally to the exposure.

Cobalt in the Reactor Coolant System is minimized by limiting the cobalt content of materials in contact with the reactor coolant. Low cobalt materials are specified for the steam generator, pressurizer, reactor coolant pump, reactor coolant loop piping, and reactor core internal structures. In addition, chemical treatment and analysis techniques are used to minimize cobalt buildup. At plant operating conditions, the primary coolant chemistry is designed to inhibit the corrosion of materials in contact with reactor coolant. At refueling shutdowns, during the cooldown period, hydrogen peroxide (H_2O_2) can be added to the Reactor Coolant System to oxidize and solubilize Co-58 and Co-60. The solubilized cobalt is removed, by demineralization, prior to lifting the reactor head. These techniques minimize the concentration of Co-58 and Co-60 in the refueling water and reduce personnel exposures during the refueling operation.

12.2.1.4 Sources in the Chemical and Volume Control System (CVCS)

One purpose of the CVCS is to provide continuous purification of the reactor coolant water. Major equipment items include the regenerative and letdown heat exchangers, mixed-bed and cation-bed demineralizers, reactor coolant filter, letdown degasifier, volume control tank, and charging pumps. The boron thermal regeneration (BTR) subsystem contains the three BTR heat exchangers and the BTR demineralizers. The seal water subsystem for the reactor coolant pumps includes the injection and return filters, and the seal water heat exchanger.

The most important ion exchangers in the CVCS, from a shielding standpoint, are the mixed-bed and the cation-bed demineralizers. The mixed-bed demineralizers are normally in continuous use. They remove fission products in cation and anion form, and are very effective in removing corrosion products. The cation-bed demineralizer is used intermittently to remove lithium for pH control, and is very effective in removing the monovalent cations cesium and rubidium.

Noble gases are removed from the letdown purification flow by the letdown degasifier and are processed by the Radioactive Gas Waste System. During periods when the letdown degasifier is in service with oxygenated letdown, residual fission gases may be discharged to the plant vent via the aerated vent header.

The boron thermal regeneration demineralizers are used to regulate the boron concentration in the reactor coolant water during load follow operations to remove boron from the coolant as the nuclear fuel is depleted. These demineralizers collect any remaining radioactive anions, such as iodine and bromine, which may have passed through the mixed bed demineralizer.

The regenerative and excess letdown heat exchangers are located in the Containment Building. They provide the initial cooling for the reactor coolant letdown. Their radiation sources include Nitrogen-16.

The letdown heat exchanger provides secondary cooling for the reactor coolant letdown flow prior to its entering the demineralizers. The seal water heat exchanger cools water from several sources, including the reactor coolant discharged from the excess letdown reheat heat exchanger. During boron release, the tube side of the letdown reheat heat exchanger heats the letdown water before the water enters the boron thermal regeneration (BTR) demineralizers.

The geometry of the various components is described in Table 12.2-6. Locations of the equipment in this system are shown in Figure 12.3-1, Figure 12.3-5, Figure 12.3-6 and Figure 12.3-7.

The source term for each component in this system has been developed individually, with regard to its function and relationship to the rest of the system. The bases and assumptions for the calculation of the source terms for the CVCS equipment follow:

a. <u>Regenerative Heat Exchanger</u>

   This component contains reactor coolant concentrations of fission and corrosion products as well as N-16 activity. The shielding source term is given in Table 12.2-5 and was used for both tube and shell sides of the heat exchanger.

b. <u>Excess Letdown Heat Exchanger</u>

   The source term used for shielding this heat exchanger (tube side) is the same as that used for the regenerative heat exchanger. The shell side contains cooling water.

c. Letdown Heat Exchanger

The tube side of this heat exchanger contains reactor coolant concentrations of radioisotopes given in Table 11.1-1. The shell side contains cooling water.

d. Mixed Bed Demineralizers and Cation Bed Demineralizer

The source terms for these components, given in Table 12.2-5, are calculated through buildup and decay of fission and corrosion products from the reactor coolant over a one-year period. Anion and cation removal efficiency of 100 percent is assumed.

e. Letdown Degasifier Regenerative Heat Exchanger

The source term for the tube side is demineralized reactor coolant, while the shell side is the same less the noble gases. Isotopic source strengths are given in Table 12.2-5.

f. Letdown Degasifier Preheater

The shielding for this component is based on the tube side containing demineralized reactor coolant and the shell side containing cooling water.

g. Letdown Degasifier

This component is represented by three different source terms: a vapor region, a liquid-vapor region, and a liquid region. The liquid region is demineralized reactor coolant less the noble gases.

The vapor region is assumed to be occupied with noble gases and carried-over iodine. The concentrations are based on an input flow of 120 gpm, a gas removal of 0.25 scfm and partition factors of 1 for noble gases and 0.0075 for iodines. The source term for the liquid-vapor region is assumed to be demineralized reactor coolant including noble gases.

h. Letdown Degasifier Recirculation Pump and Letdown Degasifier Trim Cooler

The source term for these components is the same as that for the degasifier liquid region.

i. Volume Control Tank

The source terms for this tank are based on input from the excess letdown stream which is pure reactor coolant isotopic concentrations. The liquid region source term is assumed to be reactor coolant less the noble gases. The equilibrium concentrations for the vapor region are calculated using an input flow rate of 120 gpm and a purge rate of 0.7 scfm. The volume control tank stripping efficiency is assumed to be 0.4.

j. Charging Pumps

The source term for these pumps is the same as that for the liquid region of the volume control tank.

k. Seal Water Heat Exchanger

The tube side of this heat exchanger has reactor coolant concentrations of isotopes. The shell side contains cooling water.

l. Moderating Heat Exchanger

The fluid on the tube side of this component contains demineralized reactor coolant. The shell side has the same fluid except for the iodine concentration. Iodine is assumed to be released by the boron thermal regeneration demineralizers after an accumulation period of 12 hours.

m. Letdown Chiller Heat Exchanger

This component contains demineralized reactor coolant (tube side) and cooling water (shell side).

n. Letdown Reheat Heat Exchanger

This heat exchanger contains fluid with reactor coolant concentrations on the tube side and demineralized reactor coolant on the shell side.

o. Thermal Regeneration Demineralizer

The input to these demineralizers is reactor coolant which has been processed by the mixed bed demineralizers. The thermal regeneration demineralizers are anion ion exchangers which accumulate iodines.

The buildup of iodine is based on a flow rate of 120 gpm of fluid in which 10 percent of the reactor coolant iodines pass through the mixed bed demineralizers.

Accumulation time is assumed to be 12 hours. Demineralizer efficiency is taken to be 100 percent.

p. System Liquid Filters

Shielding for the reactor coolant filter and the demineralizer prefilter is based on a contact dose rate of 500 rem/hr. All other filters in this system are shielded for a contact dose rate of 100 rem/hr.

12.2.1.5 Sources in the Residual Heat Removal System

Shielding for the components of the Residual Heat Removal System is based on shutdown source terms shown in Table 12.2-7. These source terms are the reactor coolant isotope concentrations with a four-hour delay during which the Chemical and Volume Control Purification System operates.

The components of this system are represented in the computer shielding code as cylinders approximating the actual geometries. The dimensions of these cylinders are given in Table 12.2-8.

The locations of the components are shown in Figure 12.3-4.

12.2.1.6 Sources in the Steam Generator Blowdown System

The input to the Steam Generator Blowdown System consists of steam generator secondary side blowdown at a rate of 75 gpm. Concentrations of isotopes on the secondary side are determined by a primary-to-secondary leak rate of 500 gpd. Isotopic composition of system input fluid is given in Table 12.2-9. Geometry of the equipment in this system is contained in Table 12.2-10. The location of the system components is given in Figure 12.3-7 and Figure 12.3-12.

The basis for the isotopic composition of the input to the steam generator blowdown evaporator trains is for an anticipated operational occurrence involving a primary-to-secondary leak rate of 0.5 gpm for a duration of 90 days. The location of the evaporator trains is shown in Figure 12.3-12.

a. Blowdown Flash Tank

The source concentration of the liquid is 10/7 of the system input fluid concentrations. The isotopic concentration in the steam is calculated using a carryover fraction of 0.05 and a specific volume for saturated steam at 60 psia.

b. Flash Tank Bottoms Cooler

The source strength of this component is the same as the liquid phase for the flash tank.

c. Flash Steam Condenser and Cooler

The source strength of this component is based on the carryover fraction from the flash tank and is shown in Table 12.2-9.

d. Flash Tank Distillate Pumps

The source term for these pumps is the same as that for the condenser/cooler.

e. Blowdown Evaporator

The inventory of the bottoms fluid is a 90-day accumulation of the fission and activation products from a 0.5 gpm reactor coolant leak. Specific activity is based on 1300 gallons of bottoms.

f. Bottoms Pump and Bottoms Cooler

Specific source term for these components is the same as the evaporator bottoms.

g. Evaporator Distillate Condenser, Distillate Accumulator and Distillate Pump

The source term for evaporator distillate is calculated using decontamination factors of 10^0 for noble gases, 10^2 for iodine and 10^3 for other fission and activation products.

h. Demineralizers

The Steam Generator Blowdown System demineralizers are not utilized when the secondary side of the plant has significant contamination as noted for the conditions above. For incidental levels of contamination on the secondary side, the Blowdown Demineralizer System may be utilized as long as the general area radiation levels adjacent to the demineralizer vessels are maintained within the Zone II radiation limits.

12.2.1.7 Sources in the Boron Recovery System

The input to this system has been processed by the letdown purification system and is, therefore, demineralized and degasified. The isotopic source terms for this fluid are given in Table 12.2-11.

Input flow rate is assumed to be 341 lb/hr per unit during normal operation and 200 gpm for a unit during shutdown. System equipment geometries are described in Table 12.2-12.

Location of equipment is shown in Figure 12.3-8, Figure 12.3-9, and Figure 12.3-10.

a. Primary Drain Tank Demineralizers

Source term for this component buildup considers normal bleed rate (0.68 gpm/unit) for a period of one year (7008 hrs. at 80 percent capacity factor) followed by a refueling shutdown. The isotopic content is given in Table 12.2-11. Ion removal efficiency is assumed to be zero for iodines and unity for all other isotopes.

b. Boron Waste Storage Tanks

Maximum anticipated inventory for a boron waste storage tank occurs with a full tank following a shutdown. The fluid has been processed by the cesium removal ion exchanger for which the following D.F.s were assumed; 10^0 for iodines and 10^1 for all other isotopes. The source strength for this tank is shown in Table 12.2-11.

c. Recovery Evaporator Feed Pump

Isotopic source strength for these pumps is assumed to be the same as the input to the boron waste storage tank.

d. Recovery Evaporator

Bottoms concentrations of isotopes are calculated on the basis of processing one full boron waste storage tank at a feed flow rate of 25 gpm. Distillate concentrations are based on assumed D.F.s of 10^3 for iodine and 10^2 for other solids.

e. Recovery Evaporator Bottoms Pump, Recovery Evaporator Bottoms Cooler, Recovery Evaporator Reboiler Pump and Recovery Evaporator Reboiler

The shielding for these components is based on the source term for the evaporator bottoms concentration. This source term is shown in Table 12.2-11.

f. Recovery Evaporator Distillate Condenser, Recovery Evaporator Distillate Accumulator, Recovery Evaporator Distillate Pump, Recovery Evaporator Distillate Cooler and Recovery Test Tank Pump

The shielding for these components is based on the distillate source term described in Table 12.2-11.

g. Recovery Test Tank

Source term calculations for this component are based on an input flow rate of 25 gpm of distillate condensate for buildup and decay through a 12-hour filling period. Activities for this component are presented in Table 12.2-11.

h. Recovery Demineralizer

Source term calculation for this component considers a one-year buildup from recovery test tank input and a 100 percent efficiency of removal for all isotopes. The maximum inventory of activity is given in Table 12.2-11.

i. Recovery System Filters

Shielding for all filters in this system is based on 100 rem/hr. contact dose rate.

12.2.1.8 Sources in the Primary Drain System

The input to this subsystem of the Boron Recovery System is the drainage from various primary side equipment. The isotopic composition and strength is assumed to be untreated reactor coolant including all noble gases and iodine. This source term is given in Table 1.1-1. The geometry of each component of the system is described in Table 12.2-14. The location of the equipment in this system is shown in Figure 12.3-8, Figure 12.3-9, and Figure 12.3-10.

a. Primary Drain Tank (PDT), Primary Drain Tank Transfer Pump, Primary Drain Regenerative Heat Exchanger and Primary Drain Tank Degasifier Preheater

The isotopic concentrations in the source terms for these components are assumed to be the same as the reactor coolant as given in Table 11.1-1.

b. Primary Drain Tank Degasifier

Shielding for this degasifier is based on two different source terms, one for the intermediate liquid-steam region and the lower all liquid region and the second term for the upper gas region. The liquid and liquid-steam source term is assumed to be reactor coolant (undemineralized) less the noble gases. The upper region source term is taken to be noble gas and iodine. Isotope concentrations for this component are given in Table 12.2-13.

c. PDT Degasifier Recirculation Pump and PDT Degasifier Trim Cooler

The source term for these components is the same as the liquid source term for the degasifier.

d. PDT Degasifier Prefilter

Shielding for this filter is based on 100 rem/hr contact dose rate.

12.2.1.9 Sources in the Spent Resin Sluicing System

The Spent Resin Sluicing System collects the spent resin from all the demineralizers and ion exchangers of the nuclear plant. The geometry of the components of this system is described in Table 12.2-16. The location of this equipment is shown in Figure 12.3-8 and Figure 12.3-9.

a. Spent Resin Sluice Tank and Spent Resin Transfer Pump

The source term for this equipment is based on the accumulation of the fission and activation products from the reactor coolant inventory over a period of one year. Isotopic source strength is given in Table 12.2-15. Those isotopes with half-life of less than one day are neglected unless produced from a long-lived isotope.

b. Spent Resin Sluice Pump

This pump is protected from processing the spent resin by a strainer in the spent resin sluice tank. However, the source term for this pump assumes 100 ppm of spent resin escapes the strainer, giving a source term which is 10^{-4} that of the spent resin tank.

12.2.1.10 Sources in the Spent Fuel Pool Cleanup System

The design basis source term for the equipment in this system was derived assuming mixing of the water in the spent fuel pool with the water in the containment refueling pool during fuel transfer. Shielding calculations use cylindrical approximations of equipment geometries. Dimensions for the components in this system are given in Table 12.2-18. The location of the equipment in the Spent Fuel Pool Cleanup System is shown in Figure 12.3-5.

a. Spent Fuel Pool Demineralizer

Shielding for this demineralizer is based on the peak activity level calculated for accumulation and decay of isotopes following initiation of the cleanup loop. This calculation includes a four-day cleanup period of the reactor coolant followed by dilution with refueling water before mixing of reactor cavity and spent fuel pool water. No credit was taken for the diluting effect of the spent fuel pool water. Spent fuel pool cleanup flow rate is 120 gpm. Demineralizer efficiency is assumed to be 100 percent for all isotopes. The design source term for the spent fuel pool demineralizer is given in Table 12.2-17.

b. Spent Fuel Pool Demineralizer Prefilter and Spent Fuel Pool Demineralizer Post-filter

Shielding for these filters is based on 100 rem/hr contact dose at change-out.

12.2.1.11 Sources in the Miscellaneous Chemical Drain System

The input to this system is primarily from the drains in the chemistry lab. The shielding source term was developed from expected sample volumes and frequencies. Composition of the input is shown in Table 12.2-19. Geometry of system components is given in Table 12.2-20. Location of equipment in this system is shown in Figure 12.3-8 and Figure 12.3-17.

a. Chemical Drain Tank

The shielding source term for this tank is based on the system input concentrations at an average input flow rate of 103 gpd. Consideration is given to buildup and decay during the filling of this tank. The design inventory of a full chemical drain tank is given in Table 12.2-19.

b. Chemical Drain Transfer Pump

This pump is used to transfer the contents of the chemical drain tank. Shielding is therefore based on the same source term.

c. Chemical Drain Treatment Tank

The major source of fluid for this tank is the chemical drain tank. The source term is based on buildup and decay during the filling of this tank at a rate of 103 gallons per day.

12.2.1.12 Source Terms for Water Storage Tanks

The isotopic inventory of the water storage tanks is given in Table 12.2-21 and Table 12.2-22. The location of these tanks is shown in Figure 12.3-10. The physical geometry of these components is described in Table 12.2-23.

a. Reactor Makeup Water Storage Tank

The inventory of this tank is based on 1 percent fuel clad defects and reactor coolant recycled by the Boron Recovery System.

b. Refueling Water Storage Tank

The inventory given in Table 12.2-22 is based on the refueling water storage tank refilled with cavity flood water immediately following a refueling. The reactor coolant radionuclide concentrations prior to the refueling preparation correspond to 1 percent failed fuel and the duration of the cavity flood condition is assumed to be 20 days. No credit for other decay is assumed.

12.2.1.13 Source Terms for Liquid Waste System

The source terms used in shielding calculations for the equipment in the Liquid Waste System are shown in Table 12.2-24. Each component is represented by a cylinder approximating its actual geometry. The dimensions of the cylinders are shown in Table 12.2-25. The location of this equipment is shown in Figure 12.3-8, Figure 12.3-9, and Figure 12.3-10.

a. Floor Drain Tank

The concentrations of radionuclides that form the shielding source term for the floor drain tank are given in Table 12.2-24. The source term is developed by buildup and decay calculations using an input activity of 0.075 reactor coolant at a flow rate of 120 gpm. The floor drain tank has a capacity of 10,000 gallons.

b. Floor Drain Tank Pumps

Since the floor drain tank pumps serve to recirculate the water contained in the floor drain tanks, they may, at times, contain some reactor coolant without full dilution. Therefore the shielding is based on an assumed activity of 0.1 reactor coolant.

c. Liquid Waste Evaporator

The source term for shielding the liquid waste evaporator bottoms is calculated with consideration of buildup and decay using the following parameters:

1. The input activity is the same as the floor drain tank concentrations.
2. Feed flow rate is 25 gpm.
3. Concentration time is 100 hours.

The source term for the evaporator distillate is based on a feed flow rate of 25 gpm and decontamination factors of 10^3 for iodine, 10^4 for other fission and activation products and 10^0 for noble gases.

d. Liquid Waste Evaporator Distillate Condenser

The liquid waste evaporator distillate condenser contains both vapor phase distillate and liquid phase condensate. The distillate source term is the same as that for the liquid waste evaporator, while the condensate does not contain the noble gases.

e. Liquid Waste Evaporator Distillate Accumulator, Pump and Cooler

The source term for these components is the same as that for the condensate in the liquid waste evaporator distillate condenser.

f. Waste Test Tanks

The source term for the waste test tanks (WTT) are evaluated by calculating the buildup and decay of isotopes, given the input activity of evaporator distillate condensate at a rate of 25 gpm. Each WTT has a capacity of 25,000 gallons.

g. Waste Test Tank Pumps

The WTT pumps are used at times for WTT recirculation. These pumps may then contain liquid with the activity of the WTT feed, which is the same as the liquid waste evaporator distillate condenser condensate.

h. Waste Demineralizer

Calculation of the source term for the waste demineralizer considers buildup and decay of radioisotopes over a one-year service period. The input activity is the same as the WTT concentration at a flow rate of 2,750 gpd. This demineralizer is assumed to have a 100 percent efficiency for removal of solids.

i. Liquid Waste Evaporator Reboiler, Reboiler Pump, Bottoms Pump and Bottoms Cooler

The source terms for this equipment are the same as that for the liquid waste evaporator bottoms described in Subsection 12.2.1.13c.

j. Liquid Waste System Filters

All filter cartridges will be changed when the dose rate reaches 100 rem/hr on contact. This value forms the basis for shielding calculations.

k. Skid-Mounted Waste Liquid Processing System

The skid-mounted system is designed and changeout criteria will maintain the equipment room area outside the shield wall at ≤ a zone III Radiation Area (≤ 15 mr/hr).

12.2.1.14 Source Terms for Radioactive Gaseous Waste System (RGWS)

There are three major sources of activity which serve as input to the Radioactive Gas Waste System: the letdown degasifier (Units 1 and 2) and the primary drain tank degasifier. Each degasifier produces effluent at a rate of 0.25 scfm; therefore the shielding source terms for the system are based on a gas flow rate of 0.75 scfm and are shown in Table 12.2-26 and Table 12.2-27. The RGWS also processes gas from the hydrogenated vent headers; however, this gas would dilute the concentrations of radionuclides and has not been included in the system input. The geometry of RGWS components is described in Table 12.2-28. Equipment locations are shown in Figure 12.3-12 and Figure 12.3-13.

a. Waste Gas Chillers

The concentrations in the waste gas chillers are the same as the system input. The inventory of isotopes for these components is given in Table 12.2-27.

b. Waste Gas Dryers

The specific isotopic shielding source term for these components is assumed to be the same as that for the waste gas chillers, i.e., taking no credit for iodine removal by the iodine guard beds. Each dryer contains 29 percent aluminosilicate by volume. The calculated isotopic inventory is presented in Table 12.2-27.

c. Iodine Guard Beds

The equilibrium accumulation of iodines at a gas flow of 0.75 scfm is used for the shielding source term for these components. The isotopic inventory used for shielding calculations is presented in Table 12.2-27.

d. Carbon Delay Beds

The source term for these components is based on a delay per bed of 12 days for Xenon and 17 hours for Krypton at a flow rate of 0.75 scfm. The shielding inventory for each of the beds is given in Table 12.2-27.

e. Hydrogen Surge Tank

Shielding for this tank is based on the inventory of isotopes given in Table 12.2-27. This source term assumes that the tank is filled with gas at the specific activity calculated for the outlet of the fifth carbon delay bed at tank design pressure.

f. Regenerative Compressor

Shielding for this component is based on the same specific source term as for the waste gas dryers, given in Table 12.2-27.

g. HEPA Filter

The source term for this filter was calculated on the assumption that one percent of the inventory in the last carbon delay bed is transported with charcoal fines to the filter cartridge. No credit is taken for decay of isotopes after leaving the carbon delay bed. The shielding design inventory is given in Table 12.2-27.

12.2.1.15 Source Terms for Solid Waste Management System

The source terms used for shielding calculations for the equipment in the Solid Waste Management System are listed in Table 12.2-29; the shielding geometry and dimensions for this equipment are presented in Table 12.2-30. The locations of this equipment are shown in Figure 12.3-8, Figure 12.3-9, Figure 12.3-10, Figure 12.3-11, and Figure 12.3-12.

a. Waste Concentrates Tank, Waste Concentrates Transfer Pump

The waste concentrates tank has a working capacity of 6000 gallons, and can hold roughly four batches of evaporator bottoms. The shielding source terms are based upon two batches of bottoms from the boron recovery evaporator, and one batch each from the liquid waste evaporator and the steam generator blowdown evaporator. No radioactive decay during transit has been considered.

b. Waste Feed Tanks and Waste Feed Recirculation Pumps

The input stream to these tanks contains about 12 percent solids by weight; after preparation/processing in the tanks, the output stream contains about 10.4 percent solids by weight. The shielding source terms for this equipment are about 0.87 of those for the waste concentrates tank described in a. above.

c. Spent Resin Transfer Pump, Spent Resin Hopper, Spent Resin Recirculation Pump, and Resin Centrifuge Metering Pump

The shielding source terms for this equipment are the same as those for the spent resin sluice tank described in Subsection 12.2.1.9.

d. Spent Resin Centrifuge

The input resin slurry contains about 15 percent of resin by weight; after the dewatering process, the remaining contents at the discharge from the centrifuge are about 50 percent resin by weight (with no transport water). Therefore, the shielding source terms for the spent resin centrifuge are 3.3 times those for the spent resin hopper described in c. above.

e. Waste Crystallizer/Evaporator, Crystallizer Recirculation Pump, and Crystallizer Drain Pump

The input waste concentrates contains about 10.4 percent solids by weight; after crystallizer/evaporation, the remaining slurry is about 35-50 percent total dissolved solids by weight. Therefore, the shielding source terms for this equipment are 2.9 times those for the waste feed tanks described in b.

f. Concentrates Bottom Tank, Concentrates Bottom Tank Recirculation Pump, Waste Metering Pump, and Alternate Station Concentrates Feed Pump

The shielding source terms for this equipment are the same as those for the waste crystallizer/evaporator described in e. above.

g. Entrainment Separator, Crystallizer Condenser (Shell Side), Crystallizer Distillate Tank, Crystallizer Distillate Pumps, and Crystallizer Subcooler (Shell Side)

This equipment is not expected to contain significant radioactivity; the fluids are expected to contain about 1 ppm of solids by weight. The shielding source terms for the equipment are about 8.3×10^{-6} of that for the waste concentrates tank described in a.

h. Spent Resin Dewatering Pump

This equipment is not expected to carry significant radioactivity. For conservatism, and consistency with the guidance given by NUREG-0017, the shielding source terms for this equipment are taken to be the same as those for the resin sluice pump described in Subsection 12.2.1.9.

i. Crystallizer Reflux Pot and Crystallizer Reflux Pump

This equipment is not expected to contain significant radioactivity. The shielding source terms for this equipment is taken to be the same as those for the crystallizer distillate tank described in g.

j. Extruder

In the extruder, spent resins or crystallizer bottoms are mixed with the asphalt binder in a one-to-one ratio by weight. The shielding source terms for this equipment are 0.5 times those for the spent resin centrifuge described in d.

12.2.2 Airborne Radioactive Material Sources

The major source of airborne contamination during normal operation is leakage of radioactive fluid from equipment and valves. Other sources from anticipated occurrences include opening of sealed equipment and evaporation during fuel handling. Accident sources such as DBA, fuel handling accident, and radwaste system failures are discussed in Chapter 15.

12.2.2.1 Design Basis

The ventilation system was designed to maintain the normal airborne radioactivity concentration to levels below the applicable occupational concentration values listed in 10 CFR 20 for air. On May 21, 1991, a complete revision to 10 CFR 20 was issued. Several design bases reference 10 CFR 20 and specific terms or parts of 10 CFR 20. Design bases information provides a historical perspective of the information used to formulate a particular design. References to 10 CFR 20 when used in a historical or design bases context have not been changed to reflect the revised 10 CFR 20.

Air flow is directed from areas of high occupancy and low airborne radioactive concentrations to areas of increasing contamination and lower occupancy requirements. Design basis maximum leakage rates to various buildings and compartments were chosen in accordance with NUREG-0017, as referenced by Regulatory Guide 1.112. Specific isotopic activity of the fluid leakage is based on 0.25 percent fuel clad defects, as discussed in Subsection 11.1.1.

12.2.2.2 Leakage Sources

a. Containment

The design basis maximum daily leakage to the Containment, as recommended in NUREG-0017, is 1 percent of the reactor coolant noble gas inventory and 0.001 percent of the iodine inventory. Reactor coolant isotopic concentrations for 0.25 percent failed fuel are listed in Table 11.1-1. The containment ventilation purge systems are designed to reduce airborne concentrations to acceptable levels within 20 hours after shutdown. Specific airborne concentrations in the Containment at shutdown and following purging are presented in Table 12.2-31 and Table 12.2-32.

b. Turbine Building

Calculating maximum airborne contamination in the Turbine Building assumes the leakage of 1700 pounds per hour of steam with the specific activities described in Table 11.1-4. Building volume, ventilation rates for summer and winter, and airborne isotopic concentrations are listed in Table 12.2-33 and Table 12.2-34.

c. Auxiliary Buildings

Several areas in the Primary Auxiliary Building have a concentration of piping and valves with a potential for significant contributions to airborne contamination. None of these areas requires continuous occupancy. The design basis maximum leakage rate to the Primary Auxiliary Building, per NUREG-0017, is 20 gpd. The source for these areas is reactor coolant with 0.25 percent fuel clad defects, as presented in Table 11.1-1. Airborne concentrations are presented in Table 12.2-35 for the average contaminated area.

12.2.2.3 Movement of Spent Fuel

The most significant contribution to airborne radioactive contamination during the movement of spent fuel is from tritium released by evaporation from the surface of the spent fuel pool and the refueling canal. The design concentration of tritium in the spent fuel pool and the refueling canal is 0.63 microcuries per milliliter. Tritium concentration in the water is controlled by release; in liquid form through the Boron Recovery System and in gaseous form via the ventilation system and the plant vent.

Airborne concentration is controlled by the ventilation system. Design air flow rates and tritium concentration are presented in Table 12.2-36 and Table 12.2-37. Evaporation rates of 270 lb/hr for the refueling canal (140°F) and 378 lb/hr for the spent fuel pool (175°F), which were used in the design calculations, assumed an air temperature of 50°F and a relative humidity of 20 percent.

12.2.2.4 Reactor Vessel Head Removal

The potential for release of airborne contamination due to reactor vessel head removal is significantly affected, and to a large degree controlled, by operating procedures and timing following shutdown. First, the reactor coolant is diluted with water from the reactor makeup water storage tank and/or the boric acid tanks via the Chemical and Volume Control System during cooldown. Then, there is a time lag between shutdown and vessel head lift (minimum 4 days) which allows decay. Letdown purification and degasification also continue during this time interval. Finally, the reactor coolant in the vessel is diluted by water from the refueling water storage tank by way of the Residual Heat Removal System. If necessary, portable fans can be used to mix the air in the vessel head area with the general containment air circulation to take advantage of the 31,000 cfm refueling purge flow rate.

12.2.2.5 Relief Valve Venting

Relief valves with significant potential for contribution to airborne contamination do not vent to building atmosphere. The relief valves on tanks and piping containing nondegasified fluids discharge to the primary drain tank. The primary drain tank, in turn, vents to the hydrogenated vent header for processing through the Radioactive Gas Waste System prior to environmental release.

12.2.2.6 Calculation Models and Parameters

Radionuclide input rates to building atmospheres were determined using assumed leakage rates of radioactive fluid to in-plant areas and corresponding specific activities and partition factors.

Calculations of the various airborne isotopic concentrations of interest were performed using the UE&C computer code HDOSE. The code determines isotopic concentrations from specific radionuclide input and removal processes, and allows simulation of specified, pre-determined time behavior for such processes.

In performing these calculations, credit was taken for the following removal mechanisms:

a. Natural radioactive decay

b. Recirculation - air is taken from building atmosphere, filtered, and returned to the building

c. Purge - outside air is drawn into the building and interior air is expelled to the outside.

(Recirculation and purge proceed with a mixing efficiency of 70 percent.)

Also taken into account was the formation of airborne radionuclides from:

a. The decay of airborne parent species

b. Trapped filterable parent species which decay to unfilterable daughters (e.g., iodine to xenon).

The rate equation used in HDOSE to describe the activity of isotope "j" in a compartment is given by:

$$\frac{dNc,j}{dt} = R_j + \sum_{k=1,2} B_{k,j}\,\lambda_k\,N_{c,k} + gj\sum_{k=1,2} B_{k,j}\,\lambda_k\,N_{r,k} - \left(\lambda_j + L_r\,e_j + L_e\right)N_{c,j}$$

where:

Nc, j	=	activity of isotope "j" in the compartment
R_j	=	release rate of isotope "j" into the compartment (Ci/hr)
$_j$	=	radiological decay constant of isotope "j"
$_k$	=	radiological decay constant of isotope "k"
$B_{k,j}$	=	branching ratio of decay from isotope "k" to isotope "j" (k=1 indicates parent of "j", k=2 indicates grandparent)
$N_{r,k}$	=	accumulation of nuclide "k" on recirculation filter
g_j	=	1 for gaseous isotope "j" 0 for nongaseous isotope "j"
e_j	=	recirculation cleanup filter efficiency for isotope "j"
L_e	=	compartment leak rate or purge rate
L_r	=	filtered recirculation flow rate.

12.3 RADIATION PROTECTION DESIGN FEATURES

12.3.1 Facility Design Features

The radiation protection philosophy for the design of Seabrook Station is to restrict radiation exposure to plant personnel and the general public to within the limits of 10 CFR 20 and 10 CFR 50, while ensuring high flexibility and availability within the power generation and safety objectives of plant operation.

This philosophy can be summarized in several basic design goals intended to minimize exposures:

a. To minimize, to the extent possible, the production of radioactive isotopes, e.g., the steps taken to reduce crud production described in Subsection 12.2.1.3.

b. To minimize the extent of areas housing radioactive equipment and piping through efficient arrangement of equipment and systems.

c. To shield the normally occupied areas from radiation.

d. To minimize exposures within high radiation areas by, first, controlling access to those areas, and secondly, through design of systems and equipment for reliability and ease of maintenance.

The plant is designed to permit periodic online equipment inspection and maintenance, radioactive material handling, decontamination and cleanup, and access to vital plant areas during normal plant operation, including anticipated operational occurrences. Postulated accidents are also considered in the determination of radiation exposure of Engineered Safety Features and other materials. In addition, these accidents are evaluated for access to, and habitability of, the control room, including ingress and egress, for the duration of the accident.

Offsite radiation exposures following postulated accidents are discussed in Chapter 15; those from processed radioactive material releases during normal operation are discussed in Chapter 11.

The primary objective of plant shielding is to provide for the protection and safety of all plant personnel and the general public under all normal and anticipated abnormal plant operating conditions. Reactor shielding, along with the radiation monitoring system and access control procedures, supplemented by periodic radiation surveys and radiochemical analysis, ensure that radiation exposures of the general public and plant personnel do not exceed the limits set by the federal regulatory agencies. The maximum allowable design dose rates for all plant areas, in conjunction with anticipated occupancy, limit the integrated whole body dose to less than 5 rem per calendar year. All areas that house radioactive materials are appropriately marked in accordance with Part 20 of Title 10 of the Code of Federal Regulations (10 CFR 20).

Plant operating personnel and the general public are protected by radiation shielding wherever a potential radiation source may exist. The shielding design philosophy embodies the following objectives:

a. To restrict the potential radiation dose to operating personnel during normal operation to within the limits of 10 CFR 20.

b. To adequately protect the operating personnel in the unlikely event of an accident, in order to allow termination of accident conditions and mitigation of the consequences without undue risk to the general public.

c. To protect equipment from excessive radiation exposure to prevent malfunctions due to radiation-induced failures.

d. To maintain the radiation exposure of the general public from normal operation to within the limits of 10 CFR 20.

The guidance provided by Regulatory Guide 8.8 has been utilized extensively in the plant radiation protection design philosophy, as described in the following pages of this section.

The radiation sources and associated input parameters, assumptions, and methodology described in Section 12.2 represent those used to establish original shielding design. The radiation sources utilized for the design of the primary shield were developed by the reactor vendor and are based on a standard four loop Pressurized Water Reactor. The radiation sources in the spent fuel and the N-16 source were also developed by the reactor vendor, and are based on a reactor power of 3565 MWt. The radiation sources in the primary coolant system and the auxiliary systems are based on a reactor power of 3654 MWt, a one-year fuel cycle, and 1 percent fuel element defects.

The impact on plant shielding requirements was evaluated for an analyzed reactor power of 3659 MWt and operation with an 18-month fuel cycle. This represents a minor change from the original design basis. The conservative analytical techniques used to establish the original shielding requirements, and the Station Technical Specification which will restrict the reactor coolant activity to levels significantly less than the 1 percent fuel defects, ensure that operation at the licensed core power level will have no significant impact on shielding requirements and safe plant operation.

12.3.1.1 <u>Radiation Zones and Access Control</u>

The shielding design bases for work areas are a combination of the design radiation level and anticipated occupancy times. The plant is divided into zones dependent upon the intensity of radiation within the given area. Areas within these zones are posted in accordance with the regulations of 10 CFR 20. Occupied areas within a zone are limited to the same radiation ranges as prescribed for that zone. Zone classifications are presented in Table 12.3-1 and Table 12.3-2.

Zone boundaries, decontamination facilities and location of radiation monitors are shown in Figure 12.3-1, Figure 12.3-2, Figure 12.3-3, Figure 12.3-4, Figure 12.3-5, Figure 12.3-6, Figure 12.3-7, and Figure 12.3-8, Figure 12.3-9, Figure 12.3-10, Figure 12.3-11, Figure 12.3-12, Figure 12.3-13, Figure 12.3-14, Figure 12.3-15, Figure 12.3-16, and Figure 12.3-17. The arrangement of the chemistry lab, health physics facilities and counting room is shown in Figure 12.3-17. Shield wall thicknesses for all major sources of radiation are given in Table 12.3-3, Table 12.3-4, Table 12.3-5, Table 12.3-6, Table 12.3-7, Table 12.3-8, Table 12.3-9, Table 12.3-10, Table 12.3-11, Table 12.3-12, and Table 12.3-13.

Access control points are also shown in Figure 12.3-1, Figure 12.3-2, Figure 12.3-3, Figure 12.3-4, Figure 12.3-5, Figure 12.3-6, Figure 12.3-7, Figure 12.3-8, Figure 12.3-9, Figure 12.3-10, Figure 12.3-11, Figure 12.3-12, Figure 12.3-13, Figure 12.3-14, Figure 12.3-15, Figure 12.3-16, and Figure 12.3-17, while discussion of access control and its implementation can be found in Section 12.5. The entire Containment Building is a controlled access area.

The turbine generator structure areas, administrative offices, turbine plant service areas and the control room are designated Zone I. Areas such as the local control space in the Primary Auxiliary Building, the waste disposal area, and the operating deck of the spent fuel storage area, are generally designated Zone II. Intermittently occupied work areas, such as valve galleries, are designated Zone III. Typical Zone IV areas include steam generator compartment areas (after reactor shutdown) and areas outside of containment pipe penetrations. A typical Zone V is the volume control tank area. Certain areas of the Containment are accessible for a limited time during normal plant operation.

The radiation counting room is designated Zone I (less than 0.5 mrem/hr); however, sufficient shielding is provided to assure that the background dose rate is low enough (less than 0.1 mrem/hr) to permit accurate operation of counting equipment.

Shielding has been designed by identifying source strengths within an area and then providing sufficient shielding to achieve the specified dose rate in adjacent zones. The source strengths are based on 1 percent failed fuel and maximum expected activation product levels, so that the actual radiation levels experienced within the station are expected to be less than the design values. Concrete shield thickness was in most cases determined by rounding from the calculated required thickness to the next higher 6-inch increment. In a few cases, where space was limited, the next 3-inch increment was used.

12.3.1.2 Handling of Nuclear Materials

Most systems included in the Primary Auxiliary and Waste Processing Buildings are used to process the radioactive byproducts produced in, and which leak from, the Reactor Coolant System during normal power operation. The design of the systems in the letdown purification and general waste processing systems reflects the concept of minimizing the exposure of plant personnel.

New fuel handling is discussed in Subsection 9.1.4.

Storage and handling of radioactive sources is discussed in Subsection 12.5.3.7.

12.3.1.3 Shielding and Layout Features

a. Plant Layout

Plant layout includes optimal location of radioactive components. The most radioactive systems are located toward the interior and on the lower plant levels, with less radioactive systems located toward the outside.

The plant layout provides for personnel access which maintains occupational doses as low as reasonably achievable (ALARA). Passage through a higher radiation area to obtain access to a lower radiation area is avoided, thus minimizing unnecessary accumulation of occupational exposure.

Wherever practical, components are shielded individually to keep exposure ALARA during maintenance periods. Shielded pipe chases are utilized extensively to segregate radioactive piping from normal occupied areas. Reach rods are used where required to place the operator in a low radiation area during valve operation. Auxiliary control boards are located with ALARA occupational exposure in mind.

When sources of sufficient strength are present, labyrinth entrances are utilized to minimize the contribution in the walkways. One or two scatter labyrinths are used as required by scattering calculations.

Periodic review of plant design and equipment arrangement aimed at maintaining occupational radiation exposures (ORE) ALARA resulted in a number of design changes, as illustrated in the examples below:

1. Access to the stairways in the RHR Vaults may be restricted below elevation 3'-2" due to equipment and pipe shine.
2. Shielding above the demineralizers in the Primary Auxiliary and Waste Processing Buildings was increased.
3. Primary sample heat exchanger and sink room was rearranged to reduce shine.
4. Evaporator equipment was rearranged to minimize radiation levels in adjacent walkways.
5. Areas of potentially excessive radiation levels were identified and space for possible future shielding was reserved.

b. Equipment Layout

The criteria for the arrangement of equipment containing radioactive sources were developed specifically for maintaining occupational doses ALARA and for ease of maintenance.

1. Filters (Liquid)

(a) Each potentially radioactive filter is located inside an individual shielded compartment. This minimizes the contribution to radiation levels from adjacent filters during maintenance periods. One exception to this are the vendor supplied waste liquid processing system filters. These filters may be in a common shielded compartment with demineralizer vessels, or outside a shielded area if the expected dose is Zone III or lower.

(b) For PAB and WPB filters, where changeouts are required, adequate space is provided for use of the remote filter handling cask in removing the filter (lateral room to swing the vessel head clear, and head room for lifting the cartridge), loading the filter into the cask, and transportation to the solid waste area. Attention is given to ensure that there are no interferences or obstructions in the path. The shield wall for waste liquid processing equipment is vendor supplied for the vendor system. This ensures interferences or obstructions in filter/resin manipulations are accounted for.

(c) All valves and instrumentation associated with filters are located outside the compartment. Normally, filter process valves are operated by remote manual mechanical linkage which extends to a low radiation zone, to minimize operator exposure during normal operation.

Where practical, filters are located near the solid waste area to minimize the chance of spillage in transit.

2. Demineralizers

The primary means of processing radioactive water to be discharged is through a vendor-supplied system. This system meets the intent of the design requirements in Regulatory Guide 1.143 and NUREG 0800. (see Section 1.8)

Other aspects of radioactivity processing systems are:

(a) Capability is provided to remotely remove all primary- side resins by flushing. All initially installed resin transport lines are designed to avoid resin traps. Butt welds instead of socket welds and five diameter bends instead of fittings are used in downward-sloping pipe without any sections horizontal or upward-sloping. Procedural controls are in place to prevent normal access during times of elevated dose rates.

(b) The resin fill line is shaped to prevent direct radiation streaming.

(c) Each potentially radioactive primary-side demineralizer in the Primary Auxiliary Building and Waste Processing Building is located inside shielded compartments or cubicles in order to reduce shielding problems during maintenance periods.

(d) All valves and instrumentation associated with these demineralizers are located outside the compartment. All demineralizer process valves, except for the vendor-supplied system, are operated by remote manual mechanical linkage which extends to a low radiation zone to minimize operator exposure during normal operation. The vendor-supplied system is designed to process water from floor drain tanks. The vendor-supplied components which concentrate radionuclides are all shielded to reduce any potentially elevated dose rates.

(e) Only the required process lines enter into, or pass through, the demineralizer cubicles, as access to the cubicles during normal operation will be strictly controlled.

(f) The demineralizers associated with the Steam Generator Blowdown System are not located in individual shielded cubicles, and are not provided with the capability for remote sluicing. These units normally treat secondary coolant with only minor or no contamination present. Processing blowdown through these demineralizers is based on maintaining the general area near the vessels as a radiation Zone II (<2 mr/hr). If significant primary-to-secondary leakage occurs, the primary method to process radioactive secondary liquid from the steam generators is to direct steam blowdown flash tank bottoms cooler discharge to the floor drain tanks. If no secondary pressure is available, the steam blowdown and wet lay-up pumps can be used. From the floor drain tanks, processing through the installed vendor system

(WL-SKD-135) to the waste test tanks is the preferred method (reference Subsection 11.2.2.1). In addition blowdown processing may be through the blowdown evaporators in place of the Blowdown Demineralizer System.

3. Adsorber Beds

(a) The first and second waste gas adsorber beds are located in individual shielded compartments.

(b) The third-through-fifth beds are located in a common compartment with the most active bed farthest from the entrance.

(c) All valves are located outside the bed cubicles, and can be operated from a low radiation area.

4. Recombiners

(a) Post-accident recombiners are located inside the Containment Building, and are designed for operation in the accident environment.

(b) Shielding is provided by the 4½-foot-thick containment walls.

5. Tanks

(a) Tank overflow lines are connected to prevent spillage on the floor and to prevent dissolved gases from escaping the tank. In limited cases the overflow lines are directed toward floor drains.

(b) Controlled ventilation is provided for tanks containing aerated or hydrogenated fluids.

(c) Tanks located outside of heated buildings are protected from freezing by steam heating panels.

(d) Manual valves are located in, or have handwheel extensions to, low radiation zones.

6. Evaporators

(a) Shielding is provided for individual evaporator units.

(b) Instruments, valves in service lines, and evaporator sample points are located in low radiation zones.

(c) Connections are provided for flushing and draining of pipe and equipment prior to maintenance.

7. Pumps

(a) In order to perform major maintenance in as low a radiation zone as reasonably achievable, the pump or motor can be decontaminated, if necessary, and moved to a low radiation area. Temporary local shielding may also be used.

(b) Pumps are designed with double mechanical seals to give a minimum leakage of radioactive fluid.

(c) Remote instrumentation and switching is provided, as required.

8. Steam Generators

The portions of the steam generators containing reactor coolant are shielded by the 4-foot-thick secondary shield walls.

9. Sampling Station

(a) The sample sink room is separated from the sample heat exchangers by a shield wall.

(b) Sample rates are limited to 1.5 gallons per minute by design.

(c) The sample hood is ventilated to prevent the accumulation of gases.

(d) The sample system is designed for a closed system line purge prior to sampling.

(e) Shielding is provided at local sampling points, as required.

10. Penetrations

(a) Where possible, penetrations through shield walls are offset from line-of-sight of the source.

(b) Where necessary, the annulus between a pipe and its sleeve is packed with lead wool or lead-silicone foam with a nominal density of 150 lb/ft^3.

11. Instrumentation

(a) Drains from instrument blowdowns are routed to radioactive drains.

(b) Diaphragm seals or clean water seal legs are used, wherever practical, to minimize the volume or radioactive fluids entering low radiation areas via instrument impulse lines.

(c) Radioactive gas and liquid samples are returned to process lines wherever practical.

(d) Wherever practical, instruments are located in low radiation zones to permit extended access for calibration and testing.

12. Piping and Valves

In order to minimize concentrated pockets of crud in permanently installed radioactive systems:

(a) Piping 2½" and larger is butt-welded.

(b) Spent resin sluicing lines utilize five diameter bends to minimize the number of fittings.

(c) Valves are selected to avoid crud pockets.

(d) Piping layout avoids pockets wherever possible.

12.3.2 Shielding

The material most commonly employed for shielding is concrete. Where space is limited, steel or lead is substituted for ordinary concrete in equivalent thicknesses. Whenever cast-in-place concrete is replaced by concrete blocks (removable or fixed), the design assures protection on an equivalent shielding basis.

Analytical models were selected according to the source geometry under consideration. Tanks, vessels, and large pipe-sections containing radioactive materials were shielded by considering uniform cylindrical volumetric sources. Appropriate line source approximations were used for small pipes and tall vessels with small diameters. For a conservative estimate of dose rate, sources of irregular geometry were modeled by a point source of strength equivalent to the volumetric source.

The following techniques, codes, models, and assumptions were used:

- Point kernel integration methods were used.
- Buildup factors were accounted for inside the integrals.
- Self-shielding was taken into consideration.
- Concrete density was assumed to be 2.35 gm/cc.
- The maximum calculated shield thickness was specified for each component or radiation area.
- Source term data corresponds to 1 percent failed fuel with a power level of 3654 MWt.

Special protective design features to ensure that occupational radiation exposures will be ALARA are described in Subsection 12.3.1.3. The guidance given in Regulatory Guide 8.8 can be seen in these features.

Compliance with Regulatory Guide 1.69 is addressed in Section 1.8.

12.3.2.1 Shield Configurations

a. Reactor Shielding

1. Primary Shield

The primary shield is a large mass of reinforced concrete, 7½ feet thick at core midplane, that surrounds the reactor vessel and extends upward from the containment floor to form the walls of the refueling cavity. The primary shield is designed to:

(a) Reduce, in conjunction with the secondary shield, the radiation level from sources within the reactor vessel and Reactor Coolant System, and allow limited access to the Containment during normal operation.

(b) Limit the radiation level after shutdown from sources within the vessel, and permit limited access to the areas containing reactor coolant system equipment.

(c) Limit neutron flux activation of component and structural materials over the life of the plant.

2. Secondary Shield

The secondary shield is a reinforced-concrete structure that surrounds the reactor coolant equipment, pipes, pumps and steam generators. This shield protects personnel from gamma radiation emanating from reactor coolant activation products and fission products that are transported from the core by the reactor coolant. The neutrons emitted by the decay of carried-over N-17 are effectively eliminated by the concrete shielding employed for N-16. The secondary shield also supplements the primary shield function of attenuating direct core radiation. In addition, it permits limited access to the Containment during normal operation so that inspection of essential equipment may be accomplished without requiring plant shutdown.

3. Neutron Shield

In order to reduce dose rates and equipment activation on the containment operating floor during power operation, a supplementary shield has been designed to minimize the streaming of neutrons from the reactor cavity. A neutron shield consisting of Reactor Experiments Type 277 borated concrete, and which is integral to the permanent reactor cavity seal ring, is installed around the reactor vessel refueling flange. The neutron shield is suspended from the permanent seal ring and fills the annular area between reactor vessel refueling flange and the cavity wall. The neutron shield is approximately fourteen inches thick and supported from the bottom by a one-inch thick steel plate. Sectional and plan views of the permanent reactor cavity seal ring/neutron shield are shown in Figure 6.2-26 and Figure 6.2-28, respectively.

b. Containment Shielding

The containment shielding is a steel-lined, reinforced-concrete containment structure that completely surrounds the reactor building equipment. At full power operation, this shield attenuates the radiation level outside the primary-secondary shield complex, including radiation sources which become airborne during normal operation due to primary system leakage, to ensure that radiation levels outside the Containment are less that 0.5 mrem/hr. The containment structure also shields against radiation sources inside the Containment due to fission products released following postulated accidents. The integrated direct dose is less than 260 mrem immediately outside the Containment over a period of two hours after the design basis accident (DBA), which will permit access to such vital areas as the control room. The containment wall and dome are 4½-feet and 3½-feet thick, respectively.

c. Spent Fuel Shielding

This shielding provides protection during all phases of spent fuel removal and storage. Operations that require shielding of personnel are spent fuel removal from the reactor, spent fuel transfer through the refueling canal and transfer tube, spent fuel storage, and spent fuel shipping cask loading prior to transportation. All spent fuel removal and transfer operations are performed under borated water to provide radiation protection.

All accessible areas around the tube and canal are shielded. All shields were designed for a contact radiation dose rate of less than 100 mr/hr. Four inches of lead plate were added between the liner and concrete at the bottom of the canal. In the Enclosure Building a shield box was designed around the tube. This box consists of approximately 300 bricks weighing 50 pounds each. These bricks will be explicitly marked with a sign stating that potentially lethal radiation fields are possible if the bricks are removed during fuel transfer. The access point noted in Figure 1.2-3 is an inspection hatch (manway). This hatch is shielded with a three-foot concrete plug. This plug shall also be marked as noted above.

Minimum allowable water depth above a fuel assembly during fuel handling is 10 feet in the reactor cavity. This limits the dose at the water surface to less than 10.0 mrem/hr for an assembly in a vertical position. The minimum water depth in the spent fuel pool is 13 feet above the top of the fuel assemblies in the storage racks. For this depth, the dose rate at the water surface is less than 2.5 mrem/hr. Normal water depth above the stored assemblies is about 25 feet.

The 5-foot thick concrete walls of the fuel transfer canal and the 6-foot thick spent fuel pool walls supplement the water shielding, and limit the radiation dose levels in most working areas to less than 2.5 mrem/hr, and a maximum dose in some areas less than 100 mrem/hr.

The refueling water and concrete walls also shield personnel from activated control rod clusters and reactor internals that are removed at refueling times. Dose rates are generally less than 2.5 mrem/hr in working areas. However, certain manipulations of fuel assemblies, control rod clusters, or reactor internals may produce short-term dose rate levels in excess of 2.5 mrem/hr. Radiation levels in the working areas will be closely monitored during refueling operations to ensure that exposures for plant personnel do not exceed the integrated doses specified in 10 CFR 20.

d. <u>Control Room Shielding</u>

The control room shielding is designed in accordance with applicable regulations, to permit continuous occupancy by control room personnel following a DBA. This enables control room operators to maintain full control and to shut down the plant without personal hazard. The control room shielding is 2-feet thick, based upon an integrated dose during the 30 days following the DBA which does not exceed 5 rems whole body, or its equivalent to any part of the body, as required by General Design Criterion 19 of 10 CFR 50, Appendix A. A layout drawing of the control room is shown in Figure 1.2-32.

e. Plant Auxiliary Systems Shielding

Auxiliary shielding includes all concrete walls, covers and removable blocks that shield the numerous radiation sources in the radioactive waste disposal, makeup and purification, and chemical addition and sampling systems. Typical components that require shielding include the volume control tank, thermal regeneration demineralizers, waste drumming area and reactor coolant system drain tank. Shield wall thicknesses for components in auxiliary systems are given in Table 12.3-3, Table 12.3-4, Table 12.3-5, Table 12.3-6, Table 12.3-7, Table 12.3-8, Table 12.3-9, Table 12.3-10, Table 12.3-11, Table 12.3-12, and Table 12.3-13.

f. Turbine Shielding

The radioactive material inventory in the Turbine Building is very small since only secondary steam enters the area with small amounts of primary coolant leakage. Shielding is not, therefore, a major concern here, with wall and floor thicknesses determined from structural considerations.

g. General Plant Yard Areas

All shielding is designed so that the dose rates in plant yard areas which are frequently occupied by plant personnel remain below 0.5 mr/hr. These areas are surrounded by a security fence, and are closed off from areas accessible to the public for general safety.

12.3.2.2 Plant Shielding to Provide Access to Vital Locations for Post-Accident Operations

Following an accident, significant radioactivity may be released from the reactor core, presenting unusual hazards to operating personnel. A review was conducted to assess the projected amount of activity released, systems involved in transport of this activity, effect of the transported activity on plant dose rates and acceptability of dose rates in locations requiring access for necessary operations (vital locations).

This assessment employed core fission product release source terms consistent with NUREG-0737, Section II.B.2 (100% noble gas, 50% halogen, 1% other fission product). The assessment addressed both pressurized and depressurized accidents, and projected consequences of the release at post-accident times ranging from the onset of cold leg recirculation to 1 year. The assessment was based on an analyzed power level of 3565 MWt and a 1-year fuel cycle length.

The systems considered in this assessment included containment spray, chemical and volume control, safety injection, residual heat removal and combustible gas control.

Using the source term and system transport information described above, dose rates in various plant areas where projected. These projections considered shine, scatter and radiation streaming, including effectiveness of facility shielding. Levels in areas which must be accessed for operational tasks (vital locations) were tabulated, along with occupancy times, to verify projected exposures are within applicable limits. Such locations included the control room, technical support center, post-accident sample station, chemistry laboratory, switch gear room, radwaste control station, radiation controlled area tunnels and hydrogen analyzer area. High dose-rate areas are graphically depicted on area zone maps of the plant. These maps will aid in projecting exposures for potential post-accident operations not explicitly identified in the vital location table. Results of these projections demonstrate projected exposures in vital locations are within the GDC-19 and NUREG-0737 (Item II.B.2) criteria.

The assessments described above were incorporated into the Post-Accident Dose Engineering Manual, which is used in planning for post-accident operations. Rationale for not including several areas noted in NUREG-0737 (Item II-B.2) is delineated in this manual. A copy of the manual was provided to the NRC. The information in this document will be factored into the overall post-accident response actions.

The impact of an analyzed core power level of 3659 MWt and operation with an 18-month fuel cycle was evaluated, and it was determined that the post-accident operator exposure will continue to remain within regulatory limits. The information from the Post-Accident Dose Engineering Manual, modified to reflect the licensed core power level, is factored into the overall post-accident response actions.

12.3.3 Ventilation

The station ventilation system has been designed to provide a maximum of safety and convenience for operating personnel, construction workers and site visitors working both within the station radiologically controlled area and in station buildings outside the radiologically controlled area during normal operating and anticipated operational occurrences. The potential exposure to onsite personnel and to members of the general public resulting from airborne radionuclides from station operation complies with 10 CFR Part 20 and 10 CFR Part 50, respectively.

12.3.3.1 Ventilation Design Bases

Descriptions of the ventilation systems for each building which can be expected to contain radioactive materials, including design bases, are contained in Section 9.4. Diagrams associated with the descriptions show equipment, air flow patterns, and expected flow rates for normal and emergency conditions.

A description of the ventilation systems for the control room complex is contained in Subsection 9.4.1, Figure 9.4-1, Figure 9.4-2, and Figure 9.4-3, and shows equipment, air flow patterns, and expected flow rates. Section 6.4 discusses the habitability and life support systems of the control room complex with respect to NRC General Design Criterion 19.

In each case, air flow has been directed from areas of low potential airborne radioactivity to areas of higher airborne radioactivity by exhausting from the areas of higher radioactivity. The ventilation rate for the areas of higher radioactivity was determined both from the ventilation rate required to remove equipment heat, piping and electrical losses and from the ventilation rate required to control the concentration of airborne radioactivity. The ventilation was designed to meet the exposure limits for airborne concentrations listed in 10 CFR 20. Maintenance of a negative pressure by the exhaust systems in the areas of higher radioactivity induces an air flow from corridors and operating areas preventing the exfiltration of airborne radioactivity to clean areas normally occupied by operating or maintenance personnel.

Testing and maintenance will be performed in accordance with the criteria presented in Regulatory Guide 1.52 for the safety-related filter systems.

Failure to meet these in-place testing criteria will necessitate the change-out of filters or adsorbers.

12.3.3.2 <u>Provisions for Localized Ventilation</u>

Provisions for localized ventilation during maintenance and refueling operations are provided to the extent practicable to reduce concentrations of airborne radioactivity in areas not normally occupied where maintenance of in-service inspection has to be performed.

12.3.3.3 <u>Exhaust Filtration</u>

The exhaust from each area which can be expected to contain significant airborne radioactivity is processed through HEPA or HEPA and carbon air cleaning systems before being discharged to the unit plant vent.

Air cleaning units are provided for the containment enclosure emergency exhaust, the fuel storage building emergency exhaust, the containment purge exhaust, the primary auxiliary building exhaust for areas with a potential for significant airborne radioactivity, and the waste processing building exhaust for areas with a potential for higher airborne radioactivity.

In addition, recirculation air cleaning units are provided for the main control rooms and the containment structures.

Descriptions of the air cleaning systems, including design bases for the containment enclosure emergency exhaust and the fuel storage building emergency systems, including the design bases for the remainder of the buildings, are contained in Section 9.4.

Compliance with Regulatory Guide 1.52, Revision 2, of the containment enclosure emergency air cleaning unit, the fuel storage building air cleaning unit, and the control room emergency filtration subsystem is detailed in Table 6.5-1, Table 6.5-2 and Table 6.5-3 respectively.

The containment purge exhaust air cleaning unit, the primary auxiliary building normal exhaust air cleaning unit and the waste processing building air cleaning unit are equipped with quick-release filter clamps to minimize exposure of maintenance personnel when changing prefilters, medium efficiency filters and HEPA filters. The remainder of the air cleaning units employ standard threaded filter clamping devices because they are not expected to be exposed to more than minimal quantities of radioactive particulates.

The containment purge exhaust air cleaning unit, the primary auxiliary building normal exhaust air cleaning unit, the fuel storage building emergency exhaust air cleaning units and the containment enclosure emergency exhaust air cleaning unit are provided with bulk fill adsorber beds and guard beds, where applicable. The carbon for the adsorber and guard beds is pneumatically removed and filled, which minimizes exposure of the maintenance personnel to contaminated carbon. The control room emergency recirculation air cleaning unit and the containment recirculation air cleaning unit employ tray-type carbon adsorbers. The waste processing building air cleaning unit has no adsorber bed.

Access from both sides of the air cleaning units for maintenance and changing of filters is provided for the containment purge exhaust air cleaning unit, the fuel storage building emergency exhaust air cleaning unit and the primary auxiliary building normal exhaust air cleaning unit. Aisle space and clear means of ingress and egress are provided for the handling of filters and carbon bed carbon removal/fill equipment.

A layout of the primary auxiliary building normal exhaust unit provided in Figure 12.3-18 is an example of the filter bank spacing and access for maintenance.

The radiation control area (RCA) of the Administration and Service Building is provided with a once-through ventilation system. The exhaust system maintains a negative pressure on the entire RCA portion of the building, preventing the exfiltration of airborne radioactivity to the clean areas. The exhaust air is processed through 55 percent medium efficiency filters and then through HEPA filters. There is no adsorber bed provided for this system. The air is directed within the RCA from areas of low potential airborne radioactivity to areas of higher potential airborne radioactivity.

12.3.4 Area Radiation and Airborne Radioactivity Monitoring Instrumentation

12.3.4.1 Area Radiation Monitoring Instrumentation

a. Objectives and Design Basis

1. Detectors are located in areas that may be normally occupied without restricted access and which may have a potential for radiation fields in excess of the radiation zones described in Subsection 12.3.1.

2. The detectors provide on-scale readings of dose rate that include the design maximum dose rate of the radiation zone in which they are located, as well as the maximum dose rate for anticipated operational occurrences.

3. Each monitor has local visual and audible alarms, with variable setpoint.
4. Indication and annunciation are available in the main control room.
5. The design objectives and location criteria are in conformance with 10 CFR Part 20, Part 70 and Part 50, Appendix A, General Design Criteria 63 and 64, and Regulatory Guides 1.21, 8.2 and 8.8.
6. Post-accident monitoring instrumentation is provided as discussed in Section 7.5.

b. System Description

The digital computer-based Radiation Data Management System (RDMS) consists of local microprocessors for each channel, interconnected by a redundant communication loop to a redundant host computer system. Either of the two computers can by itself provide the total computing capacity required for satisfactory operation of the RDMS. The host computer system, in turn, is connected to an operator display/control console in the control room, the health physics control point, the RDMS computer room, the Main Plant Computer System (MPCS) computer room and the hot chemistry lab. The area radiation monitoring system instrument engineering diagram, Figure 12.3-19, shows an overview of the system, its components and location.

Table 12.3-14 lists the various area radiation monitoring channels provided and their pertinent design information, such as detector type, range, background radiation, safety class, alarm setpoints, referenced drawings for location of area radiation detectors, etc.

Class 1E area radiation equipment is supplied from Class 1E uninterruptible power supplies (UPS).

Except for the post-LOCA containment monitors and other high-range monitors, each channel is equipped with a radioactive check source which can be actuated from the main control room during test. The post-LOCA monitors and other high-range monitors use an electronic signal to test the circuit.

A typical channel is shown in Figure 12.3-19.

High radiation levels during refueling at the manipulator crane area in the containment structure initiates isolation of the containment purge and vent system.

Those detectors which are designated as non-Class 1E, and are located inside the containment structure are not designed to operate following a major LOCA, and are assumed to be not available to monitor post-LOCA conditions inside Containment.

Refer to Subsection 11.5.2 for a discussion of the local microprocessor provisions and operating details.

1. Area Monitor Detectors

   The area monitors employ Geiger-Mueller and ion chamber gamma detectors, as indicated on Table 12.3-14.

2. Class 1E Requirements

   Separate redundant cabinets are provided in the control room for control, recording and remote indication for those monitors in Table 12.3-14 designated as Class 1E. These cabinets and Class 1E area monitors are powered from their respective Class 1E inverters. Class 1E monitors supply their data to the RDMS host computer through an IEEE 279 acceptable isolation device. No information or alarm setting is permitted between the RDMS host computer and the Class 1E equipment. All setpoint changes and check-source insertions are performed locally or from hard-wired modules in the control room.

3. In-Containment High Range Monitoring

   Redundant Class 1E monitors are provided to monitor containment conditions under accident situations. The detector range is 10^{0}-10^{8} R/hr. The electronics cabinet is located outside Containment in the electrical tunnels. Indication is provided on the RDMS video displays and the RDMS racks in the main control room. These monitors will be designed, located, calibrated and qualified in accordance with Table II.F.1-3 of NUREG-0737.

   The detectors are located on the steam generator biological shield wall (the "A" detector is near steam generator "D" and the "B" detector is near steam generator "B") at an approximate elevation of +31'. These locations were selected to provide the detectors as large a view of Containment as possible, consistent with affording ease of access for maintenance and calibration.

4. Area Monitor Channel Description

(a) Containment Manipulator Crane Area Monitor-Channels 6535 A and B

Redundant Class 1E detectors are located on the manipulator crane. In the event of a fuel handling accident, these monitors in conjunction with safeguards actuation signals isolate the containment online and offline purge isolation valves, trip containment air pre-entry, refueling supply and containment online purge fans. Indication and alarm are provided locally and in the main control room.

(b) Personnel Hatch (Post-LOCA) - Channels 6536-1,2

This area monitor is located external to the Containment and is aligned with the personnel hatch. This radiation monitor is intended to monitor ambient radiation conditions following a LOCA.

(c) Containment Post-LOCA - Channels 6576A, B

These detectors are intended to monitor conditions inside Containment for Post-LOCA and are Class 1E.

(d) Volume Control Tank - Channel 6540

The detector is located inside the volume control tank area.

(e) High Range Area Monitors - Channels 6508-1,2, 6563-1,2, 6517-1,2 and 6518.

There are seven high range ion chamber detectors. In compliance with Regulatory Guide 1.97, these detectors have an upper range of 10^4 R/hr. These monitors are located in areas which may require entry after an accident or which contain recirculating post-accident fluids. These monitors are:

1) PAB - High Range Area Monitor - Channels 6508-1,2 and 6563-1,2

2) RHR - High Range Area Monitor - Channel 6517-1,2

3) FSB - High Range Area Monitor - Channel 6518

(f) Other Area Radiation Monitors - Channels 6534, 6537, 6538, 6539, 6541, 6543, 6544, 6545, 6546, 6547, 6549, 6550, 6551, 6552, 6553, 6554, 6555, 6556, 6557, 6558, 6559, 6570, and 6571

These channels use Geiger-Mueller detectors and monitor the ambient radiation at various points throughout the facility as listed in Table 12.3-14.

5. Calibration and Maintenance

Refer to Subsection 11.5.2.6 for calibration and maintenance details.

12.3.4.2 Airborne Radioactivity Monitoring Instrumentation

a. Objectives and Design Basis

The ventilation airborne Radioactivity Monitoring System provides radiation measurements, indications, records, alarms and controls at selected locations to detect and control radiation levels within Containment, Service Building, Radwaste Building and the plant vent, and to verify compliance with applicable limits of 10 CFR 20 and 10 CFR 50, General Design Criteria 19, 63 and 64.

On May 21, 1991, a complete revision to 10 CFR 20 was issued. Several design bases reference 10 CFR 20 and specific terms or parts of 10 CFR 20. Design bases information provides a historical perspective of the information used to formulate a particular design. References to 10 CFR 20 when used in a historical or design bases context have not been changed to reflect the revised 10 CFR 20.

Monitored points within the station ventilation system are in areas where potential personnel exposure to radiation is most likely and in several ventilation exhaust ducts. A tabulation of the airborne Radiation Monitoring System is found in Table 12.3-15, Table 12.3-16 and Figure 12.3-20.

Those monitors which are Class 1E are listed in the above tables and further discussed in Subsection 12.3.4.1b.2.

The sensitivity of the airborne radioactivity monitors is such that they should be capable of detecting ten MPC-hours of particulate and gaseous radioactivity in those plant areas that have contained sources of airborne radioactivity and which may be occupied by personnel. Typical airborne concentrations for various plant areas are given in Table 12.2-31, Table 12.2-32, Table 12.2-33, Table 12.2-34, Table 12.2-35, Table 12.2-36, and Table 12.2-37.

As discussed in Subsection 12.5.3.1, the Health Physics Program includes requirements to perform sampling and analysis for airborne radioactivity, routinely and during specific evolutions such as opening of the primary system. Sampling equipment includes portable continuous air monitors and portable samplers. The monitoring and sampling capabilities, when combined, provide sufficient information to permit adequate protection of personnel from exposure to airborne radioactivity.

b. System Description

Subsection 12.3.4.1b describes the digital computer based RDMS. Subsection 11.5.2 describes the local microprocessor provisions. The airborne Radioactivity Monitoring System consists of two basic types of monitoring systems:

- Particulate and gaseous monitors (with iodine sampling) which are skid-mounted and utilize pumping systems.
- Gross activity monitors which consist of detectors mounted directly in duct air stream.

A typical channel is shown in Figure 12.3-20.

1. Particulate and Gaseous Monitors

Each airborne particulate and gaseous monitor has common equipment as follows:

(a) Isokinetic Sampler

Sampler and lines adhere to requirements of ANSI N13.1. Sample line sizes are one-half inch with flow rate designed for 2-3 scfm. All sample lines slope from high point (isokinetic sampler) to the low point (sample pump).

(b) Pumping System

(1) The flow control assembly includes a pump unit and selector valves that provide a representative sample (or a "fresh" sample) to the detector.

(2) The pump unit consists of:

a A pump to obtain the air sample

b A flowmeter to indicate the flow rate

c A flow control valve to provide flow adjustment

d A flow alarm assembly to provide low and high flow alarm signals.

(3) Selector valves are used to direct the desired sample to the detector for monitoring and to block flow when the channel is in the maintenance or "purging" condition.

(4) Purging is accomplished with a valve control arrangement whereby the normal sample flow is blocked and the detector purged with a "fresh" sample.

(5) A sample flow rate indicator is calibrated linearly from 0 to 4 scfm.

(6) Indicator lights are actuated by the following:

a Flow alarm assembly (low or high flow),

b The filter paper sensor (paper drive malfunction), or

c The pump power control switch (pump motor on).

(c) Detectors

The particulate channel air sample is drawn in a closed system monitored by a scintillation counter-filter paper detector assembly. The filter paper collects 99 percent of all particulate matter greater than 0.3 micron in size on its continuously moving surface, and is viewed by a photomultiplier-scintillation crystal combination.

The air sample is returned after it passes through the series-connected iodine filter and gas monitors.

The detector is a hermetically sealed scintillator crystal combination. The pulse signal is transmitted to the radiation monitoring system local cabinet.

Lead shielding reduces the background radiation level to prevent interference with the sensitivity of the detector. The filter paper mechanism, and electro-mechanical assembly which controls the filter paper movement, is provided as an integral part of the detector unit. The unit contains an approximate 25-day filter paper supply at normal speed.

The particulate filter is followed by a cartridge-type charcoal iodine filter. The iodine filter is a charcoal filter which will meet certain specifications for iodine removal. The particulate and iodine filters are suitable for laboratory analysis. The removal of all particles greater than 0.3 microns eliminates the need for maintaining isokinetic conditions downstream of the particulate and iodine filters.

The iodine filter is not monitored continuously. The iodine filter is periodically removed and analyzed as appropriate.

Next, the gaseous channel views the air sample. The sample is constantly mixed in the fixed, shielded volume, where it is viewed by a beta scintillator. The sample is then returned to its environment.

The detector assembly is in a completely enclosed housing containing a beta scintillator mounted in a constant gas volume container. Lead shielding reduces the background radiation level to prevent interference with the detector's sensitivity.

2. Particulate and Gaseous Monitor Channel Descriptions

(a) Containment Monitor - Channel 6526 and 6548

Monitor 6526 draws a containment air sample through redundant pumps from the containment atmosphere. The sample is then returned to the containment atmosphere.

Monitor 6548 is located inside Containment at zero foot elevation and acts as a backup to monitor 6526. Monitor 6548 draws air sample from the Containment via the sample pump and then discharges back to the Containment.

These monitors are classified seismic Category I.

Indication and alarm is available locally and in the main control room.

See Subsections 5.2.5.3b.2, 5.2.5.5b and 5.2.5.5c for a further discussion of monitoring requirements.

(b) Waste Process Building Monitor - Channel 6531

The major potential release of airborne radioactivity in the Waste Processing Building is that associated with the Gaseous Waste Processing System. The gas dryers, carbon delay beds and the two gas compressors are situated in their individual compartments, and these compartments are ventilated in such a way that they are at a negative pressure with respect to surrounding areas. The ducted ventilation exhaust is continuously sampled and monitored. The sample is returned to the ducted ventilation exhaust line which is directed to the plant vent. Both the sampling point and the return are downstream of the filters in the ventilation exhaust. Information from this channel is displayed and alarmed on the radiation monitoring system panel in the main control room and locally.

(c) Primary Auxiliary Building Monitor - Channel 6532

Three minimum ventilation areas have been defined for the Primary Auxiliary Building:

(1) Heat exchanger, thermal regeneration demineralizer, and mixed bed demineralizer area

(2) Volume control tank area

(3) Charging pump area.

These areas, which are potential sources of airborne activity, are maintained at a negative pressure with respect to surrounding areas. The PAB ventilation system collects potentially contaminated air through a duct system and discharges it to the plant vent via filter train F-16. The sample withdrawal point for this monitor (RM-6532) is downstream of filter train F-16. The location of this sample withdrawal point provides an early warning to the operating personnel in the event that radioactive material becomes airborne in the PAB.

Indication and alarm is available locally and in the main control room. An alarm indication on these monitors would trigger a radiological evaluation within the areas served by these monitored ventilation lines. The evaluation would be performed by station HP personnel using portable survey and/or air sampling equipment, as necessary, to locate the source of the elevated ventilation line indication.

(d) Plant Vent Monitors - Channels 6528-1, 6528-2, 6528-3, and 6495

These detectors monitor the air exhausted by the Primary Auxiliary Building, Waste Process Building, Fuel Storage Building, containment structure and containment enclosure via the plant vent. An isokinetic probe, supplemented with an integral pumping system is used to withdraw an air sample from the plant vent. The air quantities exhausted via the plant vent are indicated on Figure 9.4-5, Figure 9.4-6, Figure 9.4-7, Figure 9.4-8, Figure 9.4-9, and Figure 9.4-10.

Multisensor flow transmitters and microprocessor provide a signal to the radiation monitor (RM-6528) to permit this monitor to calculate the microcuries per cubic centimeter flowing in the duct, microcuries per second and the integrated microcuries released through the plant vent.

The air collected by the isokinetic probe passes through the wide-range gas monitor, WRGM, (RE-6528-1, RE-6528-2, and RE-6528-3). Sampling provisions are located downstream of the isokinetic nozzle. The air flow enters the sample conditioning skid of monitor RM-6528 at a flow rate of ≈0.06 cfm during postulated accident conditions. This skid is intended to provide representative particulate and radioiodine samples for laboratory analysis (for normal operation as well as accident conditions) and to prevent contamination of the gas monitors. A multiple filter arrangement is provided to allow sampling capabilities for the duration of the measurement period. Each filter is equipped with a 4π solid lead shielding and quick disconnect fittings to minimize personnel exposures. In addition, all functional control is performed remotely.

The wide-range gas monitor has the capability of detecting a wide range of radiogas concentrations over 12 decades. The monitor meets the requirements of NUREG-0737, item II.F.1 by providing an upper range of 10^5 μCi/cc for noble gases and the capability to collect post-accident plant vent grab samples. A seven-day composite grab sampling system for normal operation is also provided for particulates and iodine determination. Indication and alarm from the WRGM are available locally and in the main control room.

A portable pump with detector (RE-6495) provides backup to the WRGM mid and high range. Indication and recording are provided in the Technical Support Center (TSC).

3. Gross Activity Monitors

These units do not utilize a pumping system. The detectors are located directly in the duct air stream. These monitors employ Geiger-Mueller type detectors. The local microprocessor cabinet provisions are described in Subsection 12.3.4.1b.

4. Gross Activity Monitor Channel Descriptions

(a) Administration and Service Building Fume Hoods Monitor -Channels 6523, 6524 and 6525

These detectors are Geiger-Mueller counters and are located in each of the chemistry fume hoods. They measure the gross activity of air exhausted from the shop fume hoods to atmosphere. Local alarm and indication are available near the fume hoods. No control function is provided.

These monitors provide data to the RDMS host computer for alarm, display and documentation. Remote indication and alarm are available in the main control room.

(b) Fuel Storage Building Exhaust Monitor - Channel 6562

This detector is a Geiger-Mueller counter and is located in the fuel storage building ventilation exhaust duct downstream of the fans. This detector measures the gross activity vented from the Fuel Storage Building to the plant vent. Indication and alarm is available locally in the Fuel Storage Building near the spent fuel storage pool, and remotely in the main control room.

(c) Containment Enclosure Emergency Exhaust Monitor – Channel 6566

This detector is a Geiger-Mueller counter and is located downstream of the containment enclosure emergency exhaust filter fans and measures the gross activity exhausted to the plant vent stacks. Indication and alarm are available locally near the filter fans and remotely in the main control room.

(d) Primary Auxiliary Building, Miscellaneous Ventilation Exhaust – Channel 6567

This detector is a Geiger-Mueller counter and is located at the inlet to the primary auxiliary building cleanup filter. The following areas are monitored by this detector: valve aisle, volume control tank area, sample heat exchanger room, sample room fume hood, degasifier area, PAB lower level elevation (-)6', and PAB filter and heat exchanger area.

Indication and alarm are available locally near the cleanup filter, and remotely in the main control room.

(e) Containment Enclosure Monitor - Channel 6568

This detector is a Geiger-Mueller counter and is located in the exhaust duct from the containment enclosure at the inlet to the cleanup filter. The detector monitors the gross activity exhausted from the containment enclosure. Indication and alarm are available locally near the cleanup filter, and remotely in the control room.

(f) Control Room Air Intake Monitors - Channels 6506A and B, 6507A and B

Four detectors are located in the east air intake piping and four detectors are located in the west air intake piping. These detectors are located in the Control and Diesel Building. These GM detectors, which are Class 1E, monitor the control room air intake and automatically shut down, on a high radiation signal, the control room ventilation fans and isolation dampers. Each monitor utilizes a two-out-of-two detector logic such that two detectors must be in alarm before the monitor initiates an isolation signal. These detectors are directly mounted in the air intake stream and do not require shielding.

Indication and alarm are provided locally. Indication, recording and alarm are provided in the main control room.

(g) Containment Online Purge Monitor - Channels 6527A, 6527B

For a description of this monitor see Subsection 11.5.2.1.

5. Portable Continuous Air Monitors (CAM)

   Four portable continuous air monitors are available. The CAMs are equipped to monitor particulate and noble gas.

   The normal locations for the CAMs are as follows:

   - Waste Process Building
   - Primary Auxiliary Building
   - Fuel Storage Building
   - Containment (on the operating floor during refueling outages)
   - Control Building (during normal operations)

   CAMs may be moved to other station locations as radiological conditions dictate.

6. Calibration and Maintenance

   Refer to Subsection 11.5.2.6 for calibration and maintenance details.

12.3.4.3 Post-Accident In-Plant Iodine Assessment

The capability exists for the determination of airborne radioiodine levels in-plant under accident conditions. This capability includes the use of air samplers with radioiodine-specific sample cartridges and the use of gamma spectroscopy instrumentation for sample analysis. Information on portable air sampling and counting room equipment is discussed in Subsection 12.5.2.

This sampling and analysis is described in station procedures in which station personnel are trained. Training includes the proper handling and preparation of high-level radioactive samples and the operation and calibration of gamma ray spectroscopy equipment for post-accident sampling in addition to normal sampling techniques.

12.4 DOSE ASSESSMENT

12.4.1 Criteria and Objectives

Seabrook Station and its operational procedures are designed to ensure that radiation exposures to the station's operating and maintenance personnel during normal operations and anticipated operational occurrences are within the limits specified in 10 CFR 20 and as low as reasonably achievable (ALARA).

Regulatory Guide 8.19 was not available until after the plant was already past the early stages of construction; however, in all phases of the design and construction of the plant, careful consideration has been given to eliminating unnecessary exposures and, wherever practical, minimizing the exposures of station personnel.

The radiation exposures to station personnel presented in Subsection 12.4.2 are pre-operation estimates based on PWR operating experience in the 1970s and Seabrook Station's design features. The occupational radiation exposures in Light Water Reactors have since been reduced significantly. The estimated man-rems presented in Tables 12.4-1 through 12.4-7 are largely independent of the fine variation of the reactor power. For an operating nuclear power station, the actual occupational exposures are carefully recorded and reported. Following operation at the licensed core power level, the occupational exposures are expected to increase approximately in proportion to the reactor power.

The estimated annual direct and scattered doses at the Site Boundary presented in Subsection 12.4.3 were calculated conservatively and remain valid following operation at the licensed core power level.

12.4.2 Occupational Radiation Exposures

In general, several factors have been found which effect personnel exposures. These factors are:

a. The number of years a plant has operated. PWR exposures tend to show large increases during the first few years of operation, but tend to level out after several years

b. Plant design and equipment layout

c. The extent that maintenance is required in a specific year

d. The amount of corrosion products, fission products and activation material deposited on or circulating through various parts of the plant systems

e. Training and experience of workers

f. The extent of direct supervision inside the radiation control area

g. The extent that a utility uses unfamiliar or contractor personnel

h. Extended operation with failed fuel

i. The integrity of the primary fluid systems.

Radiation exposures in the station are primarily due to direct and scattered radiation from components and equipment containing radioactive fluids. In some plant radiation areas, personnel can also be exposed to airborne radionuclides.

Figure 12.3-1, Figure 12.3-2, Figure 12.3-3, Figure 12.3-4, Figure 12.3-5, Figure 12.3-6, Figure 12.3-7, Figure 12.3-8, Figure 12.3-9, Figure 12.3-10, Figure 12.3-11, Figure 12.3-12, Figure 12.3-13, Figure 12.3-14, Figure 12.3-15, Figure 12.3-16 and Figure 12.3-17 identify the locations of the major sources of radioactivity and the classifications of the various radiation zones in the plant. In Zone I areas, such as offices and control rooms, the maximum dose rate does not exceed 0.5 mrem/hour. Occupancy in Zones II through V is limited by administrative controls to ensure that personnel do not receive doses in excess of the 10 CFR 20 limits. (See Table 12.3-1 for estimated occupancies of radiation zones by work function.)

12.4.2.1 Operating PWR Data

Reports summarizing the occupational radiation exposures at operating nuclear power stations (References 1-5) have provided important input data for the design of Seabrook Station. To cite specific examples, the operational exposure data indicate that steam generator maintenance and in-service inspection are key high exposure activities (Reference 1). An average of 27 percent of the total annual exposures at PWRs can be attributed to steam generator work. In-service inspection is the next most significant activity, causing an average of 5.6 percent of the total annual exposures.

Operating data given in the above references has helped to identify the most significant dose-causing activities, and to establish the priorities for ALARA-related design reviews and improvements.

12.4.2.2 Direct Radiation Dose Estimates

Average annual doses within Seabrook Unit 1 were estimated by comparing the plant's design features with the appropriate historical exposure data from operating plants. Expected radiation fields and exposure times were projected for all major activities within the plant. The estimated dose rates and exposure times for each task were multiplied to obtain an estimate of the occupational radiation exposure for each task or activity, and summed to obtain the expected annual dose for the unit.

The estimated annual occupational radiation exposure is expected to be approximately 372 man-rem. Table 12.4-1 summarizes the exposures by major task; Table 12.4-2, Table 12.4-3, Table 12.4-4, Table 12.4-5, Table 12.4-6 and Table 12.4-7 provide the detailed dose and exposure time estimates by major task.

12.4.2.3 Inhalation Exposure

The annual man-rem dose from airborne activity in the Seabrook plant will be very small compared with the annual man-rem doses due to direct and scattered radiation. Potential sources of airborne radioactivity (Subsection 12.2.2) have been analyzed and reduced to levels that are considered as low as reasonably achievable for both occupational and nonoccupational exposures.

As a result of operational occurrences and maintenance activities, personnel are required to enter areas where airborne radioactivity exists, or is expected. In such cases, the airborne concentrations are monitored by health physics personnel. The necessary protective devices (respiratory protective devices, protective clothing, etc.) and, if appropriate, control devices (portable ventilation, containment, etc.) are specified according to the concentration(s) and radionuclide(s) present. The respiratory protective devices are used in conjunction with administrative controls (Section 12.5) to limit occupancy times for personnel working in a contaminated atmosphere. Records of the inhalation exposures accrued by plant personnel are maintained.

In summary, inhalation doses to personnel are limited by:

a. Maintaining positive control to ventilation air in contaminated work areas

b. Health physics survey of work areas to identify the radionuclide concentrations present in personnel work areas.

c. Controlling the occupancy time in areas with airborne contamination, and, when necessary, requiring personnel to use respiratory protective devices and protective clothing

d. Actions taken in response to positive whole body or bioassay results.

The measures described above minimize exposures to airborne radioactivity and ensure that the doses to individual employees from airborne radioactivity are small fractions of the 10 CFR 20 limits for occupational workers, and that annual man-rem doses comply with the ALARA criteria. Periodic whole body counts and special bioassays, as described in Subsection 12.5.3.5, are conducted to monitor personnel internal exposure and to evaluate the effectiveness of the program.

12.4.2.4 Exposure Reduction Methodology

Plant design features which are applicable to the Occupational Radiation Exposure (ORE) - ALARA program are enumerated in Subsection 12.3.1.

The plant layout is designed so that passage through a high radiation area to obtain access to a lower radiation area is avoided.

Wherever practicable, components are shielded individually to keep exposures ALARA during maintenance and inspection activities. The reactor cavity is specially shielded to reduce neutron dose rates at the operating floor.

Auxiliary control boards are located with ALARA concepts in mind, and shielded pipe chases are used extensively in order to segregate radioactive piping from normally occupied areas.

The plant's ventilation system (Section 9.4) is designed to maintain the concentration of airborne radioactivity in occupied areas well below the occupational concentration values specified in 10 CFR 20, Appendix B, Table 1, Column 3. The airborne radioactivity monitoring instrumentation (Subsection 12.3.4) provides radiation measurements, alarms, and controls at selected locations in areas where the potential exposures are most likely.

To further reduce exposures, Seabrook Station uses redesigned steam generators that need considerably less maintenance. Improvements in secondary system water chemistry and tube support plates have reduced the likelihood of the need to plug tubes. Also, the volatile chemical treatment employed for the secondary system alleviates the need for sludge lancing of the secondary side.

Steam generator cladding and tube inspections will be performed wherever possible with remotely operated equipment, and the mechanical tube plugging technique will be used when feasible. This process has proved successful in reducing the doses from this task by more than a factor of two compared to the manual welding method.

Seabrook Station has incorporated a simplified reactor vessel head assembly (SHA) into the plant design to reduce personnel exposure and outage time related to reactor vessel head removal and reinstallation.

The SHA is a modification to the reactor vessel head assembly which replaces the existing missile shield, eliminates the CRDM ductwork simplifies the control rod drive mechanism and control rod drive rod position indication cable configurations, and allows the reactor vessel head lift rig to remain permanently mounted on the reactor vessel head.

Installation of the SHA is estimated to save 3 person-rem each refueling outage by eliminating missile shield/ventilation duct removal and installation. These changes are reflected in the updated dose estimate shown in Table 12.4-4. Table 12.4-4 also has been updated to eliminate disconnecting/reconnecting reactor head instrument port thermocouples. Reactor head instrument port thermocouples are not employed in the Seabrook design. This results in an additional reduction of 4 person-rem from the refueling operations dose estimate shown in Table 12.4-4.

The summary in Table 12.4-1 has also been updated to reflect these changes.

12.4.3 Estimated Annual Dose at the Site Boundary and to the Populationat Large

12.4.3.1 Direct And Scattered Radiation Dose Estimate

For normal plant operations, the maximum direct and scattered dose rate external to the reactor, the Primary Auxiliary Building, and the Waste Processing Building is less than 0.5 mrem/hour by design. The shielding design is based on a conservative assumption of a 1 percent failed fuel fission product source term. Actual levels are expected to-be significantly less than the 0.5 mrem/hour design value. A level of 0.5 mrem/hour at the outside surface of the Primary Auxiliary Building would result in a dose rate of less than 0.05 mrem/year at the exclusion area boundary.

The principal sources of radioactivity not stored in plant structures are the radioactive liquids stored in the reactor makeup water storage tanks and the refueling water storage tanks that are located in the Tank Farm. The location of the Tank Farm is shown on Figure 1.2-1. The maximum expected radionuclide inventories in each of these tanks is provided in Table 12.2-21 and Table 12.2-22. The total direct radiation dose rate at the exclusion area boundary from all four of these tanks with the maximum radionuclide inventories is calculated to be less than 5 mrem/year.

12.4.3.2 Doses Due to Liquid And Gaseous Releases

Estimates of the doses at the site boundary and beyond from station liquid and gaseous releases are provided in Subsections 11.2.3 and 11.3.3, respectively.

12.4.4 References

1. "Compilation and Analysis of Data On Occupational Radiation Exposure Experienced at Operating Nuclear Power Plants," Atomic Industrial Forum, Inc., National Environmental Studies Project, September 1974.
2. "Occupational Radiation Exposure at Light Water Cooled Power Reactors 1969-1974," NUREG-75/032, June 1975.
3. "Occupational Radiation Exposures at Light Water Cooled Power Reactors 1976," NUREG-0323, March 1978.
4. "Occupational Radiation Exposure, Tenth Annual Report, 1977," NUREG-0463, October 1978.
5. "Occupational Radiation Exposure at Commercial Nuclear Power Reactors 1978," NUREG-0594, November 1979.
6. "Technology Planning Study of Occupational Radiation Exposure in LWR Nuclear Power Plants," Electric Power Research Institute, January 22, 1980.

7. "Radiation Exposure Management - The Westinghouse ALARA Program," J. D. Cohen and J. R. Magaw, presented at the International Radiation Protection Conference, Paris, December 1979.

8. "Study of the Effects of Reduced Occupational Radiation Exposure Limits on the Commercial Nuclear Power Industry," (to be published) prepared for Atomic ndustrial Forum, Inc. by Catalytic, Inc., (1979-1980).

12.5 HEALTH PHYSICS PROGRAM

The health physics functions are organized and equipped to protect plant employees from unnecessary exposure to radiation and radioactive materials.

12.5.1 Organization

The Seabrook Station staff and the organization are discussed in Section 13.1.

The health physics organization is responsible for the overall implementation of the operational health physics program. The health physics organization has the responsibility and authority to report to the Station Director on any aspect of the radiation protection program, or its implementation, as deemed necessary. The health physics organization is responsible for ensuring station compliance with the applicable federal and state radiation protection regulations.

The health physics department manager and personnel selected for temporary replacement of the health physics department manager shall meet ANSI/ANS 3.1-1981, or equivalent.

The health physics support staff may include supervisory personnel, health physicists and other personnel to assist with implementing the radiation protection program. The supervisors are responsible for directing the activities of the health physics technicians. Personnel assigned as health physics supervisors will meet the minimum qualification requirements specified in ANSI 3.1-1978, Paragraph 4.3.2.

The health physicists provide administrative and technical assistance to the health physics supervisors. Their principal responsibilities include providing program level guidance in such technical areas as internal/external dosimetry, radiation detection/measurement and dose reduction (i.e., ALARA). Minimum qualifications are as specified in ANSI 3.1-1978.

Health physics technicians are responsible for performing the routine and daily operations of the department. Technicians will meet at least the minimum qualifications applicable to their work, as specified in ANSI 3.1-1978, Paragraph 4.5.2, in order to fulfill the operating shift crew function in Subsection 13.1.2.3, and any other qualifications set by management to ensure that the technicians are capable of performing assigned work duties. These duties include performing various surveys, collecting air samples, maintaining department equipment and instrumentation and providing radiation protection/control coverage, as necessary, during station operations and maintenance activities.

Assistant health physics technicians may be employed to assist in department activities. These assistant health physics technicians would have as a minimum a high school degree. Their duties may include assisting the technicians in the performance of surveys, sampling, radiation protection, dosimetry, whole body counting, and maintaining department equipment.

The assistant technicians participate in the health physics technician qualification program, and when an individual has three years experience, or otherwise meets the technician qualification requirements, he is eligible for the technician position.

Section 13.2 contains information on training that will be given to the Health Physics department. In addition to the formal training provided by the training department, the health physics staff will provide additional specialized instruction to their department technicians.

12.5.2 Equipment, Instrumentation and Facilities

The selection criteria for equipment and instrumentation presented can be met by several manufacturers. Equipment is purchased from manufacturers that can supply suitable equipment and instrumentation, provide repair services when required, and provide replacement parts without undue delay.

Facility design and equipment are selected to facilitate dose reduction.. The facilities are designed with adequate working spaces and for ease of access from working locations. Decontamination facilities are located at the Fuel Storage Building, Waste Processing Building, and in the Administration and Services Building.

12.5.2.1 Counting Room Equipment

The instrumentation in the counting rooms is used for determining airborne radionuclide concentrations, removable contamination, and radionuclide concentrations in liquid samples.

Two counting rooms house laboratory radiation detection equipment, one for health physics support and the other for chemistry support. The health physics count room is equipped with alpha, beta and gamma detection equipment to analyze routine air samples and contamination survey smears. The health physics counting equipment is supplemented by the chemistry counting room when additional analytical capabilities in a low background area are required. The gamma detection equipment includes germanium detectors coupled to gamma spectroscopy equipment. This gamma detection capability is available in both the chemistry and health physics counting rooms.

This equipment will be capable of detecting, as a minimum, alpha, beta and gamma activity (as specified above). This counting room equipment is used primarily for quantitative and qualitative analysis of liquid, smear, and air samples.

Criteria for equipment selection are numerous and include accuracy, stability under various atmospheric conditions, sensitivity, and compatibility with many types of peripherals.

Calibration and operational equipment checks are performed and documented on a routine basis. Radiation background and detection performance factors are normally checked prior to equipment use each day. Calibration of equipment is performed by the appropriate department technicians periodically and will include an efficiency check, operational checks, and (when applicable) a plateau check. All equipment is recalibrated upon completion of major maintenance.

12.5.2.2 Portable Instruments for Measuring Radiation

The portable instrumentation consists of low- and high- range ion chamber dose rate meters, Geiger-Mueller count rate meters, scintillation or proportional alpha counters, neutron rate meters, and air samplers. The selection of instrumentation has been partially based on equipment suitability for use during emergencies, because emergency conditions require a wide range of equipment capable of withstanding extreme usage conditions.

Criteria considered for equipment selection include linearity of response at different temperatures and humidities, reliability of response, accuracy, geometric and energy dependence, stability, and dependability. Other desirable criteria include versatility, ease of obtaining spare parts on a timely basis, sturdiness, compactness, weight, and ease of operation.

Sufficient quantities of each type of instrument are available to permit calibration, preventative and corrective maintenance, and handling of peak work loads without diminishing the radiation protection program. The types, quantities, and other information about instrumentation are presented in Table 12.5-1.

Most instruments are normally stored at the main health physics control station where they are easily accessible. Extra equipment not intended for daily use is stored in other health physics storage areas. Instruments may also be stored at temporary control points to make them more accessible from some work areas.

Calibration of portable survey instrumentation in use will be conducted on a semiannual basis or as necessary due to maintenance, questionable accuracy, or possible damage. Prior to use, an operational check will be performed which will include a battery check and a source check.

Calibration techniques of air samplers will vary according to type. The air samplers will be calibrated semi-annually, after maintenance that may affect operation, and when accuracy is questionable. The calibration procedure for the low volume, high volume, and personnel air samplers consists of determining the actual flow rate. Continuous air monitor (C.A.M.) calibration procedures are dependent upon type chosen, but generally consist of flow rate adjustments, source checking of detectors, and checking filter paper speed and adjustment, if necessary.

12.5.2.3 Personnel Monitoring Instruments

The instruments used for detecting personnel contamination and exposure, listed in Table 12.5-2, are state-of-the-art when practicable. The official and permanent records of accumulated external and internal radiation exposure will be generated from data acquired by this instrumentation.

Two types of friskers (personnel external contamination monitors) are used throughout the station. The first type of frisker is a portal monitor or similar large area detection device and the second type is of the portable hand held probe design. These friskers are designed for routine use by all personnel that exit the station or Radiologically Controlled Area (RCA). Criteria for frisker selection include ease of use, sensitivity to small amounts of radioactive materials, and the ability to locate contamination on the body.

Frisker locations throughout the plant are dependent upon projected and actual personnel traffic. At least one portal monitor and one hand-held probe-type frisker are located at the facility access building during normal conditions. The number of friskers and portal monitors at the facility access point may be increased during peak personnel traffic. Personnel external contamination devices are located at the RCA entrance-exit and other locations as necessary to control contamination. Other locations may include, but are not limited to:

- Waste Processing Building
- Spent Fuel Storage Building
- Primary Auxiliary Building
- RCA machine shop
- Radio-chemistry laboratory
- Containment personnel hatch

Calibration of hand-held probe-type friskers is performed at semiannual intervals, when damage may have occurred, accuracy or reliability is questionable, and when maintenance is performed that could affect operation. Calibration procedures include source response checks, sensitivity correction, and alarm operability.

Instrumentation for determining individual exposure from external sources, such as self-reading pocket dosimeters (SRPDs) or electronic dosimeters are available on site and controlled by the health physics department. Electronic Dosimeters (EDs) are the primary means of monitoring incremental exposures to individuals. SRPDs will be used as a backup. The SRPDs are of various ranges: predominately 0 to 200 mR for general use, with 0 to 1000 mR and 0 to 5000 mR for use during specific high-exposure tasks or emergency conditions.

SRPD selection is based primarily upon results of drift test, accuracy test, drop test, and usable lifetime as determined by those in use at other facilities. Dosimeters are issued by health physics personnel and stored near the Radiologically Controlled Area control point. Calibration of SRPDs is performed every six months or prior to putting the dosimeters into use. The calibration procedure provides criteria for acceptance of both the source and drift tests which are performed as calibration checks. Calibration of SRPDS will be performed prior to placing them in service.

The official and permanent record of accumulated external radiation exposure received by individuals is obtained principally from the reading of thermo-luminescence dosimeters (TLD). These dosimetry devices are capable of measuring beta, gamma, and neutron radiations in mixed radiation fields in accordance with industry standards. Dosimetry devices are issued to radiation workers after completion of check-in requirements at the station. Criteria for issuance are outlined by procedures developed by the Health Physics Department. TLD badges and processing services are provided by a National Voluntary Laboratory Accreditation Program (NVLAP) accredited laboratory.

The TLDs are processed at a frequency not exceeding the period for which dose limits are listed in 10 CFR 20 for occupational exposure. Any official dosimetry device used at Seabrook Station requiring processing is supplied and evaluated by a laboratory accredited through the National Voluntary Laboratory Accreditation Program.

Selection criteria for the whole body counter consist of quantitative accuracy, qualitative accuracy, and the ability to determine low body or organ burdens of common PWR gamma emitting radionuclides.

The whole body counter is located onsite in a low background area. Calibration of the whole body counter will be performed or verified annually, when major maintenance is performed, and when results appear to be incorrect in accordance with health physics procedures. Operational checks are performed prior to daily use and include a background count, quality control sample measurement and software verification.

An in-vitro bioassay program has been established. A vendor laboratory having the capability of analyzing urine and fecal samples for body burdens of common PWR radionuclides, including tritium is used. The health physics organization verifies that the vendor's quality assurance program assures accurate analysis. The program is used for accidents, incidents, and suspected uptakes of radionuclides not detectable by whole body counting equipment.

12.5.2.4 Personnel Protective Clothing

Protective clothing is provided for employees required to work in areas where this clothing is necessary. The protective clothing is kept at the RCA change area and, when required, at the location of work. The quantity of protective clothing that is maintained for use is calculated considering normal usage, quantity in the laundry process, and anticipated work. Examples of the types of protective clothing that may be available are as follows:

- Coveralls
- Lab coats
- Plastic suits
- Surgeon caps or similar head covers
- Cloth hoods

- Plastic shoe covers, low and high
- Rubber boots
- Rubber shoe covers
- Cotton gloves
- Rubber gloves
- Goggles
- Face shields
- Welding shields

12.5.2.5 Respiratory Protection Equipment

A respiratory protection program has been established and procedures written to reasonably ensure that personnel exposure to radionuclide concentrations do not exceed the exposure limits for internal dose as specified in 10 CFR 20. The protection factors assigned to respiratory equipment are those specified in Appendix A of 10 CFR 20.

The equipment consists of facepiece-fitting instrumentation, full-face respirators, self-contained breathing apparatus (SCBA), airfed hoods, airlines, filter canisters and cartridges, communication devices, eyeglass adapters, welding equipment plus maintenance and repair supplies.

New respirators are checked prior to initial use by a visual inspection. Used respirators are cleaned, disinfected, checked for contamination and mechanically inspected prior to reissue. Emergency respirator equipment is checked each month and an inspection log is maintained.

Respirators and associated accessories are maintained and controlled by health physics. A health physics-controlled storage area is used to store new or infrequently used respiratory equipment. The RCA access area contains the equipment that may be used on a daily basis. Each respirator not equipped with its own storage case is placed in a bag to maintain cleanliness and stored to protect it from damage. Emergency respirators are located throughout the facility for rapid accessibility.

12.5.2.6 Health Physics Support Facilities

An instrument calibration facility has been established for the calibration of portable health physics instruments. The facility is equipped with the items necessary for instrument calibration and repair.

Personnel decontamination equipment and showers are located near the Radiologically Controlled Area Control Point. Soap, chemicals, brushes, and other materials necessary for personnel decontamination are available at this location. Instrumentation capable of detecting external contamination is also available near the RCA shower area.

A facility for the decontamination and cleaning of monitoring instruments and small tools is provided. The decontamination facility contains the necessary chemicals, cleaning agents, and utensils to meet the majority of small tool and instrument decontamination needs.

A respiratory equipment cleaning, sanitizing and repair facility is provided. Located at the facility are equipment and chemicals needed to check, clean, sanitize and repair all reusable respirators and respiratory equipment.

Two separate clothing change facilities (men and women) are available for preparing for use of anti-contamination clothing and are located near the health physics control point.

The health physics control point is used for instrumentation storage, as the primary access point, radiological data information center, and health physics operations. Figures showing the location of the health physics control point and other areas discussed in this subsection are included as part of Section 1.2.

In order to support increased Radiologically Controlled Area access/egress requirements, such as those normally experienced during maintenance and refueling outages, alternate facilities may be established by health physics personnel. The facilities will be provided with the necessary equipment and capabilities to ensure equivalent control of radioactive materials and personnel, as that provided by the primary RCA Control Point.

12.5.3 Procedures

12.5.3.1 Radiation Surveys

Radiological surveys are performed to determine the concentration of radio-activity in air, direct radiation from plant components and loose or fixed contamination on surfaces. Written procedures are available which provide the method and frequency of surveys.

Surveys are performed on a routine and nonroutine basis. Some factors that determine the frequency and scope of any survey are the location of the subject area or equipment, expected occupancy of any area, type of work to be performed, handling of radioactive materials, movement of station personnel and plant operations. The frequency of routine surveys varies from once per shift to annually. Nonroutine surveys are performed as necessary to monitor dismantled equipment contamination, to establish radiological controls for and during work activities, to determine operational requirements such as the need to change process system filters, and to follow projected changes of radiation fields due to station operations.

Instrumentation of the type listed in Table 12.5-1, Portable Health Physics Instrumentation, is used to determine radiation levels. Routine surveys to determine direct radiation from sources consist of measurements taken at fixed and/or nonfixed locations. The radiation survey data enables health physics personnel to determine trends and locations within an area that may require action to prevent unnecessary exposure accumulation.

Routine surveys to determine area contamination levels may be performed simultaneously with the direct radiation surveys. This type of survey enables personnel to determine contamination trends and also provides data used to determine if contamination control procedures such as those for step-off-pad usage and anti-contamination clothing usage are adequate and effective.

Surveys to determine airborne radionuclide concentrations are performed routinely and nonroutinely. Areas such as the Spent Fuel Building, sections of the Primary Auxiliary Building, and Waste Processing Buildings are routinely surveyed for airborne radioactivity. Nonroutine air surveys are provided for initial opening of the primary system and, when necessary, to ensure worker health protection.

The surveys mentioned above are in addition to the fixed monitoring instrumentation which has been placed throughout the facility and is described in Subsection 12.3.4, Area Radiation and Airborne Radioactivity Monitoring Instrumentation.

12.5.3.2 Procedures to Maintain Exposures as Low as Reasonably Achievable (ALARA)

The general operational concept of ALARA evaluation, analysis and review is presented in Subsection 12.1.3.1. Specific ALARA review levels and guide-lines are contained in health physics procedures as discussed in Subsection 12.1.3.2. Elements of Regulatory Guide 8.8, Revision 3 are exercised through the use of pre-emptive analysis and planning, ongoing observations and audits and historical analysis and review. Examples of these include daily individual dose-tracking through the use of self-reading pocket dosimeters (SRPDs) or electronic dosimeters and dose-tracking with respect to specific jobs, work groups and components through correlation of data obtained from the radiation work permit system and exposure records. Dose-tracking in these areas is applied as time, personnel and radiological conditions warrant or health physics supervision deems necessary.

Additionally, general performance guidelines have been identified for certain evolutions and operational tasks, such as those discussed below, to further promote personnel radiation exposure reduction.

Refueling procedures state the monitoring requirements, minimum fuel-to-water surface distance, and degassification of the Reactor Coolant System prior to removing the reactor vessel head for refueling operations. During refueling, the reactor coolant is filtered and cooled to help reduce airborne contamination caused by coolant evaporation.

Prior to initial handling of spent fuel, the equipment used is checked for possible damage and proper operation. During fuel movement the minimum top of fuel-to-water surface distance is maintained above a level as stated in procedures, and airborne activity will be monitored. During the initial use of the fuel transfer canal and tube, appropriate areas are barricaded, locked to access, and detailed radiation area surveys are conducted. Thereafter, radiation protection controls are used as necessary.

When shipping spent fuel, an approved cask is used. Fuel is loaded into the cask underwater, then the cask is capped or sealed, raised, surveyed, decontaminated, if necessary, and resurveyed as necessary for compliance with appropriate federal regulations prior to release.

When spent fuel is to be transported to the dry fuel storage site, a licensed cask and transport system is used. Fuel is loaded into the cask underwater, then the cask is capped and sealed, raised, surveyed, decontaminated and resurveyed if necessary for compliance with appropriate federal regulations prior to movement to the dry fuel storage facility.

Prior to performing in-service inspection (ISI), the personnel review prior inspections, verify proper equipment operation, and review general area radiation levels and hot spot locations in the vicinity of their work. Where practicable, remote ISI techniques are used.

Procedures for radwaste handling reflect the ALARA philosophy. Radwaste operators are usually stationed at a remote control console from which remote operations can be performed. Remote operations include moving, filling, and sealing containers. Prior to container usage, when possible, shielding and labeling operations are performed. Vehicles used for shipping radwaste are surveyed in accordance with Department of Transportation regulations and Seabrook Station procedures. A ventilation system with exhaust through a High Efficiency Particulate Air (HEPA) Filter System will be used at the dry waste compaction station.

During normal operation plant personnel will receive job-related training. A direct benefit of the training should be a reduction of exposure, which can be attributed to improved efficiency. Design features of the facility such as reach rods, remote activation controls, and control panels make it possible for most operations to be performed from a low radiation area.

Routine maintenance consists of activities such as scheduled and preventive maintenance. Maintenance procedures are reviewed to ensure completeness. Work on the RCA side of the plant requires a Radiation Work Permit (RWP) with its associated radiological information and requirements. Normally, when work is to be performed involving radiation and/or radioactive materials, the responsible group requests health physics to determine if a job specific RWP is required. Health physics personnel make the determination in accordance with procedures and, if necessary, investigate the actual and/or anticipated radiological conditions. If an RWP is necessary, Health Physics specifies radiological protection requirements and authorizes, by signature, the issuance of an RWP. Each individual working under the RWP shall initial it or sign an associated form signifying he/she has read and understands the conditions of work. (The control room is aware of work being performed in the Station through the work control procedures. Health Physics notifies the control room if significant changes in the physical or radiological status of the unit occur.)

Sample collection by chemistry of the reactor coolant system gas or water is normally obtained at one of the sample stations. The sample stations are, when required, equipped with a hood, ventilation system and HEPA filter. Shielded containers and long tongs are available for handling samples when this is necessary. A shielded area has been provided in the radio-chemistry laboratory for the storage of radioactive samples.

Some instrumentation requires in-place exposure to a radioactive source during calibration. The storage and handling of sources is described in Subsection 12.5.3.7. Transfer of the sources to and from the place of calibration is usually done by using a shielded container. During calibration, when possible personnel will use shielding, time, and distance to reduce their exposure.

12.5.3.3 Physical and Administrative Measures for Controlling Access

Access to the RCA is limited to those persons whose entry is requested by station supervisors and authorized by health physics personnel. Every area inside the RCA in which radioactive materials and radiation are present shall be surveyed and conspicuously posted with the appropriate radiation caution sign(s).

Access to high radiation areas is controlled through procedures developed by health physics in accordance with Technical Specification 6.11.

12.5.3.4 Contamination Control

The limits for surface contamination in the RCA are specified in procedures. When the contamination level limits are exceeded the area will be posted as a "Contaminated Area." Additionally, all personnel are required to wear protective clothing for entry to "Contaminated Areas." The contaminated area will, when possible, be decontaminated to minimize the spread of contamination.

Material and equipment will be given an unconditional release by Health Physics if they meet the criteria as specified by station procedures. Authorization for use or movement of radioactive material outside the RCA is obtained from health physics. Control over the packaging, labeling and movement of this material is provided by health physics personnel.

The limits for personnel contamination are specified in procedures. When the contamination levels are exceeded, decontamination is performed to reduce the contamination levels to an acceptable value.

12.5.3.5 Personnel Monitoring

Dosimetry is issued to visitors and station personnel in accordance with 10 CFR 20 and health physics procedures.

Surveillance for internal deposition of radionuclides is performed using both in-vivo and in-vitro analysis methods. Health physics procedures delineate analysis frequencies, sensitivities and exposure records dispositions for all analysis performed.

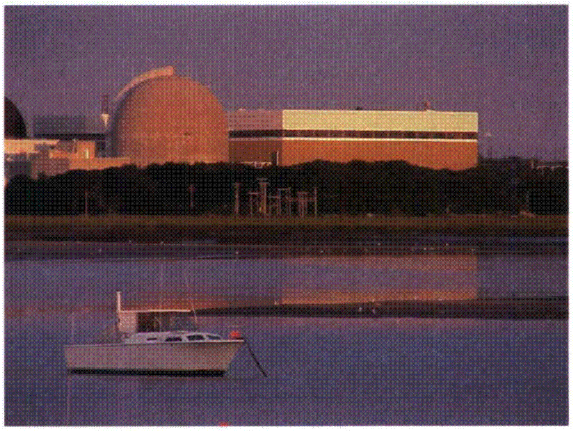
12.5.3.6 Respiratory Protection Program

Respiratory protective devices are usually required in situations arising from station operations in which airborne radioactivity areas exist or are expected. In such cases, the airborne concentrations are monitored by health physics personnel and the necessary protective devices are specified according to the concentration and type of airborne contaminants present.

SEABROOK STATION
UPDATED FINAL SAFETY ANALYSIS REPORT

CHAPTER 12
RADIATION PROTECTION

TABLES



The respiratory protection training program contains information on wearing respirators, a description of how they function, why they are used, precautions, how they are cared for, and how they are worn. A second phase of the training consists of a test, putting on a respirator, and checking for proper fit.

Respiratory protection information concerning types, uses, cleaning, and storage is given in Subsection 12.5.2.5, Respiratory Protection Equipment.

12.5.3.7 Storage and Handling of Radioactive Sources

Radioactive nonexempt sources, not part of an instrument, are controlled by health physics. The controls include source receipt, documentation, leak testing when required by Technical Specification 3/4.7.8 and assisting with the shipping process.

12.5.3.8 Updating and Auditing of the Health Physics Program

In order to keep the health physics program up-to-date and effective, procedures are reviewed periodically, rewritten as necessary, and audits are performed periodically.

Health physics procedures are reviewed periodically by health physics supervisory personnel. This review will be conducted to determine if existing procedures meet the needs of the department and, if not, to determine what modifications are necessary.

Also, when regulatory requirements change, the procedures that may need to incorporate those requirements are reviewed and, when necessary, changed.

Assessments of the health physics program effectiveness are performed regularly in accordance with 10 CFR 20 requirements.

12.5.3.9 Radiation Protection Training Program

The training given plant employees and contractors on radiation protection topics is discussed in Chapter 13.

TABLE 12.2-1 NEUTRON FLUX SPECTRUM AT REACTOR VESSEL SURFACE [HISTORICAL]

Energy	Flux (n/cm^2-sec)
E > 1 MeV	7.3+8*
5.53 keV E ≤ 1 MeV	1.2+10
0.625 eV ≤ E ≤ 5.53 keV	6.8+9
E < 0.625 eV	1.7+9

Note: The information presented in the above Table represents the radiation sources and associated input parameters, assumptions and methodology described in Section 12.2 used for the original shielding design and is retained here for historical purposes. Updating this analysis using an analyzed reactor power of 3659 MWt and operation with an 18-month fuel cycle represents a minor change in the original design basis and will have no significant impact on shielding requirements and safe plant operations.

* $7.3+8 = 7.3 \times 10^8$

TABLE 12.2-2 GAMMA FLUX SPECTRUM AT REACTOR VESSEL SURFACE [HISTORICAL]

Energy MeV	Flux (γ/cm^2-sec)
7.5	3.5+9*
4.0	3.2+9
2.5	1.6+9
0.8	9.6+8

Note: The information presented in the above Table represents the radiation sources and associated input parameters, assumptions and methodology described in Section 12.2 used for the original shielding design and is retained here for historical purposes. Updating this analysis using an analyzed reactor power of 3659 MWt and operation with an 18-month fuel cycle represents a minor change in the original design basis and will have no significant impact on shielding requirements and safe plant operations.

* 3.5+9 = 3.5×10^9

TABLE 12.2-3 SPENT FUEL SOURCE TERM [HISTORICAL]

(4 DAYS AFTER SHUTDOWN)[1]

Energy (MeV)	*MeV/cc-sec*
0.4	$1.16x10^{11}$
0.8	$4.90x10^{11}$
1.3	$1.80x10^{11}$
1.7	$2.50x10^{11}$
2.2	$9.68x10^{9}$
2.5	$2.68x10^{10}$
3.5	$9.80x10^{8}$

(100 HOURS AFTER SHUTDOWN)[2]

Energy (MeV)	*MeV/cc-sec*
0.2-0.4	$5.9x10^{10}$
0.4-0.9	$4.6x10^{11}$
0.9-1.35	$4.9x10^{10}$
1.35-1.8	$1.8x10^{11}$
1.8-2.2	$1.2x10^{10}$
2.2-2.6	$1.1x10^{10}$
2.6-3.0	$1.8x10^{8}$
3.0-4.0	$7.1x10^{7}$

Note: The information presented in the above Table represents the radiation sources and associated input parameters, assumptions and methodology described in Section 12.2 used for the original shielding design and is retained here for historical purposes. Updating this analysis using an analyzed reactor power of 3659 MWt and operation with an 18-month fuel cycle represents a minor change in the original design basis and will have no significant impact on shielding requirements and safe plant operations.

(1) Based on S.L. Anderson, L. Clemons, J.S. Moser, J. Sejvar, "Radiation Analysis Design Manual," WCAP-7664, Rev. 1.
(2) Based on "Radiation Analysis Manual Seabrook Units 1 and 2," NAH/3-1, Rev. 3, 11/78.

TABLE 12.2-3 SPENT FUEL SOURCE TERM
(PEAK ASSEMBLY SOURCE TERM 80 HOURS AFTER SHUTDOWN)[3]

Energy (MeV)	*MeV/cc-sec*
0.01	1.592×10^{10}
0.025	4.804×10^{9}
0.0375	1.242×10^{10}
0.0575	7.864×10^{9}
0.085	3.611×10^{10}
0.125	1.396×10^{11}
0.225	1.648×10^{11}
0.375	1.040×10^{11}
0.575	4.759×10^{11}
0.85	7.895×10^{11}
1.25	1.046×10^{11}
1.75	4.682×10^{11}
2.25	2.789×10^{10}
2.75	2.747×10^{10}
3.50	2.939×10^{8}
5.0	8.157×10^{2}

Note: The information presented in the above Table represents the radiation sources and associated input parameters, assumptions and methodology described in Section 12.2 used for the original shielding design and is retained here for historical purposes. Updating this analysis using an analyzed reactor power of 3659 MWt and operation with an 18-month fuel cycle represents a minor change in the original design basis and will have no significant impact on shielding requirements and safe plant operations.

[3] Based on SBC-833, Revison 1, "Minimum Water Depth Shielding Requirements for Peak Power Fuel Assembly with Extended Burnup."

TABLE 12.2-4 REACTOR COOLANT N-16 ACTIVITY* [HISTORICAL]

Position in Loop	Loop Transit Time (sec)	N-16 Activity µCi/gram
Leaving core	0	189
Leaving reactor vessel	1.1	170
Entering steam generator	1.4	164
Leaving steam generator	5.4	112
Entering reactor coolant pump	6.0	106
Entering reactor vessel	6.8	98
Entering core	9.0	86
Leaving core	9.7	189

Note: The information presented in the above Table represents the radiation sources and associated input parameters, assumptions and methodology described in Section 12.2 used for the original shielding design and is retained here for historical purposes. Updating this analysis using an analyzed reactor power of 3659 MWt and operation with an 18-month fuel cycle represents a minor change in the original design basis and will have no significant impact on shielding requirements and safe plant operations.

* S.L. Anderson, L. Clemons, J.S. Moser, J. Sejvar, "Radiation Analysis Design Manual," WCAP-7664, Rev. 3.

TABLE 12.2-5 SOURCE TERMS FOR THE CHEMICAL AND VOLUME CONTROL SYSTEM [HISTORICAL]

1. Regenerative Heat Exchanger

And

Excess Letdown Heat Exchanger

Gamma Energy (Mev/gamma)	Specific Source Strength (MeV/gm-sec)
0.4	4.5(+5)
0.8	2.7(+5)
1.3	1.7(+5)
1.7	1.2(+5)
2.2	1.4(+5)
2.5	1.6(+5)
3.5	1.9(+4)
6.1	2.2(+6)
7.1	1.8(+5)

2. Letdown Hbdt Exchanger

See Table 11.1-1

3. Mixed Bed Demineralizers

and

Cation Bed Demineralizer

Isotope	(Curies) Mixed Bed	(Curies) Cation Bed	Isotope	(Curies) Mixed Bed	(Curies) Cation Bed
I-131	1.2(+4)	-	Cs-134	4.0(+3)	4.0(+3)
I-132	7.3(+2)	-	Cs-136	3.8(+1)	3.8(+1)
I-133	2.9(+3)	-	Cs-137	2.7(+4)	2.7(+4)
I-134	1.4(+2)	-	Te-132	5.3(+2)	5.3(+2)
I-135	8.6(+2)	-	Ba-140	3.1(+1)	3.1(+1)
			La-140	3.2(+1)	3.2(+1)
Sr-89	9.1(+1)	9.1(+1)	Ce-144	3.6(+1)	3.6(+1)
Sr-90	2.1(+1)	2.1(+1)			
Sr-91	9.0(-1)	9.0(-1)	Mn-54	6.8(+1)	6.8(+1)
			Mn-56	1.9(+0)	1.9(+0)

Y-90	2.0(+1)	2.0(+1)			
Y-91	2.1(+1)	2.1(+1)	Co-58	1.0(+3)	1.0(+3)
Y-92	5.4(-2)	5.4(-2)	Co-60	8.6(+1)	8.6(+1)
Zr-95	2.4(+1)	2.4(+1)	Fe-59	2.9(+1)	2.9(+1)
Nb-95	1.4(+1)	1.4(+1)	Cr-51	1.5(+1)	1.5(+1)
Mo-99	1.3(+3)	1.3(+3)			

4. Letdown Degasifier Regenerative Heat Exchanger, Tube Side
Letdown Degasifier Preheater
Letdown Degasifier, Liquid-Vapor Region
Moderating Heat Exchanger, Tube Side
Letdown Chiller Heat Exchanger, Tube Side
and
Letdown Reheat Heat Exchanger, Shell Side

Isotope	μCi/gm	Isotope	μCi/gm
I-131	2.5(-1)	Mn-54	7.9(-5)
I-132	9.1(-2)	Mn-56	3.0(-3)
I-133	4.0(-2)	Co-58	2.6(-3)
I-134	5.8(-2)	Co-60	7.7(-5)
I-135	2.2(-1)	Fe-59	1.1(-4)
		Cr-51	9.5(-5)
Sr-89	4.1(-4)		
Sr-90	1.8(-5)	Kr-83m	4.3(-1)
Sr-91	3.1(-3)	Kr-55m	1.7(+0)
Y-90	2.2(-5)	Kr-85	1.3(-1)
Y-91	5.8(-4)	Kr-87	1.3(+0)
Y-92	1.0(-4)	Kr-88	3.4(+0)
Zr-95	6.7(-5)		
Nb-95	6.8(-5)	Xe-131m	6.7(-2)
Mo-99	3.3(-1)	Xe-133m	5.7(-1)
Cs-134	2.2(-1)	Xe-133	2.5(+1)
Cs-136	1.1(-1)	Xe-135m	8.2(-1)
Cs-137	1.1(+0)	Xe-135	3.1(+0)
Te-132	2.6(-2)	Xe-137	1.7(-1)

Ba-140	4.5(-4)	Xe-138	7.1(-1)
La-140	1.4(-4)		
Ce-144	4.4(-5)		

5. Letdown Degasifier Regenerative Heat Exchanger, Shell Side
   Letdown Degasifier, Liquid Region
   Letdown Degasifier Recirculation Pump
   and
   Letdown Degasifier Trim Cooler

Isotope	μCi/gm	Isotope	μCi/gm
I-131	2.5(-1)	Mo-99	3.3(-1)
I-132	9.1(-2)	Cs-134	2.2(-1)
I-133	4.0(-2)	Cs-136	1.1(-1)
I-134	5.8(-2)	Cs-137	1.1(+0)
I-135	2.2(-1)	Te-132	2.6(-2)
		Ba-140	4.5(-4)
Sr-89	4.1(-4)	La-140	1.4(-4)
Sr-90	1.8(-5)	Ce-144	4.4(-5)
Sr-91	3.1(-3)		
Y-90	2.2(-5)	Mn-54	7.9(-5)
Y-91	5.8(-4)	Mn-56	3.0(-3)
Y-92	1.0(-4)	Co-58	2.6(-3)
Zr-95	6.7(-5)	Co-60	7.7(-5)
Nb-95	6.8(-5)	Fe-59	1.1(-4)
		Cr-51	9.5(-5)

6. Letdown Degasifier, Vapor Region

Isotope	μCi/cc
I-131	1.2(-1)
I-132	3.8(-2)
I-133	1.9(-1)
I-134	2.0(-2)
I-135	1.0(-1)

Kr-83m	2.4(+1)
Kr-85m	1.0(+2)
Kr-85	8.4(+0)
Kr-87	6.6(+1)
Kr-88	2.0(+2)
Xe-131m	4.3(+0)
Xe-133m	3.6(+1)
Xe-133	1.6(+3)
Xe-135m	2.4(+1)
Xe-135	1.9(+2)
Xe-137	1.8(+0)
Xe-138	1.9(+1)

7. Volume Control Tank, Liquid

and

Charging Pumps

Same as reactor coolant, less noble gases.

8. Volume Control Tank, Vapor

Isotope	μCi/cc	Isotope	μCi/cc
I-131	9.8(-1)	Kr-87	5.1(+0)
I-132	4.6(-2)	Kr-88	2.7(+1)
I-133	9.4(-1)		
I-134	1.2(-2)	Xe-131m	3.7(+0)
I-135	2.6(-1)	Xe-133m	2.5(+1)
		Xe-133	1.3(+3)
Kr-83m	2.4(+0)	Xe-135m	1.1(+1)
Kr-85m	2.0(+1)	Xe-135	8.5(+1)
Kr-85	7.4(+0)	Xe-137	3.6(-2)
		Xe-138	5.5(-1)

9. Seal Water Heat Exchanger

See Table 11.1-1

10. Moderating Heat Exchanger, Shell Side

Isotope	µCi/gm	Isotope	µCi/gm	Isotope	µCi/gm
Cr-51	9.5-5	Mo-99	3.3-1	Kr-83m	4.3-1
Mn-54	7.9-5	Te-132	2.6-2	Kr-85m	1.7+0
Mn-56	3.0-3	Cs-134	2.2-1	Kr-85	1.3-1
Fe-59	1.1-4	Cs-136	1.1-1	Kr-87	1.3+0
Co-58	2.6-3	Cs-137	1.1+0	Kr-88	3.4+0
Co-60	7.7-5	Ba-140	4.5-4	Xe-131m	6.7-2
		La-140	1.4-4	Xe-133m	5.7-1
Sr-89	4.1-4	Ce-144	4.4-5	Xe-133	2.5+1
Sr-90	1.8-5	I-131	3.4+2	Xe-135m	8.2-1
Sr-91	3.1-3	I-132	3.4+1	Xe-135	3.1+0
Y-90	2.2-5	I-133	4.6+2	Xe-137	1.7-1
Y-91	5.8-4	I-134	8.4+0	Xe-138	7.1-1
Y-92	1.0-4	I-135	1.8+2		
Zr-95	6.7-5				
Nb-95	6.8-5				

11. Letdown Reheat Heat Exchanger, Tube Side

Same as reactor coolant, see Table 11.1-1

12. Thermal Regeneration Demineralizer

Iodine	Curies	µCi/cc
I-131	8.0(+1)	1.7(+2)
I-132	8.0(+0)	1.7(+1)
I-133	1.1(+2)	2.3(+2)
I-134	2.0(+0)	4.2(+0)
I-135	4.1(+1)	8.8(+1)

Note: The information presented in the above Table represents the radiation sources and associated input parameters, assumptions and methodology described in Section 12.2 used for the original shielding design and is retained here for historical purposes. Updating this analysis using an analyzed reactor power of 3659 MWt and operation with an 18-month fuel cycle represents a minor change in the original design basis and will have no significant impact on shielding requirements and safe plant operations.

TABLE 12.2-6 GEOMETRY OF EQUIPMENT IN THE CHEMICAL AND VOLUME CONTROL SYSTEM

Regenerative Heat Exchanger	
- Horizontal Sections (2)	
Diameter	6 in.
Length	15 ft
- Vertical Section (1)	
Diameter	10 in.
Length	3 ft 7 in.
Excess Letdown Heat Exchanger	
Diameter	8 in.
Length	13 ft^3
Volume of source fluid	1.15 ft
Letdown Heat Exchanger	
Diameter	22 in.
Length	15 ft
Volume of source fluid	75 gal.
Mixed Bed Demineralizer	
Diameter	32 in.
Height of resin	65 in.
Cation Bed Demineralizer	
Diameter	32 in.
Height of resin	65 in.
Letdown Degasifier Regen. Heat Exchanger	
Diameter	11 in.
Length	15 ft 4 in.
Degasifier Preheater	
Diameter	8 in.
Length	15 ft 4 in.
Volume source fluid	21 ft^3

<u>Letdown Degasifier</u>		
Upper Part:	Diameter	24 in.
	Length	12 ft 4 in.
Lower Part:	Diameter	36 in.
	Length	6 ft 11 in.
<u>Letdown Degasifier Recirculation Pump</u>		
	Diameter	8 in.
	Length	2 in.
<u>Letdown Degasifier Trim Cooler</u>		
	Diameter	11 in.
	Length	15 ft 4 in.
	Volume source fluid	3.6 ft^3
<u>Volume Control Tank</u>		
	Diameter	108 in.
	Height	144 in.
	Vapor Volume	420 ft^3
<u>Charging Pumps</u>		
- Centrifugal		
	Diameter	14 in.
	Length	48 in.
- Reciprocal		
	Diameter	4 in.
	Length	43 in.
<u>Seal Water Heat Exchanger</u>		
	Diameter	14 in.
	Length	13 ft 9 in.
	Volume of source fluid	14.7 ft^3
<u>Moderating Heat Exchanger</u>		
	Diameter	18 in.
	Length	18 ft 2 in.
	Shell Side Volume	17.2 ft^3

Letdown Chiller Heat Exchanger	
Diameter	20 in.
Length	205 in.
Volume source fluid	9.5 ft^3
Letdown Reheat Heat Exchanger	
Diameter	8 5/8 in.
Length	87 in.
Thermal Regeneration Demineralizers	
Diameter	48 in.
Height of resin	72 in.
Filters	
Diameter	7 in.
Length	28 in.

TABLE 12.2-7 SOURCE TERMS FOR RESIDUAL HEAT REMOVAL SYSTEM [HISTORICAL]

Isotope	μCi/gm	Isotope	μCi/gm
I-131	2.3+0	Zr-95	6.4-4
I-132	4.3-1	Nb-95	6.5-4
I-133	3.3+0	Mo-99	3.0+0
I-134	2.3-2	Cs-134	3.5-1
I-135	1.4+0	Cs-136	1.7-1
		Cs-137	1.7+0
Kr-83m	9.8-5	Te-132	2.4-1
Kr-85m	9.0-1	Ba-140	4.3-3
Kr-85	1.3-1	La-140	1.5-3
Kr-87	1.5-1	Ce-144	4.2-4
Kr-88	1.3+0		
		Mn-54	7.5-4
Xe-131m	6.6-2	Mn-56	9.6-3
Xe-133m	5.5-1	Co-58	2.5-2
Xe-133	2.5+1	Co-60	7.3-4
Xe-135m	4.5-1	Fe-59	1.0-3
Xe-135	2.8+0	Cr-51	9.0-4
Xe-137			
Xe-138	5.8-6		
Sr-89	3.9-3		
Sr-90	1.7-4		
Sr-91	2.2-2		
Y-90	2.1-4		
Y-91	5.6-3		
Y-92	4.4-4		

Note: The information presented in the above Table represents the radiation sources and associated input parameters, assumptions and methodology described in Section 12.2 used for the original shielding design and is retained here for historical purposes. Updating this analysis using an analyzed reactor power of 3659 MWt and operation with an 18-month fuel cycle represents a minor change in the original design basis and will have no significant impact on shielding requirements and safe plant operations.

TABLE 12.2-8 GEOMETRY OF EQUIPMENT IN RHR SYSTEM

RHR Heat Exchanger:

Mass of source fluid	5400 lbs.
Length	285 inches
Diameter	44 inches

RHR Pump

Volume of source fluid	3.2 ft^3
Length	19 inches
Diameter	9.5 inches

TABLE 12.2-9 STEAM GENERATOR BLOWDOWN SYSTEM SOURCE TERMS [HISTORICAL]

System Input

Isotope	μCi/gm	Isotope	μCi/gm
I-131	1.2-2	Mo-99	1.5-2
I-132	4.2-3	Cs-134	2.0-3
I-133	1.9-2	Cs-136	1.0-3
I-134	2.7-3	Cs-137	1.0-2
I-135	1.0-2	Te-132	1.2-3
		Ba-140	2.1-5
Sr-89	1.9-5	La-140	6.5-6
Sr-90	8.3-7	Ce-144	2.0-6
Sr-91	1.4-4		
Y-90	1.0-6	Mn-54	3.7-6
Y-91	2.7-5	Mn-56	1.4-4
Y-92	4.6-6	Co-58	1.2-4
Zr-95	3.1-6	Mn-60	3.6-6
Nb-95	3.1-6	Fe-59	5.1-6
		Cr-51	4.4-6

Note: The information presented in the above Table represents the radiation sources and associated input parameters, assumptions and methodology described in Section 12.2 used for the original shielding design and is retained here for historical purposes. Updating this analysis using an analyzed reactor power of 3659 MWt and operation with an 18-month fuel cycle represents a minor change in the original design basis and will have no significant impact on shielding requirements and safe plant operations.

1. Flash Tank

Isotope	Water (μCi/gm)	Steam (μCi/cc)	Isotope	Water (μCi/gm)	Steam (μCi/cc)
I-131	1.7-2	1.9-6	Mo-99	2.1-2	2.4-6
I-132	6.0-3	6.7-7	Cs-134	2.9-3	3.2-7
I-133	2.7-2	3.0-6	Cs-136	1.4-3	1.6-7
I-134	3.9-3	4.3-7	Cs-137	1.4-2	1.6-6
I-135	1.4-2	1.6-6	Te-132	1.7-3	1.9-7
			Ba-140	3.0-5	3.3-9
			La-140	9.3-6	1.0-9
Sr-89	2.7-5	3.0-9	Ce-144	2.9-6	3.2-10
Sr-90	1.2-6	1.3-10			
Sr-91	2.0-4	2.2-8	Mn-54	5.3-6	5.9-10
Y-90	1.4-6	1.6-10	Mn-56	2.0-4	2.2-8
Y-91	3.9-5	4.3-9	Co-58	1.7-4	1.9-8
Y-92	6.6-6	7.3-10	Co-60	5.1-6	5.7-10
Zr-95	4.4-6	4.9-10	Fe-59	7.3-6	8.1-10
Nb-95	4.4-6	4.9-10	Cr-51	6.3-6	7.0-10

2. Blowdown Heat Exchangers

See water above

Note: The information presented in the above Table represents the radiation sources and associated input parameters, assumptions and methodology described in Section 12.2 used for the original shielding design and is retained here for historical purposes. Updating this analysis using an analyzed reactor power of 3659 MWt and operation with an 18-month fuel cycle represents a minor change in the original design basis and will have no significant impact on shielding requirements and safe plant operations.

3. Flash Steam Condenser/Cooler Flash Tank and Distillate Pumps

Isotope	µCi/gm	Isotope	µCi/gm
I-131	8.6(-4)	Mo-99	1.1(-3)
I-132	3.0(-4)	Cs-134	1.4(-4)
I-133	1.4(-3)	Cs-136	7.1(-5)
I-134	1.9(-4)	Cs-137	7.1(-4)
I-135	7.1(-4)	Te-132	8.6(-5)
		Ba-140	1.5(-6)
Sr-89	1.4(-6)	La-140	4.6(-7)
Sr-90	5.9(-8)	Ce-144	1.4(-7)
Sr-91	1.0(-5)		
Y-90	7.1(-8)	Mn-54	2.6(-7)
Y-91	1.9(-6)	Mn-56	1.0(-5)
Y-92	3.3(-7)	Co-58	8.6(-6)
Zr-95	2.2(-7)	Co-60	2.6(-7)
Nb-95	2.2(-7)	Fe-59	3.6(-7)
		Cr-51	3.1(-7)

Note: The information presented in the above Table represents the radiation sources and associated input parameters, assumptions and methodology described in Section 12.2 used for the original shielding design and is retained here for historical purposes. Updating this analysis using an analyzed reactor power of 3659 MWt and operation with an 18-month fuel cycle represents a minor change in the original design basis and will have no significant impact on shielding requirements and safe plant operations.

4. Evaporator Bottoms

Isotope	Ci	µCi/gm	Isotope	Ci	µCi/gm
Cr-51	6.3(-2)	1.3(-2)	Mo-99	2.5(+1)	5.0(+0)
Mn-54	1.2(-1)	2.4(-2)	I-131	5.4(+1)	1.1(+1)
Mn-56	8.6(-3)	1.8(-3)	Te-132	2.3(+0)	4.6(-1)
Co-58	2.9(+0)	5.9(-1)	I-132	2.5(+0)	5.1(-1)
Fe-59	1.0(-1)	2.0(-2)	I-133	9.3(+0)	1.9(+0)
Co-60	1.3(-1)	2.6(-2)	I-134	5.7(-2)	1.1(-2)
Sr-89	4.0(-1)	8.0(-2)	Cs-134	7.1(+1)	1.4(+1)
Sr-90	3.0(-2)	6.1(-3)	I-135	1.7(+0)	3.4(-1)
Y-90	3.0(-2)	6.2(-3)	Cs-136	7.6(+0)	1.5(+0)
SR-91	3.4(-2)	6.8(-3)	Cs-137	3.7(+2)	7.5(+1)
Y-91	6.1(-1)	1.2(-1)	Ba-140	1.5(-1)	3.1(-2)
Y-92	4.0(-4)	8.0(-5)	La-140	1.6(-1)	3.2(-2)
Zr-95	7.2(-2)	1.5(-2)	Ce-144	6.6(-2)	1.3(-2)
Nb-95	9.6(-2)	2.0(-2)			
			Total	5.5(+2)	1.1(+2)

Note: The information presented in the above Table represents the radiation sources and associated input parameters, assumptions and methodology described in Section 12.2 used for the original shielding design and is retained here for historical purposes. Updating this analysis using an analyzed reactor power of 3659 MWt and operation with an 18-month fuel cycle represents a minor change in the original design basis and will have no significant impact on shielding requirements and safe plant operations.

5. Evaporator Distillate

Isotope	μCi/cc	Isotope	μCi/cc
Cr-51	1.1(-9)	I-131	2.9(-5)
Mn-54	9.2(-10)	Te-132	3.0(-7)
Mn-56	3.4(-8)	I-132	1.0(-5)
Co-58	3.0(-8)	I-133	4.6(-5)
Fe-59	1.3(-9)	I-134	6.3(-6)
Co-60	8.9(-10)	Cs-134	5.1(-7)
		I-135	2.5(-5)
Sr-89	4.7(-9)	Cs-136	2.5(-7)
Sr-90	2.1(-10)	Cs-137	2.5(-6)
Y-90	2.5(-10)	Ba-140	5.2(-9)
Sr-91	3.6(-8)	La-140	1.6(-9)
Y-91	6.7(-9)	Ce-144	5.1(-10)
Y-92	1.1(-9)		
Zr-95	7.8(-10)	Xe-131m	3.6(-9)
Nb-95	7.9(-10)	Xe-133m	1.2(-7)
Mo-99	3.8(-6)	Xe-133	2.1(-6)
Xe-135m	1.4(-4)		
Xe-135	1.2(-5)		
		Total	2.8(-4)

Note: The information presented in the above Table represents the radiation sources and associated input parameters, assumptions and methodology described in Section 12.2 used for the original shielding design and is retained here for historical purposes. Updating this analysis using an analyzed reactor power of 3659 MWt and operation with an 18-month fuel cycle represents a minor change in the original design basis and will have no significant impact on shielding requirements and safe plant operations.

6. Evaporator Distillate Condensate

Isotope	μCi/gm	Isotope	μCi/gm
Cr-51	9.4(-7)	Mo-99	3.3(-3)
Mn-54	7.9(-7)	I-131	2.5(-2)
Mn-56	2.9(-5)	Te-132	2.6(-4)
Co-58	2.6(-5)	I-132	8.6(-3)
Fe-59	1.1(-6)	I-133	4.0(-2)
Co-60	7.6(-7)	I-134	5.4(-3)
Sr-89	4.0(-6)	Cs-134	4.4(-4)
Sr-90	1.8(-7)	I-135	2.2(-2)
Y-90	2.1(-7)	Cs-136	2.2(-4)
Sr-91	3.1(-5)	Cs-137	2.2(-3)
Y-91	5.8(-6)	Ba-140	4.5(-6)
Y-92	9.4(-7)	La-140	1.4(-6)
Zr-95	6.7(-7)	Ce-144	4.4(-7)
Nb-95	6.8(-7)		
		Total	1.1(-1)

Note: The information presented in the above Table represents the radiation sources and associated input parameters, assumptions and methodology described in Section 12.2 used for the original shielding design and is retained here for historical purposes. Updating this analysis using an analyzed reactor power of 3659 MWt and operation with an 18-month fuel cycle represents a minor change in the original design basis and will have no significant impact on shielding requirements and safe plant operations.

TABLE 12.2-10 GEOMETRY OF EQUIPMENT IN STEAM GENERATOR BLOWDOWN SYSTEM

<u>Flash Tank</u>	
Height (water/vapor)	45/104 inches
Diameter	36 inches
Steam/Water Mass Ratio	3/7
<u>Flash Tank Bottoms Cooler</u>	
Volume Source Fluid	25 gallons
Length	124 inches
Diameter	16 inches
<u>Flash Steam Condenser/Cooler</u>	
Volume Source Fluid	55 gallons
Length	136 inches
Diameter	24 inches
<u>Flash Tank Distillate Pumps</u>	
Diameter	8 inches
Length	2 inches
<u>Blowdown Evaporator</u>	
Volume of Source Liquid	174.8 ft^3
Diameter	78 inches
Height	112 inches

Bottoms Pump	
Diameter	8 inches
Length	2 inches
Bottoms Cooler:	
12 straight sections	
11 curved sections	
Straight section:	
Volume Source Liquid	150 in.^{3}
Diameter	1.5 in.
Length	10 ft
Curved Section:	
Diameter	1.25 in.
Radius of curve	4.5 in.

TABLE 12.2-11 SOURCE TERMS FOR BORON RECOVERY SYSTEM [HISTORICAL]

System Input

Isotope	μCi/gm	Isotope	μCi/gm
Cr-51	9.5(-5)	I-131	2.5(-1)
Mn-54	7.9(-5)	Te-132	2.6(-2)
Mn-56	3.0(-3)	I-132	9.1(-2)
Co-58	2.6(-3)	I-133	4.0(-1)
Fe-59	1.1(-4)	I-134	5.8(-2)
Co-60	7.7(-5)	Cs-134	2.2(-1)
I-135	2.2(-1)		
Sr-89	4.1(-4)	Cs-136	1.1(-1)
Sr-90	1.8(-5)	Cs-137	1.1(+0)
Y-90	2.2(-5)	Ba-140	4.5(-4)
Sr-91	3.1(-3)	La-140	1.4(-4)
Y-91	5.8(-4)	Ce-144	4.4(-5)
Y-92	1.0(-4)		
Zr-95	6.7(-5)		
Nb-95	6.8(-5)		
Mo-99	3.3(-1)		

Note: The information presented in the above Table represents the radiation sources and associated input parameters, assumptions and methodology described in Section 12.2 used for the original shielding design and is retained here for historical purposes. Updating this analysis using an analyzed reactor power of 3659 MWt and operation with an 18-month fuel cycle represents a minor change in the original design basis and will have no significant impact on shielding requirements and safe plant operations.

Isotope	CRIE (Ci)	BWST (Ci)	Pump (μCi/gm)	Evaporator Bottom (Ci)	Bottom Pump & Cooler (μCi/gm)	Distillate Conc. (μCi/gm)	Distillate Condensate (μCi/gm)	Recovery Test Tk. (Ci)	Recovery Test Tk. Pump (μCi/gm)	Demin. (Ci)
Cr-51	5.0(-2)	4.6(-3)	9.5(-6)	4.3(-3)	7.5(-4)	6.3(-12)	5.4(-9)	3.7(-7)	Same as Evaporator Distillate Condensate Concentration	2.8(-6)
Mn-54	1.4(-1)	6.3(-3)	7.9(-6)	6.3(-3)	1.1(-3)	8.6(-12)	7.4(-9)	5.0(-7)		1.3(-5)
Mn-56	3.8(-1)	3.8(-2)	3.0(-4)	9.3(-4)	1.6(-4)	5.1(-11)	4.4(-8)	8.9(-7)		2.7(-7)
Co-58	2.5(+0)	1.7(-1)	2.6(-4)	1.7(-1)	2.9(-2)	2.3(-10)	2.0(-7)	1.4(-5)		1.9(-4)
Fe-59	7.8(-2)	6.4(-3)	1.1(-5)	6.1(-3)	1.1(-3)	8.7(-12)	7.5(-9)	5.1(-7)		5.2(-6)
Co-60	1.8(-1)	6.5(-3)	7.7(-6)	6.5(-3)	1.1(-3)	8.9(-12)	7.6(-9)	5.2(-7)		1.7(-5)
Sr-89	3.1(-1)	2.5(-2)	4.1(-5)	2.4(-2)	4.2(-3)	3.4(-11)	2.9(-8)	2.0(-6)		2.2(-5)
Sr-90	4.3(-2)	1.5(-3)	1.8(-6)	1.5(-3)	2.7(-4)	2.1(-12)	1.8(-9)	1.2(-7)		4.3(-6)
Y-90	4.4(-2)	1.6(-3)	2.2(-6)	1.6(-3)	2.8(-4)	2.2(-12)	1.9(-9)	1.3(-7)		4.3(-6)
Sr-91	6.1(-1)	6.1(-2)	3.1(-4)	5.6(-3)	9.9(-4)	8.3(-11)	7.1(-8)	3.3(-6)		3.6(-6)
Y-91	4.9(-1)	3.7(-2)	5.8(-5)	3.6(-2)	6.3(-3)	5.0(-11)	4.3(-8)	2.9(-6)		3.6(-5)
Y-92	1.5(-1)	1.5(-3)	1.0(-5)	4.9(-5)	8.7(-6)	2.0(-12)	1.7(-9)	4.5(-8)		1.9(-8)
Zr-95	6.0(-2)	4.3(-3)	6.7(-6)	4.1(-3)	7.3(-4)	5.8(-12)	5.0(-9)	3.4(-7)		4.5(-6)
Nb-95	8.4(-2)	5.2(-3)	6.8(-6)	5.1(-3)	9.1(-4)	7.1(-12)	6.1(-9)	4.2(-7)		6.9(-6)
Mo-99	8.3(+1)	8.3(+0)	3.3(-2)	4.2(+0)	7.4(-1)	1.1(-8)	9.7(-6)	6.2(-4)		1.9(-3)
I-131	0	7.8(+1)	2.5(-1)	6.1(+1)	1.1(+1)	1.1(-6)	9.2(-4)	6.1(-2)		2.6(-1)
Te-132	6.7(+0)	6.7(-1)	2.6(-3)	3.7(-1)	6.6(-2)	9.2(-10)	7.9(-7)	5.1(-5)		1.6(-4)
I-132	6.7(+0)	1.1(+1)	9.1(-2)	6.1(-1)	1.1(-1)	1.5(-7)	1.3(-4)	2.4(-3)		8.1(-4)

Note: The information presented in the above Table represents the radiation sources and associated input parameters, assumptions and methodology described in Section 12.2 used for the original shielding design and is retained here for historical purposes. Updating this analysis using an analyzed reactor power of 3659 MWt and operation with an 18-month fuel cycle represents a minor change in the original design basis and will have no significant impact on shielding requirements and safe plant operations.

Isotope	CRIE (Ci)	BWST (Ci)	Pump (μCi/gm)	Evaporator Bottom (Ci)	Bottom Pump & Cooler (μCi/gm)	Distillate Conc. (μCi/gm)	Distillate Condensate (μCi/gm)	Recovery Test Tk. (Ci)	Recovery Test Tk. Pump (μCi/gm)	Demin. (Ci)
I-133	0	8.8(+1)	4.0(-1)	1.7(+1)	3.0(+0)	1.2(-6)	1.0(-3)	5.6(-2)		1.1(-1)
I-134	0	3.3(+0)	5.8(-2)	2.7(-2)	4.8(-3)	4.4(-8)	3.8(-5)	2.7(-4)		2.9(-5)
Cs-134	4.7(+2)	1.8(+1)	2.2(-2)	1.8(+1)	3.1(+0)	2.4(-8)	2.1(-5)	1.4(-3)		4.5(-2)
I-135	0	4.0(+1)	2.2(-1)	2.6(+0)	4.5(-1)	5.5(-7)	4.7(-4)	1.8(-2)		1.5(-2)
Cs-136	4.0(+1)	4.0(+0)	1.1(-2)	3.4(+0)	6.0(-1)	5.5(-9)	4.7(-6)	3.2(-4)		1.6(-3)
Cs-137	2.6(+3)	9.4(+1)	1.1(-1)	9.4(+1)	1.7(+1)	1.3(-7)	1.1(-4)	7.5(-3)		2.6(-1)
Ba-140	1.6(-1)	1.6(-2)	4.5(-5)	1.4(-2)	2.4(-3)	2.2(-11)	1.9(-8)	1.3(-6)		6.6(-6)
La-140	9.9(-2)	9.8(-3)	1.4(-5)	1.3(-2)	2.3(-3)	1.4(-11)	1.2(-8)	8.6(-7)		6.0(-6)
Ce-144	7.8(-2)	3.5(-3)	4.4(-6)	3.5(-3)	6.1(-4)	4.8(-12)	4.1(-9)	2.8(-7)		7.3(-6)
Xe-131m		1.3(-1)				1.9(-12)				
Xe-133m		1.5(-1)				4.5(-11)				
Xe-133		4.9(+0)				7.8(-10)				
Xe-135m		1.2(+1)				5.4(-0)				
Xe-135		7.5(+0)				3.6(-9)				
Total	3.2(+3)	3.7(+2)	1.1(+0)	2.0(+2)	3.6(+1)	3.2(-6)	2.7(-3)	1.6(-1)		6.9(-1)

Note: The information presented in the above Table represents the radiation sources and associated input parameters, assumptions and methodology described in Section 12.2 used for the original shielding design and is retained here for historical purposes. Updating this analysis using an analyzed reactor power of 3659 MWt and operation with an 18-month fuel cycle represents a minor change in the original design basis and will have no significant impact on shielding requirements and safe plant operations.

Recovery Evaporator Feed Concentrations

Isotope	μCi/gm	Isotope	μCi/gm
Cr-51	5.4(-6)	Mo-99	9.7(-3)
Mn-54	7.4(-6)	I-131	9.2(-2)
Mn-56	4.4(-5)	Te-132	7.9(-4)
Co-58	2.0(-4)	I-132	1.3(-2)
Fe-59	7.5(-6)	I-133	1.0(-l)
Co-60	7.6(-6)	I-134	3.3(-3)
Cs-134	2.1(-2)		
Sr-89	2.9(-5)	I-135	4.7(-2)
Sr-90	1.8(-6)	Cs-136	4.7(-3)
Y-90	1.9(-6)	Cs-137	1.1(-1)
Sr-91	7.1(-5)	Ba-140	1.9(-5)
Y-91	4.3(-5)	La-140	1.2(-5)
Y-92	1.5(-6)	Ce-144	4.1(-6)
Zr-95	5.0(-6)		
Nb-95	6.1(-6)		

Note: The information presented in the above Table represents the radiation sources and associated input parameters, assumptions and methodology described in Section 12.2 used for the original shielding design and is retained here for historical purposes. Updating this analysis using an analyzed reactor power of 3659 MWt and operation with an 18-month fuel cycle represents a minor change in the original design basis and will have no significant impact on shielding requirements and safe plant operations.

Recovery Test Tank Concentrations

Isotope	μCi/gm	Isotope	μCi/gm
Cr-51	5.4(-9)	Mo-99	9.1(-6)
Mn-54	7.4(-9)	I-131	9.0(-4)
Mn-56	1.3(-8)	Te-132	7.5(-7)
Co-58	2.0(-7)	I-132	3.5(-5)
Fe-59	7.5(-9)	I-133	8.2(-4)
Co-60	7.6(-9)	I-134	9.0(-6)
Sr-89	2.9(-8)	Cs-134	2.1(-5)
Sr-90	1.8(-9)	I-135	2.7(-4)
Y-90	1.9(-9)	Cs-136	4.6(-6)
Sr-91	4.8(-8)	Cs-137	1.1(-4)
Y-91	4.3(-8)	Ba-140	1.9(-8)
Y-92	6.3(-10)	La-140	1.3(-8)
Zr-95	5.1(-9)	Ce-144	4.1(-9)
Nb-95	6.1(-9)		

Note: The information presented in the above Table represents the radiation sources and associated input parameters, assumptions and methodology described in Section 12.2 used for the original shielding design and is retained here for historical purposes. Updating this analysis using an analyzed reactor power of 3659 MWt and operation with an 18-month fuel cycle represents a minor change in the original design basis and will have no significant impact on shielding requirements and safe plant operations.

TABLE 12.2-12 GEOMETRY OF EQUIPMENT IN BORON RECOVERY SYSTEM

Cesium Removal Ion Exchanger			
Diameter			48 in.
Height			7 ft^3
Volume of Resin			75 ft
Boron Waste Storage Tank			
Diameter			32 ft
Height			38 ft 5 in.
Volume			225,000 Gal.
Recovery Evaporator Feed Pump			
Diameter			6 in.
Length			2 in.
Recovery Evaporator	Lower Part	Upper Part	Vapor Space
Diameter	2 ft 6 in.	5 ft 6 in.	5 ft 6 in.
Height	13 ft 6 in.	3 ft 3 in.	8 ft 9 in.
Recovery Evaporator Bottoms Pump			
Diameter			6 in.
Length			2 in.

<u>Recovery Evaporator Bottoms Cooler</u>

Straight Section (Qty 12)

Length	10 ft
Diameter	1.5 in.
Volume Source	150 in.^{3}

Curved Section (Qty 11)

Diameter	1.25 in.
Radius of Curve	4.5 in.

<u>Recovery Evaporator Reboiler Pump</u>

Volume	20 ft^3
Diameter	24 in.

<u>Recovery Evaporator Reboiler</u>

Diameter	36 in.
Height	14 ft
Volume Source Fluid	31 ft^3

<u>Recovery Evaporator Distillate Condenser</u>

Diameter	26 in.
Length	72 in.
Volume Source	16 ft^3

<u>Recovery Evaporator Distillate Accumulator</u>

Diameter	26 in.
Length	11 ft

<u>Recovery Evaporator Distillate Pump</u>

Diameter	6 in.
Length	2 in.

<u>Recovery Evaporator Distillate Cooler</u>

Length	11 ft 9 in.
Diameter	21 in.
Volume Source Liquid	8.7 ft^3

<u>Recovery Test Tank Pump</u>

Diameter	6 in.
Length	2 in.

<u>Recovery Test Tank</u>

Diameter	15 ft
Height	16 ft 6 in.
Volume	18,000 gallons

<u>Recovery Demineralizer</u>

Diameter	48 in.
Height	7 ft 8 in.
Volume of Resin	75 ft^3

<u>Recovery System Filters</u>

Diameter	7 in.
Height	28 in.

TABLE 12.2-13 SOURCE TERMS FOR THE PRIMARY DRAIN SYSTEM [HISTORICAL]

System Input

Same as reactor coolant concentrations (see Table 11.1-1).

1. Primary Drain Tank

   Primary Drain Tank Transfer Pump

   Primary Drain Tank Regenerative Heat Exchanger

   Primary Drain and Tank Degasifier Preheater

Same as reactor coolant concentrations (see Table 11.1-1).

2. Primary Drain Tank Degasifier

   Liquid and liquid-steam region: Same as reactor coolant concentrations less noble gases.

   Gas Region:

Isotope	μCi/cc
I-131	1.2(+0)
I-132	3.8(-1)
I-133	1.9(+0)
I-134	2.0(-1)
I-135	1.0(+0)
Kr-83m	2.4(+1)
Kr-85m	1.0(+2)
Kr-85	8.4(+0)
Kr-87	6.6(+1)
Kr-88	2.0(+2)
Xe-131m	4.3(+0)
Xe-133m	3.6(+1)
Xe-133	1.6(+3)
Xe-135m	2.4(+1)
Xe-135	1.9(+2)
Xe-137	1.8(+0)
Xe-138	1.9(+1)
Total	2.3(+3)

Note: The information presented in the above Table represents the radiation sources and associated input parameters, assumptions and methodology described in Section 12.2 used for the original shielding design and is retained here for historical purposes. Updating this analysis using an analyzed reactor power of 3659 MWt and operation with an 18-month fuel cycle represents a minor change in the original design basis and will have no significant impact on shielding requirements and safe plant operations.

TABLE 12.2-14 GEOMETRY OF EQUIPMENT IN PRIMARY DRAIN SYSTEM

<u>Primary Drain Tank (PDT)</u>	
Diameter	11 ft
Height	147 in.
<u>PDT Transfer Pump</u>	
Diameter	6 in.
Length	2 in.
<u>PDT Regenerative Heat Exchanger</u>	
Diameter	11 in.
Length	15 ft 4 in.
<u>PDT Degasifier Preheater</u>	
Diameter	8 in.
Length	15 ft 4 in.
Volume Source Fluid	2.1 ft^3
<u>PDT Degasifier</u>	
Gas Region:	
Diameter	24 $in.^3$
Volume	7 ft
<u>Liquid-Steam Region:</u>	
Diameter	24 in.
Height	12 ft 4 in.
Liquid Region:	
Diameter	36 in.

Height	6 ft 11 in.

PDT Degasifier Recirculation Pump

Diameter	8 in.
Length	2 in.

PDT Degasifier Trim Cooler

Diameter	11 in.
Length	15 ft 4 in.
Volume Source Fluid	3.6 ft^3

PDT Degasifier Prefilter

Diameter	7 in.
Length	28 in.

TABLE 12.2-15 SOURCE TERMS FOR SPENT RESIN SLUICE SYSTEM [HISTORICAL]

1. Spent Resin Sluice Tank and Spent Resin Transfer Pump

Isotope	Total (Ci)	Conc. (μCi/gm)
Cr-51	1.5(+1)	8.1(-1)
Mn-54	6.6(+1)	3.6(+0)
Mn-56	-	-
Co-58	9.9(+2)	5.4(+1)
Fe-59	2.8(+1)	1.5(+0)
Co-60	8.4(+1)	4.6(+0)
Sr-89	9.1(+1)	4.9(+0)
Sr-90	2.0(+1)	1.1(+0)
Y-90	2.0(+1)	1.1(+0)
Sr-91	-	-
Y-91	2.0(+1)	1.1(+0)
Y-92	-	-
Zr-95	2.4(+1)	1.3(+0)
Nb-95	1.3(+1)	7.1(-1)
Mo-99	1.3(+3)	7.1(+1)
I-131	1.2(+4)	6.5(+2)
Te-132	5.3(+2)	2.9(+1)
I-132	5.3(+2)	2.9(+1)
I-133	-	-
I-134	-	-
Cs-134	4.7(+3)	2.6(+2)
I-135	-	-
Cs-136	1.7(+2)	9.2(+0)
Cs-137	2.7(+4)	1.5(+3)

Ba-140	3.1(+1)	1.7(+0)
La-140	3.1(+1)	1.7(+0)
Ce-144	3.6(+1)	2.0(+0)
Total	4.8(+4)	2.6(+3)

2. Spent Resin Sluice Pump

Concentrations are 10^{-4} that of the sluice tank

Note: The information presented in the above Table represents the radiation sources and associated input parameters, assumptions and methodology described in Section 12.2 used for the original shielding design and is retained here for historical purposes. Updating this analysis using an analyzed reactor power of 3659 MWt and operation with an 18-month fuel cycle represents a minor change in the original design basis and will have no significant impact on shielding requirements and safe plant operations.

TABLE 12.2-16 GEOMETRY OF EQUIPMENT IN SPENT RESIN SLUICE SYSTEM

Spent Resin Sluice Tank	
Diameter	11 ft
Height	16 ft
Volume of Resin	650 ft^3
Spent Resin Transfer Pump	
Diameter	6 in.
Length	72 in.
Volume	0.6 ft^3
Spent Resin Sluice Pump	
Diameter	8 in.
Length	2 in.

TABLE 12.2-17 SOURCE TERM FOR SPENT FUEL POOL CLEANUP SYSTEM [HISTORICAL]

Spent Fuel Pool Demineralizer

Isotope	μCi/cc	Isotope	μCi/cc
Cr-51	4.62-4	Mo-99	8.65-2
Mn-54	5.16-4	Te-132	1.09-2
Mn-56	-	Cs-134	4.01+0
Fe-59	6.06-4	Cs-136	1.02+0
Co-58	1.54-2	Cs-137	2.03+1
Co-60	5.16-4	Ba-140	1.50-3
		La-140	1.71-3
Sr-89	2.31-3	Ce-144	2.87-4
Sr-90	1.21-4	I-131	5.51-1
Sr-91	-	I-132	1.12-2
Y-90	1.22-4	I-133	8.28-5
Y-91	3.47-3	I-134	-
Y-92	-	I-135	-
Zr-95	3.93-4		
Nb-95m	3.76-4		

Note: The information presented in the above Table represents the radiation sources and associated input parameters, assumptions and methodology described in Section 12.2 used for the original shielding design and is retained here for historical purposes. Updating this analysis using an analyzed reactor power of 3659 MWt and operation with an 18-month fuel cycle represents a minor change in the original design basis and will have no significant impact on shielding requirements and safe plant operations.

TABLE 12.2-18 GEOMETRY OF EQUIPMENT IN SPENT FUEL POOL CLEANUP SYSTEM

Spent Fuel Pool Demineralizer

Diameter	48.0 in.
Height	72.0 in.
Volume	75.0 ft^3

Spent Fuel Pool Demineralizer Prefilter

And

Spent Fuel Pool Demineralizer Postfilter

Diameter	7 in.
Height	28 in.

TABLE 12.2-19 SOURCE TERMS FOR MISCELLANEOUS CHEMICAL DRAIN SYSTEM [HISTORICAL]

System Input:

Isotope	μCi/gm	Isotope	μCi/gm
		I-131	1.1(-1)
Cr-51	4.2(-5)	Mo-99	1.5(-1)
Mn-54	3.5(-5)	Te-132	1.1(-2)
Co-58	1.1(-3)	I-132	4.0(-2)
Fe-59	4.8(-5)	Cs-134	1.9(-2)
Co-60	3.4(-5)	Cs-136	9.7(-3)
		Cs-137	9.7(-2)
Sr-89	1.8(-4)	Ba-140	2.0(-4)
Sr-90	7.9(-6)	La-140	6.2(-5)
Y-90	9.7(-6)	Ce-144	1.9(-5)
Y-91	2.6(-4)		
Zr-95	2.9(-5)		
Nb-95	3.0(-5)		

Note: Short-lived Isotopes (T½ <1 day) are neglected unless they are the daughter of a long-lived isotope.

1. Chemical Drain Tank

Isotope	Ci	μCi/gm	Isotope	Ci	μCi/gm
Cr-51	1.4(-4)	3.7(-5)	Mo-99	2.1(-1)	5.6(-2)
Mn-54	1.3(-4)	3.5(-5)	I-131	2.8E(-1)	7.5(-2)
Co-58	4.0(-3)	1.0(-3)	Te-132	1.8(-2)	4.6(-3)
Fe-59	1.7(-4)	4.4(-5)	I-132	2.0(-2)	5.2(-3)
Co-60	1.3(-4)	3.4(-5)	Cs-134	7.1(-2)	1.9(-2)
			Cs-136	2.9(-2)	7.6(-3)
Sr-89	6.4(-4)	1.7(-4)	Cs-137	3.7(-1)	9.7(-2)
Sr-90	3.0(-5)	7.9(-6)	Ba-140	5.9(-4)	1.6(-4)
Y-90	3.2(-5)	8.5(-6)	La-140	5.2(-4)	1.4(-4)

Y-91	9.3(-4)	2.4(-4)	Ce-144	7.1(-5)	1.9(-5)
Zr-95	1.0(-4)	2.7(-5)			
Nb-95	1.1(-4)	3.0(-5)	Total	1.0(+0)	2.6(-1)

2. Chemical Drain Transfer Pump

See specific source term for Chemical Drain Tank

3. Chemical Drain Treatment Tank and Chemical Drain Treatment Pump

Isotope	Ci	µCi/gm	Isotope	Ci	µCi/gm
Cr-51	3.5(-4)	2.4(-5)	Mo-99	8.7(-2)	5.9(-3)
Mn-54	4.9(-4)	3.3(-5)	I-131	3.3(-1)	2.2(-2)
Co-58	1.2(-2)	8.3(-4)	Te-132	8.4(-3)	5.7(-4)
Fe-59	4.9(-4)	3.3(-5)	I-132	8.7(-3)	5.9(-4)
Co-60	5.0(-4)	3.4(-5)	Cs-134	2.8(-1)	1.9(-2)
			Cs-136	4.8(-2)	3.3(-3)
Sr-89	2.0(-3)	1.3(-4)	Cs-137	1.4(+0)	9.7(-2)
Sr-90	1.2(-4)	7.9(-6)	Ba-140	1.0(-3)	6.8(-5)
Y-90	1.2(-4)	7.9(-6)	La-140	1.1(-3)	7.5(-5)
Y-91	2.9(-3)	1.9(-4)	Ce-144	2.7(-4)	1.8(-5)
Zr-95	3.3(-4)	2.2(-5)			
Nb-95	4.1(-4)	2.8(-5)	Total	2.2(+0)	1.5(-1)

Note: The information presented in the above Table represents the radiation sources and associated input parameters, assumptions and methodology described in Section 12.2 used for the original shielding design and is retained here for historical purposes. Updating this analysis using an analyzed reactor power of 3659 MWt and operation with an 18-month fuel cycle represents a minor change in the original design basis and will have no significant impact on shielding requirements and safe plant operations.

TABLE 12.2-20 EQUIPMENT GEOMETRY FOR THE MISCELLANEOUS CHEMICAL DRAIN SYSTEM

<u>Chemical Drain Tank</u>	
Diameter	5 ft
Height	8 ft 8 in.
<u>Chemical Drain Transfer Pump</u>	
Diameter	8 in.
Length	2 in.
<u>Chemical Drain Treatment Tank</u>	
Diameter	8 ft
Height	10 ft 9 in.
<u>Chemical Drain Treatment Pump</u>	
Diameter	6 in.
Length	2 in.

TABLE 12.2-21 MAXIMUM EXPECTED RADIONUCLIDE CONTENT OF A REACTOR MAKEUP WATER STORAGE TANK [HISTORICAL]

Radionuclide	Tank Inventory (Curies)	Radionuclide	Tank Inventory (Curies)
I-131	1.1-1	Mo-99	2.2+0
1-132	3.8-2	Cs-134	1.1-1
I-133	1.7-1	Cs-136	5.9-2
I-134	2.3-2	Cs-137	5.5-1
I-135	9.3-2	Te-132	1.1-2
		Ba-140	5.9-3
Sr-89	1.6-3	La-140	1.4-5
Sr-90	5.5-5	Ce-144	1.4-5
Sr-91	8.1-4		
Y-90	6.4-5	Mn-54	3.3-5
Y-91	2.5-3	Mn-56	1.2-3
Y-92	3.1-4	Co-58	1.1-3
Zr-95	3.0-5	Co-60	3.1-5
Nb-95	2.9-5	Fe-59	4.2-5
		Cr-51	3.9-5

TOTAL = 3.4 Curies

Note: The information presented in the above Table represents the radiation sources and associated input parameters, assumptions and methodology described in Section 12.2 used for the original shielding design and is retained here for historical purposes. Updating this analysis using an analyzed reactor power of 3659 MWt and operation with an 18-month fuel cycle represents a minor change in the original design basis and will have no significant impact on shielding requirements and safe plant operations.

TABLE 12.2-22 MAXIMUM EXPECTED RADIONUCLIDE CONTENT OF A REFUELING WATER STORAGE TANK [HISTORICAL]

Radionuclide	Tank Inventory (Curies)	Radionuclide	Tank Inventory (Curies)
I-131	7.7+1	Mo-99	6.0+0
I-132	<1.0-7	Cs-134	4.2+1
I-133	7.4-5	Cs-136	8.0+0
I-134	<1.0-7	Cs-137	2.2+2
I-135	1.0-7	Te-132	6.3-1
		Ba-140	7.8-2
Sr-89	4.8-1	La-140	1.4-5
Sr-90	2.2-2	Ce-144	5.2-2
Sr-91	<1.0-7		
Y-90	1.4-4	Mn-54	1.2-1
Y-91	7.8-1	Mn-56	<1.0-7
Y-92	<1.0-7	Co-58	3.4+0
Zr-95	9.5-2	Co-60	1.2-1
Nb-95	7.7-2	Fe-59	1.2-1
		Cr-51	9.3-2

TOTAL = 359 Curies

Note: The information presented in the above Table represents the radiation sources and associated input parameters, assumptions and methodology described in Section 12.2 used for the original shielding design and is retained here for historical purposes. Updating this analysis using an analyzed reactor power of 3659 MWt and operation with an 18-month fuel cycle represents a minor change in the original design basis and will have no significant impact on shielding requirements and safe plant operations.

TABLE 12.2-23 GEOMETRY OF WATER STORAGE TANKS

Reactor Makeup Water Storage Tank

Diameter	26 ft
Height	29 ft 3 in.

Refueling Water Storage Tank

Diameter	44 ft
Height	43 ft

TABLE 12.2-24 LIQUID WASTE SYSTEM SOURCE TERMS [HISTORICAL]

Isotope	Feed (µCi/gm)	Floor Drain Tank (Ci)	Floor Drain Tank Pump (µCi/gm)	Waste Evap Bottom (Ci)	Waste Evap Distillate (µCi/cc)	Distillate Condensate (µCi/gm)	Waste Test Tank (Ci)	Waste Test Tank Concentration (µCi/gm)	Waste Demin. (Ci)	Evap. Bottom Pump (µCi/gm)
Cr-51	7.1(-5)	2.7(-3)	9.5(-5)	3.8(-2)	8.3(-12)	7.1(-9)	6.7(-7)	7.1(-9)	3.0(-6)	6.7(-3)
Mn-54	5.9(-5)	2.2(-3)	7.9(-5)	3.3(-2)	6.8(-12)	5.8(-9)	5.5(-7)	5.8(-9)	1.3(-5)	5.8(-3)
Mn-56 (1)										
Co-58	2.0(-3)	7.6(-2)	2.6(-3)	1.1(+0)	2.3(-10)	2.0(-7)	1.9(-5)	2.0(-7)	2.0(-4)	1.9(-1)
Fe-59	8.3(-5)	3.1(-3)	1.1(-4)	4.5(-2)	9.6(-12)	8.2(-9)	7.7(-7)	8.1(-9)	5.5(-6)	7.9(-3)
Co-60	5.8(-5)	2.2(-3)	7.7(-5)	3.3(-2)	6.8(-12)	5.8(-9)	5.5(-7)	5.8(-9)	1.7(-5)	5.8(-3)
Sr-89	3.1(-4)	1.2(-2)	4.1(-4)	1.8(-1)	3.7(-11)	3.2(-8)	3.0(-6)	3.2(-8)	2.4(-5)	3.2(-2)
Sr-90	1.4(-5)	5.3(-4)	1.8(-5)	8.0(-3)	1.6(-12)	1.4(-9)	1.3(-7)	1.4(-9)	4.2(-6)	1.4(-3)
Y-90	1.7(-5)	5.3(-4)	1.8(-5)	8.0(-3)	1.6(-12)	1.7(-9)	1.3(-7)	1.4(-9)	4.2(-6)	1.6(-3)
Sr-91 (1)										
Y-91	4.4(-4)	1.7(-2)	5.8(-4)	2.5(-1)	5.3(-11)	4.5(-8)	4.3(-6)	4.5(-8)	3.9(-5)	4.4(-2)
Y-92 (1)										
Zr-95	5.0(-5)	1.9(-3)	6.7(-5)	2.8(-2)	5.8(-12)	5.0(-9)	4.7(-7)	5.0(-9)	4.7(-6)	4.9(-3)

(1) Those isotopes with half-life less than one day are neglected unless they are decay products of long-lived isotopes

(2) For the above decay product, parent activities are used for source term assuming secular equilibrium is achieved.

Isotope	Feed (μCi/gm)	Floor Drain Tank (Ci)	Floor Drain Tank Pump (μCi/gm)	Waste Evap Bottom (Ci)	Waste Evap Distillate (μCi/cc)	Distillate Condensate (μCi/gm)	Waste Test Tank (Ci)	Waste Test Tank Concentration (μCi/gm)	Waste Demin. (Ci)	Evap. Bottom Pump (μCi/gm)
Nb-95	5.1(-5)	1.9(-3)	6.8(-5)	2.8(-2)	5.8(-12)	5.0(-9)	4.7(-7)	5.0(-9)	7.1(-6)	4.9(-3)
Mo-99	2.5(-1)	9.4(+0)	3.3(-1)	8.8(+1)	2.9(-8)	2.5(-5)	2.2(-3)	2.3(-5)	9.6(-4)	1.5(+1)
I-131	1.9(-1)	7.2(+0)	2.5(-1)	9.1(+1)	2.2(-7)	1.9(-4)	1.7(-2)	1.8(-4)	2.2(-2)	1.6(+1)
Te-132	2.0(-2)	7.5(-1)	2.6(-2)	7.5(+0)	2.3(-9)	2.0(-6)	1.8(-4)	1.9(-6)	9.3(-5)	1.3(+0)
I-132	6.8(-2)	7.5(-1)	2.6(-2)	7.5(+0)	6.8(-8)	5.8(-5)	1.8(-4)	1.9(-6)	9.3(-5)	1.5(+1)
I-133(1)										
I-134(1)										
Cs-134	3.3(-2)	1.3(+0)	4.4(-2)	1.9(+1)	4.0(-9)	3.4(-6)	3.2(-4)	3.4(-6)	9.1(-3)	3.3(+0)
I-135(1)										
Cs-136	1.7(-2)	6.4(-1)	2.2(-2)	8.7(+0)	2.0(-9)	1.7(-6)	1.6(-4)	1.7(-6)	3.3(-4)	1.5(+0)
Cs-137	1.7(-1)	6.5(+0)	2.2(-1)	9.7(+1)	2.0(-8)	1.7(-5)	1.6(-3)	1.7(-5)	5.1(-2)	1.7(+1)
Ba-140	3.4(-4)	1.3(-2)	4.5(-4)	1.7(-1)	4.0(-11)	3.4(-8)	3.2(-6)	3.4(-8)	6.5(-6)	3.0(-2)
La-140	1.7(-4)	4.3(-3)	1.4(-4)	1.2(-1)	1.3(-11)	1.1(-8)	1.3(-6)	1.4(-8)	6.9(-6)	2.1(-2)
Ce-144	3.3(-5)	1.3(-3)	4.4(-5)	1.9(-2)	4.0(-12)	3.4(-9)	3.2(-7)	3.4(-9)	7.4(-6)	3.3(-3)
Xe-131m					4.0(-12)					
Xe-133m					1.3(-10)					

Isotope	Feed (µCi/gm)	Floor Drain Tank (Ci)	Floor Drain Tank Pump (µCi/gm)	Waste Evap Bottom (Ci)	Waste Evap Distillate (µCi/cc)	Distillate Condensate (µCi/gm)	Waste Test Tank (Ci)	Waste Test Tank Concentration (µCi/gm)	Waste Demin. (Ci)	Evap. Bottom Pump (µCi/gm)
Xe-133					2.3(-9)					
Xe-135m					1.8(-7)					
Xe-135					1.2(-8)					
TOTAL		2.7(+1)	9.2(-1)	3.2(+2)	5.4(-7)	3.0(-4)	2.2(-2)	2.3(-4)	8.4(-2)	6.9(+1)

Note: The information presented in the above Table represents the radiation sources and associated input parameters, assumptions and methodology described in Section 12.2 used for the original shielding design and is retained here for historical purposes. Updating this analysis using an analyzed reactor power of 3659 MWt and operation with an 18-month fuel cycle represents a minor change in the original design basis and will have no significant impact on shielding requirements and safe plant operations.

TABLE 12.2-25 GEOMETRY OF EQUIPMENT IN THE LIQUID WASTE SYSTEM

Floor Drain Tank

Diameter	11 ft
Height	15 ft 1 in.

Floor Drain Tank Pumps

Diameter	2½ in.
Height	13 in.

Waste Evaporator	Lower Part	Upper Part	Vapor Space
Diameter	2 ft 6 in.	5 ft 6 in.	5 ft 6 in.
Height	13 ft 6 in.	3 ft 3 in.	8 ft 9 in.

Waste Evaporator Distillate Condenser

Diameter	2 ft 2 in.
Height	6 ft

Waste Evaporator Distillate Accumulator

Diameter	2 ft 2 in.
Height	11 ft

Waste Evaporator Distillate Pump

Diameter	2½ in.
Height	13 in.

Waste Evaporator Distillate Cooler

Diameter	1 ft 9 in.
Height	10 ft

Waste Test Tank (WTT)	
Diameter	11 ft
Height	35 ft
WTT Pumps	
Diameter	2½ in.
Height	13 in.
Waste Demineralizer	
Diameter	4 ft
Height	6 ft
Waste Evaporator Reboiler	
Diameter	36 in.
Height	14 ft
Volume Source Fluid	31 ft^3
Waste Evaporator Reboiler Pump	
Volume	20 ft^3
Diameter	24 in.
Waste Evaporator Bottoms Pump	
Diameter	2½ in.
Height	13 in.

Waste Evaporator Bottoms Cooler
Straight Section (Qty-12)

Length	10 ft
Diameter	1½ in.
Volume of Source Fluid	150 in.

Curved Section (Qty-11)

Diameter	1¼ in.
Radius of Curve	4½ in.

TABLE 12.2-26 RADIONUCLIDE CONCENTRATIONS IN WASTE GAS STREAMS [HISTORICAL]

	Concentration (μCi/cc)		
ISOTOPE	LTDN DEGASIFIER	PDT DEGASIFIER	RGWS INPUT
Kr-83m	2.4+1**	2.4+1*	2.4+1*
Kr-85m	1.0+2	1.0+2	1.0+2
Kr-85	8.4+0	8.4+0	8.4+0
Kr-87	6.6+1	6.6+1	6.6+1
Kr-88	2.0+2	1.0+2	1.0+2
Xe-131m	4.3+0	4.3+0	4.3+0
Xe-133m	3.6+1	3.6+1	3.6+1
Xe-133	1.6+3	1.6+3	1.6+3
Xe-135m	2.4+1	2.4+1	2.4+1
Xe-135	1.9+2	1.9+2	1.9+2
Xe-137	1.8+0	1.8+0	1.8+0
Xe-138	1.9+1	1.9+1	1.9+1
I-131	1.2-1	1.2+0	4.8-1
I-132	3.8-2	3.8-1	1.5-1
I-133	1.9-1	1.9+0	7.6-1
I-134	2.0-2	2.0-1	8.0-2
I-135	1.0-1	1.0+0	4.0-1

Note: The information presented in the above Table represents the radiation sources and associated input parameters, assumptions and methodology described in Section 12.2 used for the original shielding design and is retained here for historical purposes. Updating this analysis using an analyzed reactor power of 3659 MWt and operation with an 18-month fuel cycle represents a minor change in the original design basis and will have no significant impact on shielding requirements and safe plant operations.

* 2.4+1 = 2.4×10^1

TABLE 12.2-27 RADIOACTIVE SOURCE TERMS FOR GAS WASTE SYSTEM [HISTORICAL]

Radionuclide Inventory *(Ci)

Isotope Filter	I-Guard Bed	Chiller	Dryer	H_2 Surge Tank	Hepa
Kr-83m	6.0-3	5.5-2	5.0-1	-	-
Kr-85m	2.5-2	2.3-1	2.1+0	-	-
Kr-85	2.1-3	1.9-2	1.7-1	1.4+2	2.3+0
Kr-87	1.7-2	1.5-1	1.4+0	-	-
Kr-88	4.5-2	4.4-1	4.0+0	-	-
Xe-131m	1.1-3	9.9-3	9.0-2	2.2+0	7.9-1
Xe-133m	9.0-3	8.2-2	7.5-1	-	-
Xe-133	4.0-1	3.8+0	3.4+1	1.1+1	5.8+0
Xe-135m	6.0-3	5.5-2	5.0-1	-	-
Xe-135	4.8-2	4.4-1	4.0+0	-	-
Xe-137	4.5-4	4.1-3	3.7-2	-	-
Xe-138	4.8-3	4.4-2	4.0-1	-	-
I-131	4.5+1	2.7-4	2.5-3	-	-
I-132	1.7-1	8.5-5	7.8-4	-	-
I-133	7.8+0	4.4-4	4.0-3	-	-
I-134	3.3-2	4.4-5	4.0-4	-	-
I-135	1.3+0	2.3-4	2.1-3	-	-

Note: The information presented in the above Table represents the radiation sources and associated input parameters, assumptions and methodology described in Section 12.2 used for the original shielding design and is retained here for historical purposes. Updating this analysis using an analyzed reactor power of 3659 MWt and operation with an 18-month fuel cycle represents a minor change in the original design basis and will have no significant impact on shielding requirements and safe plant operations.

* Source Terms apply to a single component.

Radionuclide Inventory (Ci)

Isotope	1st Bed	2nd Bed	3rd Bed	4th Bed	5th Bed
Kr-83m	8.2+1	1.5-1	--	--	--
Kr-85m	7.5+2	5.2+1	3.5+0	2.4-1	1.7-2
Kr-85	1.8+2	1.8+2	1.8+2	1.8+2	1.8+2
Kr-87	1.5+2	1.4-2	<1.0-2	--	--
Kr-88	1.0+3	1.5+1	2.2-1	<1.0-2	--
Xe-131m	1.1+3	5.7+2	2.8+2	1.4+2	7.0+1
Xe-133m	3.5+3	8.8+1	2.2+0	5.6-2	<1.0-2
Xe-133	3.0+5	6.1+4	1.3+4	2.6+3	5.4+2
Xe-135m	1.2+1	--	--	--	--
Xe-135	3.2+3	--	--	--	--
Xe-137	2.1-1	--	--	--	--
Xe-138	8.3+0	--	--	--	--
I-131	1.6+2	1.0+1	6.4-1	4.0-2	<1.0-2
I-132	5.9-1	3.7-2	<1.0-2	--	--
I-133	2.7+1	1.7+0	1.1-1	<1.0-2	--
I-134	1.2-1	<1.0-2	--	--	--
I-135	4.6+0	2.9-1	1.8-2	<1.0-2	--

Note: The information presented in the above Table represents the radiation sources and associated input parameters, assumptions and methodology described in Section 12.2 used for the original shielding design and is retained here for historical purposes. Updating this analysis using an analyzed reactor power of 3659 MWt and operation with an 18-month fuel cycle represents a minor change in the original design basis and will have no significant impact on shielding requirements and safe plant operations.

TABLE 12.2-28 GEOMETRY OF EQUIPMENT IN RADIOACTIVE WASTE GAS SYSTEM

Waste Gas Chiller	
Diameter	3 in.
Length	34 in.
Waste Gas Dryer	
Diameter	6 in.
Height	53 in.
Iodine Guard Bed	
Diameter	2 ¾ in.
Height	13 in.
Carbon Delay Bed	
Diameter	36 in.
Height	72 in.
Hydrogen Surge Tank	
Diameter	36 in.
Height	72 in.
Regenerative Compressor	
Diameter	24 in.
Height	6 in.
Gas Volume	85 $in.^3$
HEPA Filter	
Diameter	2 ¾ in.
Height	11½ in.

TABLE 12.2-29 SHIELDING SOURCE TERMS FOR EQUIPMENT IN SOLID WASTE MANAGEMENT SYSTEM [HISTORICAL]

Nuclide	Shielding Source Terms (μCi/gm)				
	(a)	(b)	(c)	(d)	(e)
I-131	1.2+01*	1.0+01	6.5+02	2.2+03	3.5+01
I-132	4.7-01	4.1-01	2.9+01	9.7+01	1.4+00
Sr-89	3.0-02	2.6-03	4.9+00	1.6+01	8.8-02
Sr-90	2.0-03	1.7-03	1.1+00	3.7+00	5.8-03
Y-90	2.1-03	1.8-03	1.1+00	3.7+00	6.1-03
Y-91	4.4-02	3.8-02	1.1+00	3.7+00	1.3-01
Zr-95	5.3-03	4.6-03	1.3+00	4.3+00	1.5-02
Nb-95	6.7-03	5.8-03	7.1-01	2.4+00	2.0-02
Mo-99	5.4+00	4.7+00	7.1+01	2.4+02	1.6+01
Cs-134	5.9+00	5.1+00	2.6+02	8.7+02	1.7+01
Cs-136	1.1+00	9.5-01	9.2+00	3.1+01	3.2+00
Cs-137	3.2+00	2.8+00	1.5+02	5.0+02	9.3+00
Te-132	4.7-01	4.1-01	2.9+01	9.7+01	1.4+00
Ba-140	1.6-02	1.4-02	1.7+00	5.7+00	4.7-02
La-140	1.4-02	1.2-02	1.7+00	5.7+00	4.1-02
Ce-144	4.4-03	3.8-03	2.0+00	6.7+00	1.3-02
Mn-54	8.0-03	6.9-03	3.6+00	1.2+01	2.3-02
Co-58	2.1-01	1.8-01	5.4+01	1.8+02	6.1-01
Co-60	8.5-03	7.4-03	4.6+00	1.5+01	2.5-02
Fe-59	7.5-03	6.5-03	1.5+00	5.0+00	2.2-02
Cr-51	5.3-03	4.6-03	8.1-01	2.7+00	1.5-02

(a) Waste Concentrates Tank and Waste Conc. Transfer Pump
(b) Waste Feed Tanks and Waste Feed Recir. Pump
(c) Spent Resin Transfer Pump, Spent Resin Hopper, Spent Resin Recir. Pump, and Resin Centrifuge Metering Pump
(d) Spent Resin Centrifuge
(e) Waste Crystallizer/Evaporator, Crystallizer Recir. Pump, and Crystallizer Drain Pump
* $1.2+01 = 1.2 \times 10^1$

Nuclide	Shielding Source Terms (μCi/gm)				
	(f)	(g)	(h)	(i)	(j)
I-131	3.5+01	1.0-04	1.6-01	1.0-04	1.1+03
I-132	1.4+00	3.9-06	7.3-03	3.9-06	4.9+01
Sr-89	8.8-02	2.5-07	1.3-03	2.5-07	8.0+00
Sr-90	5.8-03	1.7-08	2.7-04	1.7-08	1.9+00
Y-90	6.1-03	1.8-08	2.7-04	1.8-08	1.9+00
Y-91	1.3-01	3.7-07	2.7-04	3.7-07	1.9+00
Zr-95	1.5-02	4.4-08	3.3-04	4.4-08	2.2+00
Nb-95	2.0-02	5.6-08	1.8-04	5.6-08	1.2+00
Mo-99	1.6+01	4.5-05	1.8-02	4.5-05	1.2+02
Cs-134	1.7+01	4.9-05	6.5-02	4.9-05	4.4+02
Cs-136	3.2+00	9.2-06	2.4-03	9.2-06	1.6+01
Cs-137	9.3+00	2.7-05	3.7-02	2.7-05	2.5+02
Te-132	1.4+00	3.9-06	7.3-03	3.9-06	4.9+01
Ba-140	4.7-02	1.3-07	4.3-04	1.3-07	2.9+00
La-140	4.1-02	1.2-07	4.3-04	1.2-07	2.9+00
Ce-144	1.3-02	3.7-08	5.0-04	3.7-08	3.4+00
Mn-54	2.3-02	6.7-08	9.0-04	6.7-08	6.0+00
Co-58	6.1-01	1.8-06	1.4-02	1.8-06	9.0+01
Co-60	2.5-02	7.1-08	1.2-03	7.1-08	7.5+00
Fe-59	2.2-02	6.3-08	3.8-04	6.3-08	2.5+00
Cr-51	1.5-02	4.4-08	2.1-04	4.4-08	1.4+00

Note: The information presented in the above Table represents the radiation sources and associated input parameters, assumptions and methodology described in Section 12.2 used for the original shielding design and is retained here for historical purposes. Updating this analysis using an analyzed reactor power of 3659 MWt and operation with an 18-month fuel cycle represents a minor change in the original design basis and will have no significant impact on shielding requirements and safe plant operations.

(f) Conc. Bottom Tank, Conc. Bottom Tank Recir. Pump, Waste Metering Pump, and Alternate Station Conc. Feed Pump

(g) Entrainment Separator, Crystallizer Condenser (Shell Side), Crystallizer Dist. Tank, Crystallizer Dist. Pump, and Crystallizer Subcooler (Shell Side)

(h) Spent Resin Dewatering Pump

(i) Crystallizer Reflux Pot and Crystallizer Reflux Pump

(j) Extruder

TABLE 12.2-30 GEOMETRY OF EQUIPMENT IN SOLID WASTE MANAGEMENT SYSTEM

<u>Waste Concentrates Tank</u>	
Diameter	10 ft
Height	11 ft 9 in.
<u>Waste Concentrates Transfer Pump</u>	
Diameter (internal)	4 in.
Length	17 in.
<u>Waste Feed Tank(s)</u>	
Diameter	5 ft
Height (including heads)	9 ft 10 in.
<u>Waste Feed Recirculation Pump(s)</u>	
Diameter (internal)	2 in.
Length	1 ft 6 in.
<u>Spent Resin Transfer Pump</u>	
Diameter (internal)	2 in.
Length	6 ft
<u>Spent Resin Hopper</u>	
Diameter	5 ft
Height (cylinder)	6 ft
Height (cone)	3 ft

Spent Resin Recirculation Pump

Diameter	3 in.
Length	3 ft 9 in.

Resin Centrifuge Metering Pump

Diameter (internal)	2 in.
Length	1 ft 9 in.

Spent Resin Centrifuge

Diameter of bowl	1 ft
Depth of bowl	1 ft 3 in.

Evaporator/Crystallizer Vapor Body

Diameter	3 ft
Height (cylinder)	9 ft 6 in.
Height (cone)	2 ft 9 in.

Crystallizer Recirculation Pump

Diameter (internal)	10 in.
Length	6 ft

Crystallizer Drain Pump

Diameter (internal)	10 in.
Length	3 ft

Concentrates Bottom Tank

Diameter	5 ft
Height	8 ft 6 in.

Concentrates Bottom Tank Recirculation Pump

Diameter (internal)	10 in.
Length	3 ft 6 in.

Waste Metering Pump

Diameter (internal)	3/4 in.
Length	1 ft 6 in.

Alternate Station Concentrates Feed Pump

Diameter (internal)	20 in.
Length	4 ft 6 in.

Entrainment Separator

Diameter	2 ft
Height (including heads)	12 ft 6 in.

Crystallizer Condenser

Diameter	13 in.
Height	11 ft 6 in.

Crystallizer Subcooler

Diameter	6 in.
Height	11 ft 6 in.

Spent Resin Dewatering Pump

Diameter (internal)	1 in.
Length	1 ft 3 in.

<u>Crystallizer Reflux Pot</u>

Diameter	1 ft 6 in.
Height (head to head)	29 in.

<u>Crystallizer Reflux Pump</u>

Diameter (internal)	1 in.
Length	1 ft

<u>Evaporator/Extruder</u>

Diameter (internal)	2 in.
Length	15 ft 9 in.

TABLE 12.2-31 CONTAINMENT AIRBORNE ISOTOPIC CONCENTRATIONS AT SHUTDOWN [HISTORICAL]

Volume = 2.70×10^6 ft^3

Ventilation Rate = 1000 cfm, continuous prior to shutdown

Isotope	Specific Activity (µCi/cc)	MPC Fraction
H-3	4.02-04*	8.03+01
Kr-85m	3.09-06	5.15-01
Kr-85	2.65-06	2.65-01
Kr-87	7.30-07	7.30-01
Kr-88	4.01-06	4.01+00
Xe-131m	9.22-06	6.08-02
Xe-133m	6.19-06	6.19-01
Xe-133	3.76-04	3.76+01
Xe-135m	1.82-06	1.82+00
Xe-135	1.55-05	3.88+00
Xe-138	7.63-08	7.62-02
I-131	4.11-08	4.57+00
I-132	8.98-10	4.49-03
I-133	2.55-09	8.51-02
I-134	2.30-10	4.61-04
I-135	5.77-09	5.76-02
TOTAL	8.13-04	1.35+02

Note: The information presented in the above Table represents the radiation sources and associated input parameters, assumptions and methodology described in Section 12.2 used for the original shielding design and is retained here for historical purposes. Updating this analysis using an analyzed reactor power of 3659 MWt and operation with an 18-month fuel cycle represents a minor change in the original design basis and will have no significant impact on shielding requirements and safe plant operations.

* 4.02-04 = 4.02×10^{-4}

TABLE 12.2-32 CONTAINMENT AIRBORNE ISOTOPIC CONCENTRATIONS 20 HOURS AFTER SHUTDOWN [HISTORICAL]

Volume	=	2.70×10^6 ft^3
Ventilation Rate	=	1000 cfm, continuous prior to shutdown
Ventilation Rate	=	11,000 cfm, for 17 hours following shutdown, then 40,000 cfm, after 17 hours following shutdown
Recirculation Rate	=	4000 cfm, for 16 hours, with 90% removal of iodine species

Isotope	Specific Activity (μCi/cc)	*MPC Fraction*
H-3	3.41-06*	6.81-01
Kr-85m	1.12-09	1.87-04
Kr-85	2.25-08	2.25-03
Kr-87	1.09-13	1.09-07
Kr-88	2.36-10	2.36-04
Xe-131m	1.02-08	2.53-05
Xe-133m	4.10-10	4.10-03
Xe-133	2.87-06	2.87-01
Xe-135m	4.48-08	4.48-02
Xe-135	6.75-08	1.69-02
Xe-138	0.	0.
I-131	1.33-10	1.47-02
I-132	7.19-15	3.60-08
I-133	4.54-12	1.52-04
I-134	0.	0.
I-135	2.53-12	2.53-05
TOTAL	6.43-06	1.05+00

Note: The information presented in the above Table represents the radiation sources and associated input parameters, assumptions and methodology described in Section 12.2 used for the original shielding design and is retained here for historical purposes. Updating this analysis using an analyzed reactor power of 3659 MWt and operation with an 18-month fuel cycle represents a minor change in the original design basis and will have no significant impact on shielding requirements and safe plant operations.

* 3.41-06 = 3.41×10^{-6}

TABLE 12.2-33 TURBINE BUILDING AIRBORNE ISOTOPIC CONCENTRATIONS – SUMMER [HISTORICAL]

Volume = 7.72×10^6 ft^3

Ventilation Rate = 1.00×10^6 cfm

Isotope	Specific Activity (μCi/cc)	*MPC Fraction*
Kr-85m	1.26-13*	2.11-08
Kr-85	9.74-15	9.73-10
Kr-87	8.91-14	8.90-08
Kr-88	2.43-13	2.43-07
Xe-131m	5.07-15	2.53-10
Xe-133m	4.15-14	4.15-09
Xe-133	1.88-12	1.88-07
Xe-135m	7.78-14	7.79-08
Xe-135	2.35-13	5.87-08
Xe-138	3.59-14	3.59-08
I-131	1.04-12	1.15-04
I-132	6.79-14	3.39-07
I-133	1.16-12	3.87-05
I-134	1.67-14	3.31-08
I-135	3.70-13	3.70-06
TOTAL	5.41-12	1.59-04

Note: The information presented in the above Table represents the radiation sources and associated input parameters, assumptions and methodology described in Section 12.2 used for the original shielding design and is retained here for historical purposes. Updating this analysis using an analyzed reactor power of 3659 MWt and operation with an 18-month fuel cycle represents a minor change in the original design basis and will have no significant impact on shielding requirements and safe plant operations.

* 1.26-13 = 1.26×10^{-13}

TABLE 12.2-34 TURBINE BUILDING AIRBORNE ISOTOPIC CONCENTRATIONS – WINTER [HISTORICAL]

Volume = 7.18×10^6 ft^3

Ventilation Rate = 0 cfm

Isotope	Specific Activity (µCi/cc)	*MPC Fraction*
Kr-85m	4.82-12*	8.03-07
Kr-85	1.72-10	1.72-05
Kr-87	1.04-12	1.04-06
Kr-88	5.95-12	5.96-06
Xe-131m	2.24-11	1.12-06
Xe-133m	2.40-11	2.39-06
Xe-133	2.24-09	2.24-04
Xe-135m	6.55-12	6.54-06
Xe-135	3.96-11	9.89-06
Xe-138	1.03-13	1.08-07
I-131	1.70-09	1.89-01
I-132	1.38-12	6.88-06
I-133	2.05-13	6.84-03
I-134	1.38-13	2.77-07
I-135	2.13-11	2.13-04
TOTAL	4.44-09	1.96-01

Note: The information presented in the above Table represents the radiation sources and associated input parameters, assumptions and methodology described in Section 12.2 used for the original shielding design and is retained here for historical purposes. Updating this analysis using an analyzed reactor power of 3659 MWt and operation with an 18-month fuel cycle represents a minor change in the original design basis and will have no significant impact on shielding requirements and safe plant operations.

* 4.82-12 = 4.82×10^{-12}

TABLE 12.2-35 PRIMARY AUXILIARY BUILDING AIRBORNE ISOTOPIC CONCENTRATIONS [HISTORICAL]

Volume = 1.45×10^5 ft^3

Ventilation Rate = 41,500 cfm

Isotope	Specific Activity (μCi/cc)	MPC Fraction
Kr-85m	2.61-08*	4.35-03
Kr-85	2.03-09	2.03-04
Kr-87	1.92-08	1.92-02
Kr-88	5.12-08	5.12-02
Xe-131m	1.04-09	5.22-05
Xe-133m	8.60-09	8.59-04
Xe-133	3.87-07	3.87-02
Xe-135m	1.24-08	1.24-02
Xe-135	4.80-08	1.20-02
Xe-138	8.89-09	8.89-03
I-131	2.90-10	3.23-02
I-132	1.03-10	5.17-04
I-133	4.60-11	1.53-03
I-134	6.48-11	1.30-04
I-135	2.51-10	2.51-03
TOTAL	5.65-07	1.85-01

Note: The information presented in the above Table represents the radiation sources and associated input parameters, assumptions and methodology described in Section 12.2 used for the original shielding design and is retained here for historical purposes. Updating this analysis using an analyzed reactor power of 3659 MWt and operation with an 18-month fuel cycle represents a minor change in the original design basis and will have no significant impact on shielding requirements and safe plant operations.

* 2.61-08 = 2.61×10^{-8}

TABLE 12.2-36 REFUELING AIRBORNE TRITIUM CONCENTRATION CONTAINMENT REFUELING POOL [HISTORICAL]

Volume = 2.70×10^6 ft^3

Ventilation Rate = 31,000 cfm

Isotope	Specific Activity (μCi/cc)	MPC Fraction
H-3	2.09-06	4.19-01

Note: The information presented in the above Table represents the radiation sources and associated input parameters, assumptions and methodology described in Section 12.2 used for the original shielding design and is retained here for historical purposes. Updating this analysis using an analyzed reactor power of 3659 MWt and operation with an 18-month fuel cycle represents a minor change in the original design basis and will have no significant impact on shielding requirements and safe plant operations.

TABLE 12.2-37 REFUELING AIRBORNE TRITIUM CONCENTRATION SPENT FUEL POOL [HISTORICAL]

Volume = 263,000 ft^3

Ventilation Rate = 34,000 cfm

Isotope	Specific Activity (μCi/cc)	*MPC Fraction*
H-3	2.67-06	5.35-01

Note: The information presented in the above Table represents the radiation sources and associated input parameters, assumptions and methodology described in Section 12.2 used for the original shielding design and is retained here for historical purposes. Updating this analysis using an analyzed reactor power of 3659 MWt and operation with an 18-month fuel cycle represents a minor change in the original design basis and will have no significant impact on shielding requirements and safe plant operations.

TABLE 12.3-1 RADIATION ZONING AND ACCESS CONTROL

Zone	Dose Rate Mrem/Hour	Allowed Occupancy
I	Less than 0.5	No Restriction on Access
II	0.5 - 2.5	Occupational Access
III	2.5 – 15	Periodic Access
IV	15 – 100	Limited Access
V	Greater than 100	Restricted Access

TABLE 12.3-2 RADIATION ZONE AREAS

Location	Maximum Dose Rate (mrem/hr)
Normal Operation	
Site Boundary	0.0005
Reactor Building Interior	
During Operation (Below Elev. 25'-0")	15
(Above Elev. 25'-0")	150
Certain Equipment Areas in Auxiliary Building	15
Fuel Handling Areas	15
Auxiliary Building Corridors, Cable Room, Local Control Panels, Equipment Room, Containment Operating Floor During Shutdown	2.5
Control Room	0.5
Turbine Building	0.5
Administration Building	0.5
Radiation Counting Room	0.5
Accident Condition	
Inside Control Room following DBA	5 rem integrated whole body dose over duration of accident

TABLE 12.3-3 COMPONENT SHIELDING THICKNESS

Reference Figure: 12.3-1

Bldg: Containment Elevation: (-) 26'-0"

COMPONENT	CONCRETE SHIELD THICKNESS (IN)					
	WALLS					
	NORTH	EAST	SOUTH	WEST	CEILING	FLOOR
Reactor Coolant Drain Tank	18	36	24	48	24	-
Excess Letdown Heat Exchanger	48	42	30	32	24	-
Regenerative Heat Exchanger	36	36	36	36	24*	-
Reactor Coolant Drain Tank Heat Exch.	12	12	12	36	24	-

* Plus 4 inches of lead plate

TABLE 12.3-4 COMPONENT SHIELDING THICKNESS

Reference Figure: 12.3-4

Bldg: Vaults Elevation: (-) 61'-0" & above

COMPONENT	CONCRETE SHIELD THICKNESS (IN)					
	WALLS				CEILING	FLOOR
	NORTH	EAST	SOUTH	WEST		
RHR Heat Exch. - A	48	30	30	30		30
RHR Pump - A	48	30	30	30	30	48
RHR Heat Exch. - B	30	30	36	30	24	30
RHR Pump – B	30	30	48	30	30	48

TABLE 12.3-5 COMPONENT SHIELDING THICKNESS

Reference Figure: 12.3-5

Bldg: PAB Elevation: 2'-0"

COMPONENT	CONCRETE SHIELD THICKNESS (IN)					
	WALLS					
	NORTH	EAST	SOUTH	WEST	CEILING	FLOOR
BTR Demin – A	36	36	24	24	48	48
BTR Demin - B, C	24	36	24	24	48	48
BTR Demin - D, E	24	30	24	24	48	48
Cation Demin	24	42	36	30	48	48
Mixed Bed Demins	36	42	36	36	48	48
Spent Fuel Pool Demin	36	30	30	24	48	48
Letdown Heat Exch.	30	24	24	30	36	48
Letdown Reheat Hx	24	24	24	30	36	48
Letdown Chiller Hx	24	24	24	36	36	48
Moderating Heat Exch	24	24	24	36	36	48
Seal Water Hx	24	24	24	24	36	48
Seal Water Return Fltr	24	24	24	24	36	48
Seal Injection Filter	24	24	24	24	36	48
Reactor Coolant Fltr	24	24	24	36	36	48
Demin Pre-Filter	24	24	24	36	36	48
Fuel Pool Post-Filter	24	24	24	24	36	48
Fuel Pool Pre-Filter	24	24	24	24	36	48

TABLE 12.3-6 COMPONENT SHIELDING THICKNESS

Reference Figure: 12.3-5

Bldg: PAB Elevation: 7'-0"

COMPONENT	CONCRETE SHIELD THICKNESS (IN)					
	WALLS				CEILING	FLOOR
	NORTH	EAST	SOUTH	WEST		
Cent. Charging Pump – A	33	24	24	0 (15)*	36	36
Cent. Charging Pump – B	24	24	24	0 (15)	36	36
Recip. Charging Pump	24	24	30	0 (15)	36	36
Letdown Degasifier (Liquid)	30	30	30	24	--	48
Letdown Degasifier Pumps	30	24	30	30	24	48

* (n) - Denotes "n" feet of air available for additional shielding.

TABLE 12.3-7 COMPONENT SHIELDING THICKNESS

Reference Figure: 12.3-6

Bldg: PAB Elevation: 25'-0"

COMPONENT	CONCRETE SHIELD THICKNESS (IN)					
	WALLS				CEILING	FLOOR
	NORTH	EAST	SOUTH	WEST		
Letdown Degasifier (Vapor)	42	48	42	42	30	--
Sample Heat Exch's	36	24	24	24	48	24

TABLE 12.3-8 COMPONENT SHIELDING THICKNESS

Reference Figure: 12.3-7

Bldg: PAB Elevation: 53'-0"

COMPONENT	CONCRETE SHIELD THICKNESS (IN)					
	WALLS					
	NORTH	EAST	SOUTH	WEST	CEILING	FLOOR
Volume Control Tank	48	48	48	48	48	48

TABLE 12.3-9 COMPONENT SHIELDING THICKNESS

Reference Figure: 12.3-8

Bldg: WPB Elevation: (-)31'-0"

COMPONENT	CONCRETE SHIELD THICKNESS (IN)					
	WALLS				CEILING	FLOOR
	NORTH	EAST	SOUTH	WEST		
Resin Sluice Tanks	36	36	30	42	30	48
Resin Transfer Pump	30	24	36	42	24	48
Floor Drain Tanks	36	24	30	42	24	48
Floor Drain Tank Pumps	36	12	30	24	21	48
Waste Concentrates Tank	30	30	24	30	24	48
Waste Concentrates Transfer Pump	24	30	24	24	24	48
Chemical Drain Treatment Transfer Pump	24	30	24	0 (10)*	30	48
Chemical Drain Treatment Tanks	24	30	24	30	24	48

* (n) - Denotes "n" feet of air available for additional shielding

TABLE 12.3-10 COMPONENT SHIELDING THICKNESS

Reference Figure: 12.3-9

Bldg: WPB Elevation: (-)3'-0"

COMPONENT	CONCRETE SHIELD THICKNESS (IN)					
	WALLS					
	NORTH	EAST	SOUTH	WEST	CEILING	FLOOR
Cesium Removal Ion Exchanger – A	36	30	24	36	30	30
Cesium Removal Ion Exchanger – B	24	30	24	36	30	30
Recovery Demineralizers	24	30	24	30	30	30
Waste Demineralizer	24	30	24	30	30	30
Recovery Filter – A	36	30	24	24	30	30
Recovery Filter - B	24	30	24	24	30	30
Recovery Evaporator Filter – A	24	30	24	24	30	30
Recovery Evaporator Filter – B	24	30	66	24	30	30
Recovery Demineralizer Filter – A	66	30	24	24	30	30
Recovery Demineralizer Filter – B	24	30	24	24	30	30
Floor Drain Filter – A	24	30	24	24	30	30
Floor Drain Filter – B	24	30	66	24	30	30
Waste Demineralizer Filter	66	30	24	24	30	30
Primary Drain Tank Degasifier Filter	24	30	48	24	30	30

COMPONENT	CONCRETE SHIELD THICKNESS (IN)					
	WALLS					
	NORTH	EAST	SOUTH	WEST	CEILING	FLOOR
Resin Sluice Filter	30	24	12	36	24	24
Primary Drain Tanks	36	24	30	36	30	30
Primary Drain Tank Transfer Pumps	36	18	30	24	21	21
Primary Drain Tank Degasifier Recirc Pumps	24	24	0(7)*	48	24	24
Waste Evaporator Reboiler	12	24	24	24	24	24
Waste Evaporator Bottoms Package	24	24	12	24	24	24
Boron Waste Storage Tank – A	30	24	24	30	24	24
Boron Waste Storage Tank – B	30	30	24	24	24	24
Recovery Evaporator Feed Pumps	24	24	24	24	24	24
Recovery Evaporator Reboilers	12	24	24	24	24	24
Recovery Evaporator Bottoms Packages	24	24	12	24	24	24

* (n) - Denotes "n" feet of air available for additional shielding

TABLE 12.3-11 COMPONENT SHIELDING THICKNESS

Reference Figure: 12.3-10

Bldg: WPB Elevation: 25'-0"

COMPONENT	CONCRETE SHIELD THICKNESS (IN)					
	WALLS				CEILING	FLOOR
	NORTH	EAST	SOUTH	WEST		
Primary Drain Tank Degasifier (Lower Half)	24	24	34	42	--	24
Primary Drain Tank Degasifier (Upper Half)	24	24	36	42	36	--
Refueling Water Storage Tank	24	24	24	24	--	--

TABLE 12.3-12 COMPONENT SHIELDING THICKNESS

Reference Figure: 12.3-11

Bldg: WPB Elevation: 42'-5"

COMPONENT	CONCRETE SHIELD THICKNESS (IN)					
	WALLS					
	NORTH	EAST	SOUTH	WEST	CEILING	FLOOR
Crystallizer Recirc. Pump	18	12	30	30	24	24
Crystallizer Drain Pump	18	12	30	30	24	24
Crystallizer Reflux Pump	18	12	30	30	24	24
Evaporator/Extruder	*	30	18	18	24	18/24

* 2" lead brick (as needed)

TABLE 12.3-13 COMPONENT SHIELDING THICKNESS

Reference Figure: 12.3-12, 12.3-13

Bldg: WPB Elevation: 53'-0"

COMPONENT	CONCRETE SHIELD THICKNESS (IN)					
	WALLS					
	NORTH	EAST	SOUTH	WEST	CEILING	FLOOR
Waste Hopper	36	36	36	30	30	30
Waste Gas Chiller-Dryer	24	36	24	24	24*	24
Carbon Delay Bed – A	48	36	36	36	36	36
Carbon Delay Bed – B	48	48	48	36	36	36
Carbon Delay Beds - C,D,E	42	30	30	30	30	24
Hydrogen Surge Tank	42	30	30	30	30	24
Iodine Guard Beds	24	36	36	36	30	24
Waste Feed Tanks	30	30	30	36	12	24
Waste Feed Recirc. Pumps	30	30	12	36	12	24
Spent Resin Recirc. Pump	36	30	30	36	12	24
Resin Centrifuge Metering Pump	24	36	18	12	12	24
Spent Resin Centrifuge	12	12	12	24	12	24
Evaporator/Crystallizer Vapor Body	12	30	30	24	15	24
Concentrates Bottom Tank	30	30	30	36	12	24
Concentrates Bottom Tank Recirc. Pump	30	30	12	36	12	24

* Plus 1" Steel Plate

COMPONENT	CONCRETE SHIELD THICKNESS (IN)					
	WALLS					
	NORTH	EAST	SOUTH	WEST	CEILING	FLOOR
Waste Metering Pump	24	36	18	12	12	24
Alternate Station Concentrates Feed Pump	30	30	12	36	12	24
Entrainment Separator	12	30	30	24	15	24
Crystallizer Condenser	30	30	12	36	12	24
Crystallizer Subcooler	30	30	12	36	12	24
Spent Resin Dewatering Pump	36	30	30	36	12	24
Crystallizer Reflux Pot	12	30	30	24	15	24

TABLE 12.3-14 AREA RADIATION MONITORS

INSTRUMENT TAG NO. RE-	DESCRIPTION	DETECTOR TYPE	DETECTOR BACK-GRD. mr/hr	RANGE LOW-HIGH mr/hr	(Note 5) ALARM SET POINT mr/hr	DETECTOR QTY.	IEEE CLASS	UFSAR FIGURE REFERENCE
	Containment Structure							
6534	In-Core Instrument Seal Table	GM (Note 4)	15	10^{-1}-10^{4}		1	Non 1E	12.3-2
6535A, B (Note 1)	Manipulator Crane	GM (Note 4)	15	10^{-1}-10^{4}		2	1E	12.3-3
6536-1, 2	Personnel Hatch (Post-LOCA)	Ion Chamber	2.5	10^{+1}-10^{9}		2	Non 1E	12.3-3
6576A, B	Containment (Post-LOCA)	Ion Chamber	25	10^{0}-10^{8} r/hr		2	1E	12.3-3
	Primary Auxiliary Building							
6537	Sampling Room	GM	2.5	10^{-1}-10^{4}		1	Non 1E	12.3-6
6538, 6539	RHR Pump Area	GM	>100	10^{-1}-10^{4}		2	Non 1E	12.3-4
6540	Volume Control Tank Area	Ion Chamber	$8x10^{4}$	10^{1}-10^{7}		1	Non 1E	12.3-7
6541	Lower Level	GM	2.5	10^{-1}-10^{4}		1	Non 1E	12.3-5
6543	Entrance	GM	>100	10^{-1}-10^{4}		1	Non 1E	12.3-5
6544	Entrance	GM	2.5	10^{-1}-10^{4}		1	Non 1E	12.3-6
6545, 6546 6547	Charging Pump Room	GM	110	10^{-1}-10^{4}		3	Non 1E	12.3-5

INSTRUMENT TAG NO. RE-	DESCRIPTION	DETECTOR TYPE	DETECTOR BACK-GRD. mr/hr	RANGE LOW-HIGH mr/hr	(Note 5) ALARM SET POINT mr/hr	DETECTOR QTY.	IEEE CLASS	UFSAR FIGURE REFERENCE
6508-1,2	PAB-HRAM	Ion Chamber	>100	10^{-2}-10^{4} r/hr		2	Non 1E	12.3-5
6563-1,2	PAB-HRAM	Ion Chamber	>100	10^{-2}-10^{4} r/hr		2	Non 1E	12.3-5
6517-1,2	RHR - Pump Vault HRAM	Ion Chamber	>100	10^{-2}-10^{4} r/hr		2	Non 1E	
	Fuel Storage Building							
6549	Spent Fuel Pool Area	GM	2.5	10-1-10^{4}		1	Non 1E	12.3-15
6518	Spent Fuel - HRAM	Ion Chamber	2.5	10^{-2}-10^{4} r/hr		1	Non 1E	12.3-15
	Control Room							
6550	Main Control Board Area	GM	0.5	10^{-1}-10^{4}		1	Non 1E	1.2-32
	Waste Processing Building							
6551	Waste Gas Processing Area	GM	2.5	10^{-1}-10^{4}		1	Non 1E	12.3-11
6552	Truck Loading Area	GM	1.5	10^{-1}-10^{4}		1	Non 1E	12.3-10
6553	Radwaste Control Room	GM	0.5	10^{-1}-10^{4}		1	Non 1E	12.3-10
6554	Waste Management Control Panel Area	GM	5	10^{-1}-10^{4}		1	Non 1E	12.3-10
6570	Extruder/Evaporator Manifold Area	GM	(Note 6)	10^{0}-10^{5}		1	Non 1E	12.3-10
6571	Compacted Rad Waste Storage Area	GM	2.5	10^{-1}-10^{4}		1	Non 1E	12.3-10

INSTRUMENT TAG NO. RE-	DESCRIPTION	DETECTOR TYPE	DETECTOR BACK-GRD. mr/hr	RANGE LOW-HIGH mr/hr	(Note 5) ALARM SET POINT mr/hr	DETECTOR QTY.	IEEE CLASS	UFSAR FIGURE REFERENCE
	Administration & Service Building							
6555	Hot Chemistry Laboratory	GM	0.5	10^{-1}-10^{4}		1	Non 1E	12.3-16
6556	Decontamination Room	GM	0.5	10^{-1}-10^{4}		1	Non 1E	12.3-16
6557	RCA Shop (Note 2)	GM	0.5	10^{-1}-10^{4}		1	Non 1E	12.3-16
6558	RCA Personnel Decontamination Area	GM	0.5	10^{-1}-10^{4}		1	Non 1E	12.3-16
6559	RCA Women's Locker Room	GM	0.5	10^{-1}-10^{4}		1	Non 1E	12.3-16

(Note 1) 6535-A and 6535-B will automatically terminate containment purge in the event of high radiation during fuel handling operations.

(Note 2) RCA – Radiologically Controlled Area.

(Note 3) Deleted.

(Note 4) GM – Geiger-Mueller.

(Note 5) Radiation monitoring setpoints are varied during operation to follow station conditions. Setpoints are maintained within the bounds established in the Technical Specifications. The methodology for establishing the setpoints is found in the station operating procedures.

(Note 6) >100

TABLE 12.3-15 AIRBORNE RADIATION MONITORS (SKID MOUNTED DETECTORS)

INSTRUMENT TAG NO. RE-	DESCRIPTION	DETECTOR TYPE	REFERENCE ISOTOPE	DETECTOR BACK-GRD. mr/hr	RANGE LOW-HIGH μCi/cc	(Note 5) ALARM SET POINT μCi/cc	DETECTOR QTY.	IEEE CLASS	ENERGY LEVEL	LOOP DIAG. 1-NHY	P&ID 1-NHY
6526-1	Containment Air Particulate (Note 4)	Beta	I^{131}, Cs^{137}	2.5	10^{-10}-10^{-6}		1	Non 1E	Note 3	506135	20504
6526-2	Containment Radiogas	Beta	Xe^{133}	2.5	10^{-6}-10^{-2}		1	Non 1E	Note 1	506135	20504
6528-1	Plant Vent – WRGM (Low Range)	Beta	Xe^{133}, Kr^{85}	2.5	10^{-7}-10^{-1}		1	Non 1E	Note 1	506607	20494
6528-2	Plant Vent – WRGM (Mid Range)	Beta	Xe^{133}, Kr^{85}	2.5	10^{-3}-10^{3}		1	Non 1E	Note 1	506607	20494
6528-3	Plant Vent – WRGM (Hi Range)	Beta	Xe^{133}, Kr^{85}	2.5	10^{-1}-10^{5}		1	Non 1E	Note 1	506607	20494
6495	WRGM Backup	Ion Chamber	Xe^{133}, Kr^{85}	2.5	10^{-1}-10^{7} mr/hr		1	Non 1E	Note 1	506607	RM-20509
6531-2	WPB Radiogas	Beta	Xe^{133}	2.5	10^{-7}-10^{-3}		1	Non 1E	Note 1	506885	20498
6532-2	PAB Radiogas	Beta	Xe^{133}	2.5	10^{-7}-10^{-3}		1	Non 1E	Note 1	506598	20510
6548	Containment Radiogas Backup	Beta	Xe^{133}	2.5	10^{-6}-10^{-2}		1	Non 1E	Note 1	506136	RM-20510

NOTES

Note	Isotopes	Max. Beta Energy (Mev)	Predominant Gamma Energy (Mev)
1	Xe^{133}	0.346	0.081
	Xe^{135}	0.92	0.249
	Kr^{85}	0.67	0.514
	Kr^{85m}	0.82	0.150
2	I^{131}	0.606	0.364
	I^{133}	1.27	0.53
	Cs^{134}	0.662	0.604
	Cs^{137}	0.514	0.662
	Co^{58}	0.474	0.81
	Co^{60}	0.314	1.17, 1.33
3	Same as Note 2, plus:		
	Rb^{88}	5.3	1.863

4 Containment air particulate monitor functions as a leakage detector and must survive the SSE (reference Regulatory Guide 1.45 and Standard Review Plan 5.2.5).

5 Rádiation monitoring setpoints are varied during operation to follow station operating conditions. Setpoints are maintained within the bounds established in the Technical Specifications. The methodology for establishing the setpoints is found in the station operating procedures.

TABLE 12.3-16 AIRBORNE RADIATION MONITORS (DETECTORS MOUNTED IN DUCT)

INSTRUMENT TAG NO. RE-	DESCRIPTION	DETECTOR TYPE	BACK-GRD. mr/hr	RANGE LOW-HIGH CPM	(Note 1) ALARM SET POINT CPM	DETECTOR QTY.	IEEE CLASS	LOOP DIAG.	P&ID	LOCATION 9763-F-
6506A1, A2, B1, B2 (1 RM)	CTL Rm East Air Intake	GM	0.5	10^1-10^6			1E	506151	NHY-CBA-20303	500210
6507A1, A2, B1, B2, (1 RM)	CTL Rm West Air Intake	GM	0.5	10^1-10^6		8	1E	506152	NHY-CBA-20303	500210
	Administration Building									
6523	Fume Hood Exh FN-115	GM	0.5	10^1-10^6		1	Non 1E	506668	NHY-AAH-20004	615002 & 11
6524	Fume Hood Exh FN-116	GM	0.5	10^1-10^6		1	Non 1E	506668	NHY-AAH-20004	615002 & 11
6525	Fume Hood Exh FN-117	GM	0.5	10^1-10^6		1	Non 1E	506668	NHY-AAH-20004	615002 & 11
	Fuel Storage Building									
6562	FAH-Fuel Stor Bldg Exh	GM	0.5	10^1-10^6		1	Non 1E	506452	NHY-MAH-20497	500178 500332
	Containment Enclosure									
6566	EAH-Contn Encl Emerg Exh	GM	0.5	10^1-10^6		1	Non 1E	506602	NHY-MAH-20495	500179
	Primary Auxiliary Building									
6567	PAB-Misc Ventilation	GM	2.5	10^1-10^6		1	Non 1E	506603	NHY-MAH-20494	500175 & 8
6568	PAB-Contn Enclosure	GM	2.5	10^1-10^6		1	Non 1E	506596	NHY-MAH-20494	500175 & 8

TABLE 12.4-1 ESTIMATED AVERAGE ANNUAL DOSE OF SEABROOK PERSONNEL

Activity	Total Dose (Man-Rem)	Percent of Total Dose
Reactor Operations and Surveillance	35.0	9
Routine Maintenance	63.0	17
In-Service Inspection	42.0	11
Special Maintenance	164.0	44
Waste Processing	28.0	8
Refueling	39.84	11
	371.84	100 percent

TABLE 12.4-2 OCCUPATIONAL DOSE ESTIMATES DURING ROUTINE OPERATIONS AND SURVEILLANCE

Activity	Radiation Field (R/Hr)	Exposure Time (Hrs/Year)[(6)]	Total Dose (Manrems/Year)
Walking In Radiation Zones	0.0005[(1)]	10,000	5.0
Control Room	0.0001[(2)]	25,000	2.5
Routine Surveillance Inside Containment	0.05[(3)]	100	5.0
Health Physics Surveys	0.0025[(4)]	3,600	9.0
Collection of Radioactive & Water Chemistry Samples	0.0025[(4)]	1,460	3.65
Radiochemistry Analysis	0.0012[(5)]	2,500	3.00
Routine Surveillance Primary Auxiliary Building	0.001[(2)]	3,000	3.00
Routine Surveillance Fuel Storage Building	0.005[(2)]	750	3.75
Miscellaneous			0.00
			34.9

Information Sources:

(1) The estimated occupancies of the different radiation zones provided in Table 12.3-1 indicate that approximately 75 percent of the total manhours of occupancy will be in Radiation Zone I areas. The average field was taken to be equal to the design maximum for Zone I areas. Since the average field in the Zone I areas is expected to be 20 percent or less of the design value, this conservatism takes into account the 25 percent of total time personnel spend in the radiation zones with more intense fields (II, III, and IV).

(2) Average fields assumed to be 20 percent of the maximum design fields for these areas. Radiation zones and justification for this assumption provided in Section 12.3.

(3) Estimated field consistent with other FSARs and the projected design range of exposures for these areas at Seabrook Station.

(4) Estimated occupancies of radiation zones by work function in Table 12.3-1 indicate that the majority of time spent in radiation zones by chemistry and health physics personnel is in Zone II areas. Therefore, the average exposure is conservatively assumed equal to the design maximum for Zone II.

(5) Information in appendix of "Technology Planning Study of Occupational Radiation Exposure in LWR Nuclear Power Plants," Electric Power Research Institute, January 33, 1980 (Reference 6) indicates that exposure for this activity is approximately half that generally encountered in the health physics surveys.

(6) Exposure time estimates based on engineering judgment.

TABLE 12.4-3 OCCUPATIONAL DOSE ESTIMATES DURING ROUTINE MAINTENANCE

Activity	Radiation[1] Field (R/Hr)	Exposure [2] Time (Hours/Year)	Total Dose[3] (Man-Rems/Year)
Polar Crane Maintenance	0.005	160	0.8
Check & Repair Snubbers	0.015	200	3.0
Reactor Coolant Pump Motor Work	0.100	60	6.0
Pressurizer Work	0.005	60	0.3
Containment Pressure Test & Valve Repair Containment Instrument Calibration	0.050	50	2.5
Repair Containment Sump Pumps & Level Indicators	0.050	50	2.5
Replace Excore Detectors	0.050	90	4.5
Repair Dampers & Ducts	0.005	120	0.6
Primary Aux, Bldg. Valve Repair	0.010	200	2.0
Instrument Work & Calibration	0.005	800	4.0
Gen. Maintenance	0.005	1000	5.0
Filter, Ion Exchanger & Demineralizer Repair	0.150	40	6.0
Radwaste Evaporator Repair	0.150	40	6.0
General Decon. & Revamping	0.015	750	11.0
Remove/Replace Filters For Containment Cleanup	0.100	12	1.2
Radwaste Bldg. General Maintenance	0.100	40	4.0
Boron Recovery Evaporator Repair	0.100	40	4.0
Miscellaneous			0.00
			63.4

Information Sources:

(1) Exposure rates for these activities assumed to be comparable to exposure rates for similar types of activities reported in "Radiation Exposure Management - The Westinghouse ALARA Program," J. D. Cohen and J. R. Magaw (Reference 7); and "Study of the Effects of Reduced Occupational Radiation Exposure Limits on the Commercial Nuclear Power Industry," Atomic Industrial Forum, Inc. Abstracts of both papers appear in the preliminary draft of "Technology Planning Study of Occupational Radiation Exposure in LWR Nuclear Power Plants," Electric Power Research Institute, January 22, 1980 (Reference 6).

(2) Estimate of total exposure time based on engineering judgment.

(3) Total annual exposures consistent with historical data for similar activities reported in "Compilation and Analysis of Data on Occupational Radiation Exposure Experienced at Operating Nuclear Power Plants" (Reference 1).

TABLE 12.4-4 OCCUPATIONAL DOSE ESTIMATE - REFUELING OPERATIONS

Activity	Radiation Field (Rem/Hr)	Exposure Time (Man-Hrs)	Total Dose (Man-Rem)
Move, Setup, Check Fuel Transfer Equip.	0.1-0.005	32	0.62
Remove Seismic Supports	0.025	8	0.20
Disconnect CRDM & DRPI Cables	0.025	28	0.70
Remove RV Head Insulation	0.1	20	2.00
Remove Transfer Tube Quick Closure Hatch	0.1	See Miscellaneous	
Retract In-Core Thimbles	0.005	20	0.10
Connect RV Head Eductor	0.1	2	0.20
Lower Stud Tensioners Racks to Cavity	0.005-0.025	12	0.15
Relax RV Studs	0.05	30	1.50
Remove RV Studs	0.05	21	1.05
Install Stud Hole Plugs	0.05	9	0.45
Install RV Guide Studs	0.05	12	0.60
Install Permanent Cavity Seal Ring Hatches	0.05	2	0.1
Install Head Lift Rig Load Cell	0.1	6	0.60
Remove RV Head	0.005-1	8	0.21
Disconnect Control Rod Drive Shafts	0.1	9	0.90
Remove Upper Internals	0.005-0.1	16	0.27
Index Refueling Crane	0.005	4	0.02
Shuffle Fuel	0.005	960	4.80
Map Fuel Core Locations	0.005	15	0.07
Clean, Inspect UT Studs, Nuts & Tensioners	0.005	60	0.30
Install New RV Head O Rings	0.5	4	2.00
Install Upper Internals	0.005-0.1	12	0.25

Activity	Radiation Field (Rem/Hr)	Exposure Time (Man-Hrs)	Total Dose (Man-Rem)
Latch Control Rod Drive Shafts	0.1	9	0.90
Drain Cavity	--	3	--
Clean Vessel Flange Surface	0.5	10	5.00
Install In-Core Thimbles	0.005	20	0.10
Install RV Head Lifting Rig	0.005-0.5	14	0.80
Cavity Decontamination	0.025	96	2.40
Remove Stud Hole Plugs	0.05	16	0.80
Clean Stud Holes	0.05	16	0.80
Install RV Head Studs	0.05	54	2.70
Tension RV Head Studs	0.05	105	5.25
Install Transfer Quick Closure Hatch	0.1	See miscellaneous	
Flood Cavity	--	3	--
Connect CRDM & DPRI Cables	0.025	28	0.70
Remove Permanent Cavity Seal Ring Hatches	0.05	2	0.1
Install RV Head Insulation	0.1	20	2.00
Install Seismic Supports	0.025	8	0.20
Miscellaneous*			1.00
		TOTAL	39.84

* Includes the removal and installation of the transfer tube quick closure hatchInformation Source:"Radiation Exposure Management - The Westinghouse ALARA Program," J. D. Cohen and J. R. Magaw of Westinghouse Nuclear Energy Systems (Reference 7); paper presented at the International Radiation Protection Conference, Paris, December 1979. The abstract of this paper appeared in the preliminary draft of "Technology Planning Study of Occupational Radiation Exposure in LWR Nuclear Power Plants," which was prepared for the Electric Power Research Institute by Stone & Webster Engineering Corporation, January 22, 1980 (Reference 6).

TABLE 12.4-5 OCCUPATIONAL DOSE ESTIMATES DURING WASTE PROCESSING

Activity	Average Dose Rate (Mrem/Hr)	Exposure[3] Time Per Event (Man Hours)	Frequency	Dose (Man-Rem/Year)
Control Room Panel	0.5[1]	8	3/day	4.37
Sampling & Filter Changing	15.0[2]	16	1/week	12.48[4]
Panel Operation, Inspection & Testing	0.5[1]	2	1/day	0.37
Operation of Waste Processing and Packaging Equipment	5.0[2]	40	1/week	10.4[4]
Miscellaneous				0.00
			Total	27.60

(1) Average dose rate assumed to be 20 percent of the maximum design dose rate for the radiation zone in which the equipment is located. Radiation zones and justification for this assumption provided in Table 12.3-1.

(2) Average dose rate for these activities assumed to be equal to the maximum design dose rate for the radiation zone in which the equipment is located.

(3) Exposure time estimates based on engineering judgment.

(4) Total annual exposure for the activity consistent with historical data reported in "Compilation and Analysis of Data on Occupational Radiation Exposure," (Reference 1).

TABLE 12.4-6 OCCUPATIONAL DOSE ESTIMATES DURING IN-SERVICE INSPECTION

Activity	Radiation[1] Field (R/Hr)	Exposure[2] Time (Hrs/Year)	Total Dose[3] (Man-Rem/Year)
Erect/Remove Scaffolding	0.02	50	1.0
Remove/Replace Insulation	0.05	100	5.0
Remove/Replace Primary Manway Covers	0.02	125	2.5
Remove/Replace Secondary Manway Covers	0.005	80	0.4
Complete Steam Generator Tube Inspection	0.12	125	15.0
Steam Generator Secondary Side	0.01	50	0.5
Reactor Coolant Pumps	0.01	50	0.5
Reactor Vessel	0.01	80	0.8
Makeup Pumps	0.005	80	0.4
Pressurizer	0.005	40	0.2
Containment Piping	0.15	60	9.0
Reactor Vessel Internals	0.05	140	7.0
		Total	42.3

Information Sources:

(1) Exposure rates for these activities assumed to be comparable to exposure rates for similar types of activities reported in "Radiation Exposure Management - The Westinghouse ALARA Program," J. D. Cohen and J. R. Magaw, and "Study of the Effects of Reduced Occupational Radiation Exposure Limits on the Commercial Nuclear Power Industry," Atomic Industrial Forum, Inc. Abstracts of both papers appear in the preliminary draft of "Technology Planning Study of Occupational Radiation Exposure in LWR Nuclear Power Plants," Electric Power Research Institute, January 22, 1980.

(2) Estimate of total exposure time based on engineering judgment.

(3) Total annual exposures consistent with historical data for similar activities reported in "Compilation and Analysis of Data on Occupational Radiation Exposure Experienced at Operating Nuclear Power Plants" (Reference 1).

TABLE 12.4-7 OCCUPATIONAL DOSE ESTIMATES DURING SPECIAL MAINTENANCE

Activity	Radiation Field (Rem/Hr)	Exposure Time (Man-Hrs)	Total Dose (Man-Rem)
Residual Heat Removal Pump Maintenance	0.200[1]	60	12.00
Reactor Coolant Pump Maintenance	0.100[1]	150	15.00
Steam Generator Maintenance and Tube Plugging	0.150[1]	360	54.00
Steam Generator - Eddy Current Testing	0.125[1]	240	30.00
Block Valve Maintenance	0.250[1]	72	18.00
Water Lancing	0.125[2]	120	15.00
Misc Pipe Repair	0.100[2]	50	5.00
Replace Upper Guide Structure on Control Rods	0.125[2]	120	15.00
Miscellaneous			0.00
		Total	164.00

Information Sources:

(1) "Study of the Effects of Reduced Occupational Radiation Exposure Limits on the Commercial Nuclear Power Industry," (to be published) prepared for Atomic Industrial Forum, Inc., by Catalytic, Inc., (1979-1980) (Reference 8). The abstract of this paper appeared in the preliminary draft of "Technology Planning Study of Occupational Radiation Exposure in LWR Nuclear Power Plants" which was prepared for the Electric Power Research Institute by Stone & Webster, Engineering Corporation, January 22, 1980. (Reference 6).

(2) Exposure rates for these activities assumed to be the same as for similar types of activities reported in Reference 1.

(3) Average total annual exposures for these activities obtained from "Compilation and Analysis of Data on Occupational Radiation Exposure Experienced at Operating Nuclear Power Plants," Atomic Industrial Forum, Inc., September 1974 (Reference 1); and material appended to the preliminary draft of "Technology Planning Study of Occupational Radiation Exposure in LWR Nuclear Power Plants," Electric Power Research Institute, January 22, 1980 (Reference 6).

TABLE 12.5-1 PORTABLE HEALTH PHYSICS INSTRUMENTATION

Type of Instrument	Quantity Minimum	Sensitivity	Range	Calibration Method	Calibration Frequency
Ion Chamber (Low Range)	16	Beta, Gamma	0 to 1 R/hr.	Source	Semiannual
Ion Chamber (Mid Range)	8	Gamma	0 to 50 R/hr.	Source	Semiannual
Ion Chamber (High Range)	4	Gamma	Up to 10,000 R/hr.	As recommended by manufacturer	
Geiger Mueller Detector	6	Beta, Gamma	0 to 50,000 cpm	Source and Pulse Generator	Semiannual
	10	Beta, Gamma	0 to 200 mR/hr	Source	
Alpha Scintillation or Proportional Detector	4	Alpha	0 to 500,000 cpm	Source	Semiannual
Telescoping Survey Instrument	4	Gamma	0 to 1,000 R/hr.	Source	Semiannual
Neutron Dose Rate Detector	3	Neutron	0.001 to 5 rem/hr	As recommended by manufacturer	
Air Sampler (Low Volume)	10	Particulate and Iodine	--	Flow Rate	Semiannual
Air Sampler (High Volume)	6	Particulate and Iodine	--	Flow Rate	Semiannual
Air Sampler (Personnel)	10	Particulate	--	Flow Rate	Semiannual
Continuous Air Monitor	4	Particulate and Noble Gas Monitor and Iodine Sampler	--	Flow Rate and Response	Semiannual

TABLE 12.5-2 PERSONNEL MONITORING INSTRUMENTS

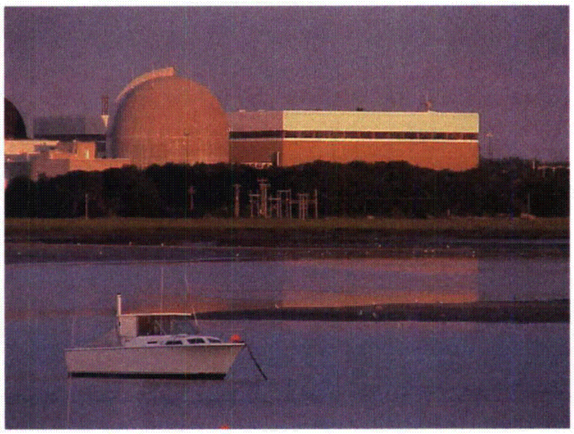
Type of Detector	Minimum Quantity	Sensitivity	Range
Hand held frisker	24	Beta, Gamma	0 to $5x10^4$ cpm
Portal Monitor or large area detection device	2	Beta or Gamma	Variable Alarm Setpoint
Self-reading dosimeter Note 1	No Min. Required	Gamma	Multiple ranges available
TLD	750 Station Use	Beta, Gamma, Neutron	Meets industry Standards
	250 Emergency	Beta, Gamma	
Whole body counter	1	Gamma	Meets industry Standards
Electronic Dosimeters	300	Gamma	Meets industry Standards

Note 1: The Station has transitioned to use of electronics dosimeters. SRPDs are available for use; however, they are not maintained calibrated. Calibration would be performed prior to use.

SEABROOK STATION
UPDATED FINAL SAFETY ANALYSIS REPORT

CHAPTER 12
RADIATION PROTECTION

FIGURES



See 805184

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Radiation Zone Map - Containment Structure - Plan at Elevation (-) 26'-0"	
		Figure 12.3-1

See 1-NHY-805185

See 805186

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Radiation Zone Map - Containment Structure - Plan at Elevation 25'-0	
		Figure 12.3-3

See 805187

See 805188

See 805189

See 805190

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Radiation Zone Map - Primary Auxiliary Building - Plans at Elevation 53'-0 And 81'-0	
		Figure 12.3-7

See 805896

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Radiation Zone Map - Waste Processing Building - Plans at Elevations (-) 31'-0, (-) 18'-3 and 13'-0	
		Figure 12.3-8

See 805891

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Radiation Zone Map - Waste Processing Building - Plan at Elevation (-) 3-0	
		Figure 12.3-9

See 1-NHY-805892

See 1-NHY-805897

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Radiation Zone Map - Waste Processing Building - Plans at Elevations 42'-5 And 65 '-0
	Figure 12.3-11

See 1-NHY-805893

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Radiation Zone Map - Waste Processing Building - Plan at Elevation 53'-0	
		Figure 12.3-12

See 1-NHY-805894

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Radiation Zone Map - Waste Processing Building - Partial Plans at Elevation 53'-0 And 9 '-0	
		Figure 12.3-13

See 805895

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Radiation Zone Map - Waste Processing Building - Plan at Elevation 86'-0	
		Figure 12.3-14

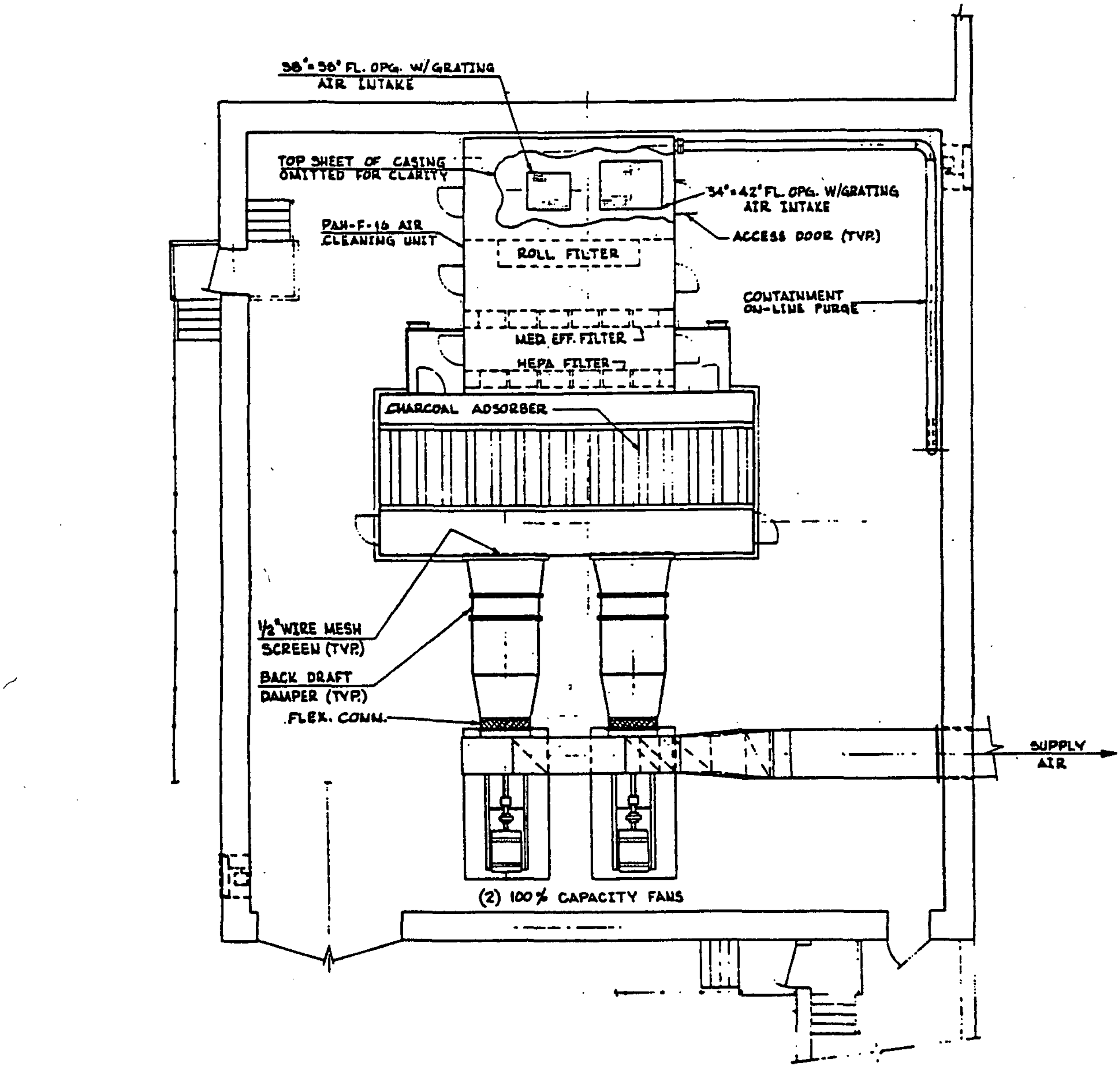
See 805181

See 805182

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Radiation Zone Map - Fuel Storage Building - Plan at Elevations 7'-0 And 10 '-0	
		Figure 12.3-16

See 1-NHY-805183

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	RCA Boundary - Administration and Service Building - First Floor Plan at Elevation 21'	
		Figure 12.3-17



SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Normal Exhaust Unit – Primary Auxiliary Building	
		Figure 12-3-18

See 1-NHY-500017 Sh. 1

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Area Radiation Monitors System - Instrumentation Engineering Diagram	
		Figure 12.3-19 Sh. 1 of 2

See 1-NHY-500017 Sh. 2

See 1-NHY-500016 Sh. 1

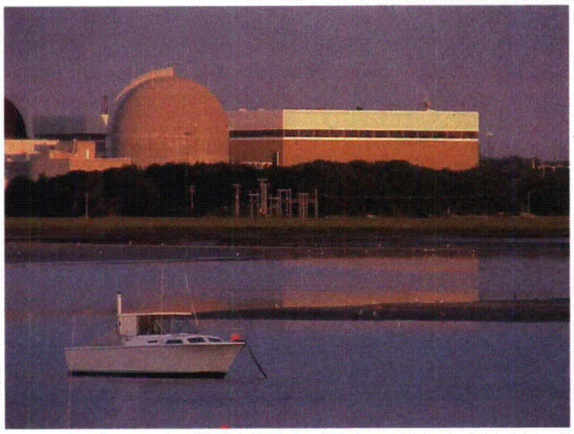
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Airborne Radiation Monitoring System - Instrumentation Engineering Diagram	
		Figure 12.3-20 Sh. 1 of 2

See 1-NHY-500016 Sh. 2

Figure 12.5-1 Deleted

SEABROOK STATION
UPDATED FINAL SAFETY ANALYSIS REPORT

CHAPTER 13
CONDUCT OF OPERATIONS



13.1 ORGANIZATIONAL STRUCTURE

This section describes the organization of NextEra Energy Seabrook, LLC (NextEra Seabrook). NextEra Seabrook is an indirect, wholly-owned subsidiary of NextEra Energy Resources, LLC, the independent power producer subsidiary of FPL Group, Inc. FPL Group affiliates operate, and have ownership interests in, several other nuclear units besides Seabrook Station. These include St. Lucie Units 1 and 2 and Turkey Point Units 3 and 4 in Florida operated by Florida Power and Light Company (FP&L), and Duane Arnold Energy Center (DAEC) and Point Beach Units 1 and 2 operated by NextEra Energy Resources. While NextEra Seabrook does not rely on other organizations to establish its technical qualifications, additional support is available from FP&L's Nuclear Division. Contracts may also be established with third parties (e.g., NSSS, AE) for support services. Notwithstanding any service that may be provided to Seabrook Station by FP&L or third parties, NextEra Seabrook remains responsible, at all times, for the management, operation and maintenance of Seabrook Station.

13.1.1 Management and Technical Support Organization

NextEra Seabrook is responsible for the operation and maintenance of Seabrook Station. The Senior Vice President and Nuclear Chief Operating Officer has final site authority and responsibility for the overall safe operation and maintenance of Seabrook Station. This responsibility has been delegated to the Site Vice President who is the management official in overall charge of the station.

The internal corporate organizational relationships are described in the FPL Quality Assurance Topical Report (QATR). Resumes of corporate management personnel are available upon request.

13.1.1.1 Seabrook Station Organization, Responsibilities and Authority

The Seabrook Station organization is under the overall direction of the Senior Vice President and Nuclear Chief Operating Officer (NCOO) who has overall responsibility for the operation and operational support for Seabrook Station. The Site Vice President reporting to the NCOO has overall charge of the station and is responsible for day to day nuclear site operations. The Plant General Manager, reporting to the Site Vice President, is responsible for day to day activities associated with operation and maintenance of the station.

The Director of Engineering, reporting to the Site Vice President, is responsible for NextEra Seabrook engineering design, engineering support, configuration management, plant engineering and configuration control.

The Security Manager, reporting to the Director of Fleet Security, is responsible for the day to day activities associated with the physical security of Seabrook Station.

The responsibilities, training, organization and qualifications of the Station Staff are discussed in Subsection 13.1.2. The responsibilities of the Training Staff are discussed in Subsection 13.1.1.2.

13.1.1.2 Seabrook Training Organization, Responsibilities and Authority

NextEra Seabrook has recognized the importance of training by establishing training facilities and by providing a Seabrook site-specific simulator. The Training Center is located on the Seabrook site outside of the protected area.

The training facilities contain classrooms, office space, a library, study areas, instructor material preparation rooms, a computer room, administrative areas, and a simulated Seabrook control room with a full-size main control board and various main control room panels. The simulator control board was manufactured by Link, a division of the Singer Company. Link has had extensive experience with nuclear simulators and a myriad of simulators for military applications. Seabrook represents the eighteenth simulator built by Link for the nuclear industry. The simulator control room is not only similar to the actual control room in appearance, but is also operated under the same working conditions as the actual main control room to provide a realistic atmosphere for operator training.

The Nuclear Training Manager reports to the Site Vice President. Resumes of key training personnel are available upon request.

Operator training is performed under the cognizance of the Operations Training Supervisors who are responsible for the implementation of initial and requalification training for licensed and non-licensed operators. The training program for operations instructors that are licensed is identical to that for operations SRO licensed personnel and, for SRO certified instructors, similar to that for operations SRO licensed personnel. All instructors also attend requalification training annually related to instructional skills.

The Training Department performs training for technical and management staff. This function is further discussed in Subsection 13.2.2.

13.1.2 Operating Organization

13.1.2.1 Station Organization

The Seabrook Station organization includes all the technically trained personnel necessary to support all aspects of Unit 1 operation.

The key supervisory positions for the station organization were filled in 1979. Personnel to meet the operational requirements of Unit 1 were hired on a phased basis consistent with the training and licensing requirements of the individual positions.

The Unit 1 on-duty operating shift crews are composed as shown in Technical Specification Table 6.2-1, and meet the requirements outlined in Technical Specification Subsection 6.2.2 describing the plant organization. Manpower necessary to staff six shift crews is provided. Each member of the station organization meets, or exceeds, the minimum qualifications recommended for comparable positions in Regulatory Guide 1.8 as described in Chapter 1, Section 1.8.

A retraining and replacement licensed training program for the Station Staff shall be maintained under the direction of the Training Manager in accordance with the Seabrook Station Institute of Nuclear Power Operations (INPO) Accredited Programs.

The employees assigned to the station organization have been trained as described in Section 13.2.

13.1.2.2 Station Personnel Responsibilities and Authorities

a. Overall Station Management

The Site Vice President is responsible for overall management of Unit 1, including operation and maintenance.

The Plant General Manager is responsible for day to day activities associated with operation and maintenance of Unit 1. In his absence, the Operations Manager will assume these responsibilities. The Shift Manager assumes these responsibilities when station management is not within the station. In addition, the Plant General Manager may designate in writing other qualified personnel to assume these responsibilities in his absence.

The Plant General Manager reports to the Site Vice President for all activities related to the station and is responsible for performance improvement.

The Performance Improvement Manager reports to the Plant General Manager and is responsible for administration of the corrective action and self-assessment programs.

This position is also responsible for NUREG-0737, Action Plan Item I.B.1.2 technical review functions committed to regarding the oversight, implementation, and coordination of internal and external operating experience.

Also reporting to the Plant General Manager are the following:

1. Operations Manager
2. Maintenance Manager
3. Work Control Manager
4. Radiation Protection Manager
5. Chemistry Manager
6. Industrial Safety Supervisor

The functions, responsibilities and authorities for station positions under the direct cognizance of these managers are defined below.

b. Operations

The Operations Manager reports to the Plant General Manager and is responsible for the safety and operation of the unit's equipment in accordance with written and approved station procedures. He has the authority to order the shutdown of the reactor, when in his judgment such action is required to protect the safety of the station or the health and safety of the public. The Operations Manager holds, or has held, a Senior Reactor Operator's License at Seabrook Station. He also supervises the Assistant Operations Manager, Operations Support Manager, Operations Work Management Manager, and the Firefighter Supervisor.

The Assistant Operations Manager - Operations directs the activities of the Shift Managers. He reports to the Operations Manager and assumes the responsibilities of the Operations Manager in the formers absence. He is responsible for the safety and operation of the unit's equipment in accordance with written and approved station procedures. He has the authority to order the shutdown of the reactor, when in his judgment such action is required to protect the safety of the station or the health and safety of the public. The Assistant Operations Manager - Operations holds a Senior Reactor Operator's License.

The Operations Support Manager reports to the Operations Manager and directs the activities of the Operations Department Procedure and Technical Projects groups. The Operations Support Manager coordinates the resolution of departmental discrepancies discovered during internal or external audits, inspections, and review activities and coordinates and updates Operations Department programs developed in support of Station Operations. The Operations Support Manager is responsible for implementing changes to Operations Department procedures to include the scheduling, review and approval of procedures. The Operations Support Manager has held a Senior Reactor Operator's License.

1. Operating Shift Crew

An operating shift crew normally consists of a Shift Manager and one Unit Supervisor, two Control Room Operators and three Nuclear Systems Operators. The Shift Manager and Unit Supervisor possess Senior Reactor Operator's Licenses; Control Room Operators possess Reactor Operator's Licenses. The minimum shift crew composition for various modes of unit operation is shown in Technical Specification Table 6.2-1.

(a) Shift Manager (SM)

Each Shift Manager reports to the Assistant Operations Manager. The SM is responsible for the safety and operation of the station's equipment in accordance with written and approved station procedures. Each Shift Manager has the authority to order the shutdown of the reactor when in his/her judgment such action is required to protect the safety of the unit or health and safety of the public. The Shift Manager has in addition to a Senior Reactor Operator's License, the training and qualifications of a Shift Technical Advisor or a qualified Shift Technical Advisor will be assigned to his/her shift. If qualified, the Shift Manager functions as the Shift Technical Advisor and provides requisite technical expertise to the Unit Supervisor in the event of any abnormal operational occurrences.

(b) Shift Technical Advisor (STA)

If the Shift Manager is not qualified to function as the Shift Technical Advisor, a dedicated STA will be assigned to his/her shift. The STA provides engineering and accident assessment advice to the Shift Manager in the event of abnormal or accident conditions. The STA should assume an active role in shift activities by reviewing plant logs, participating in shift turnover activities and maintaining an awareness of plant configuration and status.

(c) Unit Supervisor (US)

The Unit Supervisor is responsible for ensuring all unit operations are conducted in accordance with appropriate station orders, procedures and Technical Specifications. The US is responsible for maintaining a record of all shift activities and establishing unit electrical load, as directed by the Shift Manager or as emergency conditions dictate. The US directs the Control Room Operators and the Nuclear Systems Operators in their daily activities. The US has the authority to order the shutdown of the reactor when in his/her judgment such action is required to protect the safety of the unit or the health and safety of the public. Each Unit Supervisor holds a Senior Reactor Operator's License

(d) Control Room Operators (CRO)

The Control Room Operators monitor the unit's status and make adjustments, as needed, to maintain control of the various plant processes. Most CRO duties are confined to the control room although they may perform specific activities in other areas of the station under the direction of the Unit Supervisor. The Control Room Operators each hold a Reactor Operator's License.

(e) Nuclear Systems Operators (NSO)

The Nuclear Systems Operator performs routine inspections and surveillance activities in other areas of the unit. The NSOs maintain various logs and records as required by station procedures. They also perform routine or special radiation surveys commensurate with the duties of their job. During periods when the unit is shut down, NSOs conduct routine tests and clear/return equipment to service as directed by the Unit Supervisor. The Nuclear Systems Operators are unlicensed.

c. Radiation Protection Manager

The Radiation Protection Manager reports to the Plant General Manager and has the responsibility and authority for all aspects of the Radiation Protection Program or its implementation. He is responsible for monitoring station activities for compliance with Heath Physics-related regulations and programs.

The Radiation Protection Manager is also responsible for the operation of the Radioactive Waste Processing System and the collection, processing, packaging and loading of radioactive material. This individual provides decontamination services, shielding installation and labor support.

d. Chemistry Department Manager

The Chemistry Department Manager reports to the Plant General Manager and has the direct responsibility for ensuring that the nuclear and steam portions of the station operate within the appropriate water quality specifications which includes water treatment and conditioning for specific station needs. He is responsible for verifying that all liquid, resin, gaseous and hazardous wastes are properly analyzed and processed for station reuse or disposal.

e. Maintenance

The Maintenance Manager reports to the Plant General Manager and is responsible for the coordination and direction of Instrumentation and Control (I&C), Mechanical Maintenance, Electrical Maintenance, and Maintenance Technical Departments. The Maintenance Manager directs support functions that include the corrective action and preventative maintenance programs, maintenance related surveillance activities and station modification and repair activities, including scheduling the performance of the work, controlling the material and the personnel and process involved.

The Work Control Manager reports to the Plant General Manager and is responsible for the daily work scheduling, planning and scheduling of planned maintenance outages, long-range planning and scheduling of refueling outages and outage coordination, and forced outage planning and scheduling. In addition, the Work Control Manager is responsible for document control and the work status tracking and Work Control Program trend reporting.

13.1.2.3 Operating Shift Crews

The position titles, applicable operator licensing requirements, and the minimum numbers of personnel planned for each shift are described in detail in Subsection b and Technical Specification Subsection 6.2.2. During normal operations, an operating shift consists of five Nuclear Systems Operators, two Control Room Operators, a Unit Supervisor and a Shift Manager for the station.

During unit refueling operations, when the reactor core configuration is being altered, an individual having a Senior Reactor Operator's license directly supervises the refueling activities in the reactor containment.

Nuclear Systems Operators are trained in applicable station radiation protection procedures to perform routine or special radiation surveys commensurate with the duties of their job. They receive radiation worker training which includes the use of protective barriers and signs, protective clothing and breathing apparatus and limits of personnel exposure. The Shift Manager is responsible for the radiation protection program in the absence of the Health Physics Department Manager or his designated alternate. When fuel is in the reactor, a qualified health physics technician is assigned to the onsite shift to provide additional support to the Shift Manager.

When the unit is in operational modes 1 through 4, a chemistry technician qualified in primary and secondary chemistry analysis is assigned to the onsite shift to provide additional support to the Shift Manager.

13.1.3 Qualification of Nuclear Plant Personnel

13.1.3.1 Qualifications Requirements

The recommendations of Regulatory Guide 1.8, "Qualification and Training of Personnel for Nuclear Power Plants," have been used as the basis for establishing minimum qualifications for all management, supervisory and professional-technical personnel in the station organization. See Section 1.8, "Regulatory Guide 1.8" for specifics.

The education, training and experience requirements for operators, technicians and mechanics equals or exceeds the qualifications for the positions stated in ANS 3.1 and Regulatory Guide 1.8. A retraining and replacement licensed training program for the Station Staff shall be maintained under the direction of the Training Manager in accordance with the Seabrook Station Institute of Nuclear Power Operations (INPO) Accredited Programs. Established company training programs include documented academic and on-the-job training plus comprehensive qualification examinations applicable to the skill level of the position assignment. Where desirable, offsite facilities may be used for specialized training. Records of the scope, general content and level of accomplishment for each person attending offsite training are retained at the station.

The titles of plant management and supervisory personnel who will meet the minimum requirements of ANS 3.1 and Regulatory Guide 1.8 are listed below with their equivalent ANS 3.1 title.

	Station Title	ANS 3.1 Title
a.	Plant General Manager	Plant Manager
b.	Operations Manager	Operations Manager
c.	Assistant Operations Manager -Operations	Operations Manager
d.	Operations Support Manager	Operations Manager
e.	Maintenance Manager	Maintenance Manager
f.	Shift Manager	Supervisor with NRC License
g.	Unit Supervisor	Supervisor with NRC License
h.	Chemistry Department Manager	Supervisor without NRC License
i.	Radiation Protection Manager	Supervisor without NRC License

j.	Nuclear Training Manager	Supervisor without NRC License
k.	Senior Project Manager	Supervisor without NRC License
l.	Work Control Manager	Supervisor without NRC License

13.1.3.2 <u>Qualifications of Station Personnel</u>

The key management individuals have been thoroughly trained in their specialty and have filled supervisory and technical positions in the station organization. In addition, most of the individuals have had extensive experience at operating nuclear power plants in their specialty. The nuclear experience of senior personnel at the time of startup was generally in the range of 8 to 20 years. Resumes for personnel holding key positions in the initial plant organization are available upon request. These personnel include the Plant General Manager, Director of Plant Support, Operations Manager, Shift Managers, Instrumentation and Control Department Manager. Many of the key personnel had Senior Operator Licenses or Operator's Licenses at other operating plants or have had extensive nuclear submarine operational responsibilities. Most of the Unit Supervisors and Control Room Operators and at least one individual in each of the major technical disciplines (nuclear engineering, chemistry, health physics, instrumentation and controls) have at least five years of similar experience.

13.2 TRAINING

13.2.1 Licensed Operator Training

a. General Discussion

The licensed operator training programs provide personnel with skills and knowledge, related to the operation of Seabrook Station, necessary to ensure that each individual can safely and effectively perform various assignments. Eligibility of individuals to license or renew a license pursuant to the requirements of 10 CFR 55 is certified by the Site Vice President.

The overall objectives of the licensed operator training programs are

- to train the staff to operate the unit safely, dependably and economically, and
- to prepare Shift Managers, Unit Supervisors, Control Room Operators, and selected members of the station staff for the NRC licensing examination for Reactor Operator (RO) and Senior Reactor Operator (SRO).

The safe, efficient operation of a nuclear power plant depends on the qualifications and proficiency of its personnel. Several basic categories of training are necessary to provide licensed personnel with a high degree of competence and professionalism, and these categories of training are conducted in the following Seabrook Station programs:

1. Licensed Operator Initial Training Program

   This program provides the training necessary for all personnel who require NRC operating licenses for Seabrook Station, and it meets or exceeds the minimum requirements of 10 CFR 55 and Regulatory Guide 1.8.

2. Shift Technical Advisor Training Program

   This program trains Shift Managers, Unit Supervisors or other designated personnel for the duties and responsibilities of the Shift Technical Advisor on shift. The program meets or exceeds the requirements of 10 CFR 50.120, and Regulatory Guide 1.8.

3. Senior Reactor Operator Training Program

This program provides both operations and skills-based training to ensure senior license candidates can safely operate Seabrook Station as SROs. The program meets or exceeds the minimum requirements of 10 CFR 55 and Regulatory Guide 1.8.

4. Licensed Operator Requalification Training Program

This program provides the training necessary to maintain the proficiency of all Seabrook licensed personnel. The program meets or exceeds the minimum requirements of 10 CFR 55.59.

b. Program Effectiveness and Evaluation

A program for monitoring training effectiveness is established for all areas of license training. Program reviews and evaluations are conducted as directed by the Plant Training Advisory Board (PTAB) and its subcommittees, Training Review Committees (TRCs). Evaluations are used to measure, control, and improve training programs, and they are accomplished by monitoring job performance and reviewing reports from independent parties such as auditors.

Training program deficiencies, identified through evaluation or self-assessment, will be brought to the attention of training management as soon as practical for disposition and/or presentation to the PTAB and TRC along with recommendations for corrective action.

c. Program Accreditation and Instructor Certification

Seabrook Station is committed to the accreditation process implemented by the Institute of Nuclear Power Operations (INPO) and endorsed under the NRC's Final Policy Statement on Training and Qualification of Nuclear Power Plant Personnel, as amended. The company's training programs, including licensed operator training, are accredited by the National Nuclear Accrediting Board under this commitment. This accreditation qualifies Seabrook Station for the membership it maintains in the National Academy for Nuclear Training.

Accreditation of training ensures that Seabrook Station's licensed operator training programs meet or exceed the requirements for a systematic approach to training (SAT) as stated in 10 CFR 55, Section 4. The programs qualify for the special status granted to systems-based training by Sections 31 and 59 of 10 CFR 55.

Seabrook Station's training policy requires instructors who provide instruction in safety systems, integrated response, transients, and simulator courses to be NRC-licensed or certified by the facility as having equivalent knowledge.

Guest lecturers considered being experts by the nature of their work responsibilities, will be used on a limited basis to supplement training staff instructors. These guest lecturers are exempt from having an NRC operator's license or facility certification.

d. Responsibilities

The following personnel are responsible for various areas of Seabrook Station's operator license training programs.

1. Site Vice President

The Site Vice President has overall responsibility for qualification and proficiency for individuals working at Seabrook Station.

2. Nuclear Training Manager

The Nuclear Training Manager provides direction and control for the conduct of training at Seabrook Station and reports to the Director of Nuclear Fleet Training. The Training Manager has responsibility for administrative activities, program development and evaluation, and record keeping of accredited training programs. The Training Manager is also responsible for the development and implementation of training conducted in support of the following training programs:

Fire protection

Emergency preparedness

General Employee Training including Plant Access Training

Instructional skills training

Dry fuel storage

13.2.1.1 Licensed Operator Initial Training Program

The licensed operator initial training program provides an individual with the knowledge and skills necessary to safely operate Seabrook Station and to obtain an NRC RO or SRO License.

A program description is used to ensure the licensed operator initial training program meets or exceeds the requirements of 10 CFR 55. The program description applies until it is superseded. Following are general descriptions of the various training segments:

a. Generic Fundamentals

This training consists of reactor physics, thermodynamics, and plant components.

Training methods may include classroom instruction, computer-based training (CBT), in-plant walkthroughs, laboratory exercises, and simulator demonstrations.

b. Detailed Systems

This consists of electrical distribution, as well as primary, secondary, and balance-of-plant systems. The training also covers appropriate theory review, integrated systems response, procedures, and administrative controls.

Training methods may include classroom instruction, CBT, in-plant walkthroughs, and simulator demonstrations. Component malfunctions are covered in simulator training using abnormal procedures.

c. Mitigating the Consequences of Core Damage and Transient and Accident Analysis

This training is devoted to mitigating the consequences of core damage and transient and accident analysis. The training may consist of classroom instruction, CBT, sessions on the simulator, and supervised study with problem solving.

d. Simulator Training

Simulator training stresses diagnostics and teamwork throughout to give each license candidate practice in applying the knowledge and skills needed to perform control room tasks during operating conditions. Topics include the use of normal, abnormal, and emergency response procedures, reactor start-ups, equipment locations, integrated plant operations and emergency preparedness.

e. On-Shift Participation

On-shift participation is at the level of the license being sought, and it is designed to give each license candidate an opportunity to observe plant operations and complete work assignments in the control room alongside licensed operators. On-the-job training (OJT) lasts a minimum of thirteen weeks, not necessarily sequential, for license candidates who must perform, observe, or discuss activities and tasks as specified in the Qualification Guide. Per 10 CFR 55.45, initial license candidates are required to perform a minimum of five significant control manipulations in the plant or on the simulator. Candidates are permitted to manipulate controls that affect the reactor's power level and/or reactivity under the supervision of a licensed operator.

13.2.1.2 Shift Technical Advisor Training Program

The Shift Technical Advisor (STA) training program meets or exceeds the requirements of 10 CFR 50.120 and Regulatory Guide 1.8. The specific requirements to be an STA include having an SRO license and a Bachelor of Science degree in engineering, engineering technology, or physical science from an accredited college or university. STA training emphasizes emergency tasks, emergency response procedures, and transient and accident analysis. This training provides Seabrook Station with on-shift engineering expertise and operations abilities necessary to ensure that control room shift activities support safe and efficient operation during normal and emergency plant conditions.

13.2.1.3 Shift Manager Training Program

Shift manager training is conducted under a program designed to provide a broad perspective of plant operations and a high degree of proficiency when interacting with people to implement plant policies and procedures. The program recognizes the skills and knowledge required to act as the senior manager on shift with responsibility for safe and reliable operation of plant equipment, and for protecting the health and safety of the general public and plant personnel.

Development of the shift manager training program was based on analyses of training needs for shift managers using the systematic approach to training (SAT) methodology. The analyses included consideration of a generic job and task analysis performed by the INPO and an industry panel. The program encompasses all of the terminal and enabling learning objectives derived from those analyses, and it meets or exceeds the requirements of 10 CFR 50.120.

13.2.1.4 Senior Reactor Operator Training Program

The senior reactor operator training program provides an individual with the knowledge and skills necessary to safely operate Seabrook Station and to obtain an NRC SRO license.

A program description is used to ensure the senior reactor operator training program meets or exceeds the requirements of 10 CFR 55. The program description applies until it is superseded. A brief description of the various training segments of this program is found in Subsection 13.2.1.1 of this document. Following is a list of those training segments:

a. Mitigating the Consequences of Core Damage and Transient and Accident Analysis

b. Simulator Training

c. On-Shift Participation

13.2.1.5 Licensed Operator Requalification Training Program

The licensed operator requalification training program provides instruction for all licensed operators so as to maintain proficiency and a high level of knowledge for their jobs. A program description is used to ensure the licensed operator requalification training program complies, within the framework of a systems approach to training, with 10 CFR 55, "Operators' Licenses." The program description applies until it is superseded. Following is a brief description of the interrelated elements which make up this program:

a. Lecture Series

1. Design Change, Procedure Revision, and Industry Experience Review

This portion of the program will ensure that appropriate changes and revisions to plant design, changes to procedures and Technical Specifications, and industry experiences are reviewed by each licensee. Design changes, procedure revisions and industry experience reviews are incorporated into lessons via Training Development Recommendations (TDRs) when appropriate.

All licensees review the applicable operating experience of Seabrook Station as well as selected operational information from the nuclear industry. The following reports and publications are among those that will be considered in obtaining information for analysis and review:

- Licensee Event Reports (LERs)
- Condition Reports (CRs)
- Audit, Evaluation, and Inspection Reports

- NRC IE Information Notices and Bulletins
- Publications and periodicals covering nuclear industry information
- INPO Significant Event Reports (SERs) and Significant Operating Experience Reports (SOERs).

2. <u>Retraining Lectures for License Holders</u>

A formal classroom lecture series is conducted as part of the requalification program. The level of instruction will be consistent with the level of license held. This lecture series covers two general areas:

- Fundamentals and Systems Review
- Procedures and Administrative Controls

Fundamentals and Systems Review lectures present instruction based on information from standard reference sources relating to topics such as reactor theory, plant design, and radiation control. Procedures and Administrative Controls lectures cover topics involving essential plant operational guidelines such as Technical Specifications and administrative and operating procedures.

b. <u>On-the-Job Training</u>

On-the-job training is designed to ensure that all licensed personnel operate reactor controls and participate in major evolutions. On-the-job training is conducted throughout the term of the operator's license, and all required on-the-job training is completed prior to license renewal.

Seabrook Station's certified simulation facility is used to ensure that required control manipulations not performed in the plant are performed on the simulator during the term of the operator's license.

The simulator is used in requalification training to emphasize such areas as infrequently performed procedures, required responses to abnormal and emergency procedures, and significant operating events.

c. Requalification Examinations

An examination or quiz is administered in accordance with the training program description. These quizzes examine a combination of classroom and simulator objectives presented to date, and they will parallel, in content and degree of difficulty, segments of an annual requalification examination. Examinations and quizzes will be retained as a part of the training record.

A comprehensive written examination will be administered to all licensees at least every two years. These examinations incorporate many of the requirements of NUREG-1021, Operator Licensing Examination Standards for Power Reactors.

Each application for renewal of an RO or SRO license will be accompanied by a statement, signed by the Site Vice President, certifying that the applicant has satisfactorily completed the requalification program during the effective term of his or her current license, and that he or she has discharged license responsibilities competently and safely.

13.2.1.6 Requalification Training Program Records

Requalification training program records will be maintained for a minimum of six years from the date of the recorded event to document the participation of each licensed RO and SRO in the program. The records will include copies of written examinations administered, answers provided by the licensees, and results of evaluations.

13.2.1.7 Activation of Inactive License Program

An inactive license is defined as a license held by an individual who has participated fully in the Seabrook Station licensed operator requalification program, but has not actively performed the functions of a licensed RO or SRO for a minimum of seven, eight-hour shifts or five, twelve-hours shifts per calendar quarter.

For an individual with an inactive license to resume the functions authorized by the license, the conditions specified by Section f of 10 CFR 55.53 must be met.

13.2.1.8 Applicable Documents

The training programs listed under Items 1 through 4 in Subsection 13.2.1.a will be conducted in accordance with applicable requirements in the NRC section of Title 10 of the CFR, and they will meet the intent of applicable recommendations provided by Regulatory Guides and other publications. The recommendations and regulatory requirements are included in the program descriptions where applicable.

13.2.2 Training for Nonlicensed Personnel

a. General Discussion

The comprehensive training programs conducted for nonlicensed personnel comply with the provisions and intent of Regulatory Guides and other publications identified in the individual program descriptions. Subsections 13.2.2.1 through 13.2.2.11 have brief descriptions of the training each nonlicensed group receives. The programs meet or exceed the requirements of 10 CFR 50.120.

The overall objective of the nonlicensed training programs is to train a staff to operate and maintain the unit safely, dependably, and economically. The nonlicensed training programs provide instruction to personnel in various disciplines who participate in the maintenance and operation of Seabrook Station. The degree to which an employee is trained is consistent with his or her experience, the task lists for his or her job, and regulatory requirements.

A program for monitoring training effectiveness is established for all areas of nonlicensed training. That program is described in Subsection 13.2.1.b of this document.

Seabrook Station applies the same systematic approach to training for nonlicensed personnel that it applies to licensed operator training. The company's nonlicensed training programs that come under INPO's accreditation program are accredited by the National Nuclear Accrediting Board.

Qualification guides specify prerequisite training and requisite job performance criteria for various job descriptions. SAT is used to identify needs, develop and deliver training, solicit plant feedback, and modify training as indicated by the feedback. Training settings include classroom, self study, CBT, simulator, laboratory, and on-the-job training (OJT).

OJT practices are defined in the Training and Qualification Manual (NAQM). Individuals qualified to perform specific tasks and trained to conduct OJT for the tasks are selected by the department manager to serve as OJT instructors and task performance evaluators. Employees may receive credit for designated segments of training under a formal process for the validation of equivalent previous training.

b. Responsibilities

The responsibilities for implementation of nonlicensed training programs are described in Subsection 13.2.1.d. of this document.

13.2.2.1 Instrumentation and Control Technicians

Designated technicians are trained on selected plant protection and control systems including Solid-State Protection, 7300 Process, Nuclear Instrumentation, Rod Control and Rod Position Indication. Selected technicians are also trained on additional plant systems including Radiation Monitoring, Main Turbine Electrohydraulic Control, and selected plant fluid and electrical systems.

13.2.2.2 Mechanics

Mechanics, including radwaste technicians, receive training in hand tools, basic mechanical maintenance, and mechanical maintenance of site equipment. Training in such areas as bolting, snubbers, welding, nondestructive examination, crane operation, advanced vibration, and hydraulic wrenches is available for those who perform work requiring such training.

13.2.2.3 Electricians

Electricians receive training in basic electrical maintenance and applied electrical maintenance of site equipment.

13.2.2.4 Radwaste Technicians

Radwaste technicians receive training using classroom and OJT formats, and that training includes lessons in decontamination, radwaste processing, handling, shipping, and disposal as well as annual instruction on industry events.

13.2.2.5 Health Physics Personnel

Health Physics personnel receive training on plant systems, as well as on health physics practices, procedures, and equipment.

13.2.2.6 Chemistry Technicians

Chemistry technicians receive instruction in four fundamental areas: basic chemistry, instrumental/analytical chemistry, plant systems chemistry and chemical control, and radiochemistry and radiochemical analyses.

13.2.2.7 Emergency Preparedness Training

All personnel assigned to the emergency response team receive emergency preparedness training to perform the functions of their position as specified in Section 12.2 of the Seabrook Station Radiological Emergency Plan.

13.2.2.8 Engineering Support Personnel (ESP)

The ESP training program uses a systematic approach to training to provide broad-based training for Seabrook Station personnel who participate in engineering support activities. ESP training supplements entry-level knowledge and skills and provides Seabrook-specific information. This enhances their ability to perform assigned duties in a manner that promotes safe and reliable plant operation. Some specialized topics in the ESP program are taught by technical subject matter experts from engineering support groups.

13.2.2.9 Fire Protection Personnel

The fire protection training program for Seabrook Station follows the guidance provided in Appendix R of 10 CFR 50 and the NRC document titled, "Nuclear Plant Fire Protection Functional Responsibilities, Administrative Controls, and Quality Assurance." Following is a brief description of the training program:

a. Fire Brigade

All fire brigade members attend regularly scheduled practice sessions on the proper method of fighting various types of fires. These sessions must be conducted at least annually to provide brigade members with hands-on experience in extinguishing actual fires using equipment available at Seabrook Station.

Brigade members will practice as a team in periodic drills. Drills for each fire brigade will occur at regular intervals not to exceed three months.

Regular planned meetings shall be held at least every 3 months for all brigade members to review changes in the fire protection program and other subjects as necessary.

Periodic refresher training sessions shall be held to repeat the classroom instruction program for all brigade members over a two-year period. These sessions may be concurrent with the regular planned meetings.

b. Other Station Employees

All full-time employees, temporary employees and construction personnel receive instruction on fire protection safety, evacuation routes, and the procedures for reporting a fire as part of Plant Access Training (PAT).

Security personnel receive instruction on procedures for entry of offsite fire departments, crowd control, and procedures for reporting potential fire hazards observed when touring the facility.

13.2.2.10 Management and Supervisory Training

Seabrook Station conducts programs that provide position-specific training on the knowledge and skills required for effective management and supervision. These programs are described below.

a. Management and Supervisory Personnel Training Program

The management and supervisory training program provides instruction to meet the different needs of each management and supervisory level at Seabrook Station. Participants in the program include both licensed and nonlicensed members of management. Major features of the management and supervisory training program are listed below:

- Instruction within the program covers the knowledge and skills needed for effective management or supervision for directors, managers, supervisors, and first line supervisors.

- The program provides training on the special needs of new supervisors during the first few months of their work.

Management and supervisory training includes problem analysis, decision analysis, potential problem analysis, written and oral communications, behavior observation, safety, team-building, management techniques, and business fundamentals. Detailed information about the program is provided in the program description.

Personnel assigned to the shift manager training program mentioned in Subsection 13.2.1.3 also receive this training.

b. Maintenance Supervisory Personnel Training Program

The maintenance supervisory personnel training program provides initial and continuing training for supervisory personnel. These personnel receive the knowledge and skills needed to independently perform supervisory duties in a manner that promotes safe and reliable plant maintenance and operation. This program supplements training received in the management and supervisory personnel training program.

The program applies to supervisory personnel in I&C, mechanical and electrical maintenance, as well as contractor supervisory personnel under the supervision of the Maintenance Department.

13.2.2.11 Plant Access Training (PAT)

Plant Access Training (PAT) is provided to all personnel requiring unescorted access to the protected area. This training assumes a new employee has no familiarity with nuclear power plants, and the training is followed by an exam. Instruction covers site familiarization, fitness for duty, escort procedures, security, radiation protection, industrial safety including fire safety, the radiological emergency plan, and quality assurance. All who require unescorted access to a Radiologically Controlled Area (RCA) must complete training in radiation protection, including a practical factors check-out.

13.2.3 Retraining

Nonlicensed personnel receive retraining and training updates in the following subjects:

Procedures (as appropriate)

Radiation Protection

Security

Radiological Emergency Plan

Fire Safety

Safe Work Practices

Skills Refresher

Industry and Plant Experience

Refresher training is scheduled on a periodic basis depending on the volume of the material to be taught.

13.2.4 Position Job Analyses

A position job analysis determined training needs from the INPO generic task lists and from information provided by station departments. These two sources were used to develop site-specific task lists that are revised periodically to reflect changes in job activities.

13.2.5 Program Evaluation

Testing accompanies most knowledge- and skills-based training in the form of performance and written exams. Training procedures/instructions provide for the regular evaluation of programs to ensure training effectiveness.

13.3 EMERGENCY PLANNING

A comprehensive Radiological Emergency Plan for Seabrook Station is provided as a separate volume to this application. This plan is maintained and controlled as a separate document, and is updated in accordance with the requirements of 10 CFR 50.54(q).

The Seabrook Station Radiological Emergency Plan was developed in accordance with the requirements of Paragraph 50.34(b), 50.47(b) and Appendix E to Title 10 of the Code of Federal Regulations Part 50, "Licensing of Production and Utilization Facilities," and with 10 CFR 72.32(c) and (d).

The radiological analysis information required by Regulatory Guide 1.70 (Rev. 3), which is not part of the Seabrook Station Radiological Emergency Plan, is contained in Appendix 13A.

13.4 REVIEW AND AUDIT

Operating phase activities that affect nuclear safety are reviewed and audited through a comprehensive program. The review and audit program assures proper review and evaluation of proposed changes, tests, experiments and unplanned events. ASME NQA-1, 1994 forms the basis for the program.

13.4.1 Onsite Review

13.4.1.1 SORC

A Station Operation Review Committee (SORC) performs the onsite operational review responsibilities. The purpose of the SORC is to advise the station management on all matters related to nuclear safety. The function, composition, meeting frequency, responsibilities and authority of the SORC are contained in Appendix A of the FPL Quality Assurance Topical Report (QATR).

13.4.1.2 Operations Phase Reviews

The scope of SORC operational phase review is specified in Appendix A of the FPL QATR.

13.4.2 Audit Program

Audits are normally performed under the quality assurance audit program described in the FPL Quality Assurance Topical Report.

13.5 PLANT PROCEDURES

All station procedures and any changes to them are reviewed and approved by appropriate station supervisory and management personnel in accordance with the FPL Quality Assurance Topical Report.

In the event of an emergency condition, which could likely affect the health and safety of the public, if not promptly corrected, the Plant General Manager or his designated alternates may authorize emergency repairs and activities that deviate from written procedures. When such emergency is undertaken, the nature of the emergency, its cause, the emergency corrective action and justification therefore shall be documented and submitted for review and approval as required for temporary procedures.

13.6 INDUSTRIAL SECURITY

A description of the physical security program for Seabrook Nuclear Generating Station Unit 1 has been provided to the NRC as a separate part of the application withheld from public disclosure pursuant to paragraph 2.390 (d), 10 CFR Part 2, Rules of Practice.

APPENDIX 13A RADIOLOGICAL ANALYSIS

An analysis has been performed to provide curves for thyroid doses of 5, 25, 150 and 300 rem and whole body ($\beta + \gamma$) doses of 1, 5 and 25 rem. Each curve represents a conservative estimate of the elapsed time to reach the specified dose level as a function of distance from the release point under the conditions postulated for the release of radioactivity for the design bases loss-of-coolant accident (Subsection 15.6.5.4). These curves do not represent dose as a function of time and distance (x) for any one discrete circumstance, but rather the locus or envelope of highest values of dose that would be expected under all wind speed conditions. The doses have been maximized by imposing the concept of an optimal or worst wind speed ($\mu = \frac{2x}{T}$)such that the radioactive plume front transit time ($\frac{x}{\mu}$) is equal to the actual exposure time ($T - \frac{x}{\mu}$), the sum of the two being the total elapsed time from the onset of release (T). Such an assumption can, however, result in plume velocities that are unrealistically low particularly for short distances. In order to be consistent with Chapter 15, plume velocities (wind speeds) have been bounded to a lower limit of 1.78 meters sec^{-1}. This lower limit for wind speed has been determined from the calculated EAB LOCA thyroid dose and 1 hour χ/Q value used in the Chapter 15 design bases accident analysis (i.e., a wind speed of 1.78 m/s in conjunction with Pasquill stability Class F assumptions, results in the 2-hour thyroid dose as calculated in Chapter 15, Subsection 15.6.5.4).

Doses have been calculated using the dose conversion factors from Regulatory Guide 1.109, "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." The following equation has been used to calculate offsite thyroid and whole body ($\beta + \gamma$) doses:

$$D = \frac{F(x)}{\mu}\left[T - \frac{X}{\mu}\right]\frac{D_o}{T_o\,(\chi/Q)_{x_o}}$$

Where μ = $\frac{2x}{T}$, μ min. = 1.78 (meters sec^{-1})

x = Distance from release point to receptor (meters)

T = Total elapsed time from onset of release (sec)

F(x) = $1/\pi\sigma_y\,\sigma_z$, Pasquill stability Class F for distance x

D_o = EAB calculated two-hour doses for LOCA releases (52 rem thyroid and 3.8 rem whole body)

T_o = 2 hours (7200 sec)

$(\chi / Q)_{x_o}$ = 2.67×10^{-4} sec/m^3, EAB conservative case one hour dispersion coefficient

D = Dose (rem) at distance x, for time period T

Plots showing projected ground-level doses for stationary individuals as a function of time and distance are presented in Figure 13A-1 and Figure 13A-2.

SEABROOK STATION
UPDATED FINAL SAFETY ANALYSIS REPORT

CHAPTER 13
CONDUCT OF OPERATIONS

TABLES

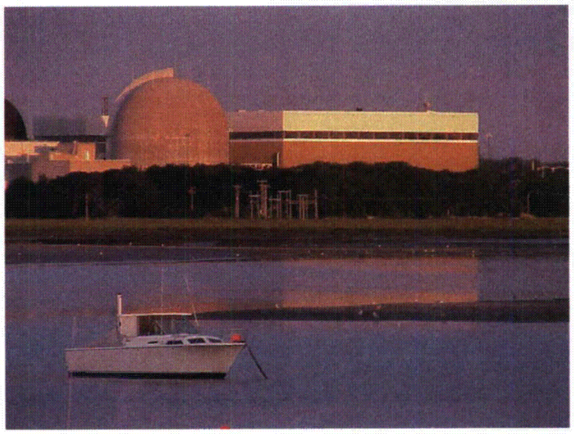


TABLE 13.5-1 Deleted

TABLE 13.5-2 CONTROL ROOM OPERATING PROCEDURES

Listed below are the categories of Operating Procedures which are used by licensed operators in the control room to perform various activities and plant evolutions. The descriptions are given to indicate the scope of procedures which cover the significant operations listed by Regulatory Guide 1.33, Appendix A, where appropriate.

1. General Plant Operating Procedures

   a. Major Plant Evolutions

      COLD SHUTDOWN to HOT STANDBY; HOT STANDBY to minimum load; approach to criticality; power increase and decrease; minimum load to hot standby; hot standby to cold shutdown; post trip review.

   b. Turbine Operations

      Turning gear, starting, phasing, shutdown; electro-hydraulic and standby control operations.

   c. Refueling Operations

      Refueling cavity fill, purification, and drain; operations of refueling machine, Fuel Transfer System, upender, fuel pit bridge, RCCA change machine, and various handling tools; containment integrity and closeout.

2. Procedures for Startup, Operations, and Shutdown of Safety-Related Systems

   Instructions for energizing, filling, venting, draining, startup, shutdown, and changing modes of operation have been prepared, as appropriate, for the following systems:

   a. Reactor Coolant System

      RCS evacuation, fill, vent, and drain tank operation; RCP operation; pressurizer bubble formation; reactor makeup water fill, vent, and operation; pressurizer relief tank operation.

   b. Control Rod Drive System

      Rod control - Automatic and manual; MG set operation.

   c. Shutdown Cooling System (RHR)

      System startup, operation, and shutdown.

   d. Emergency Core Cooling System

      SI operation; inadvertent containment isolation; spray additive tank and containment spray operations.

   e. Component Cooling Water System

      PCCW, SCCW, and thermal barrier fill and vent; PCCW, SCCW, and thermal barrier startup, operation, and shutdown; PCCW makeup and chemical addition.

f. Containment

Containment enclosure ventilation; containment air purge; containment online purge.

g. Atmosphere Cleanup System

Containment ventilation system operation

h. Fuel Storage Pool Purification and Cooling System

Spent fuel pool cooling and purification system fill, vent, and operation.

i. Main Steam System

Main steam, MSR, and auxiliary steam reducing station operations; extraction steam and turbine seal steam operations.

j. Pressurizer Pressure and Spray Control System

Heater and spray operations contained in CVCS procedures.

k. Feedwater System

Feedwater fill and vent; main feed and startup feed pumps startup, operation, and shutdown.

l. Auxiliary (Emergency) Feedwater System

Aligning and restoring EFW for auto-initiation.

m. Service Water System

System fill, vent, and operations; cooling tower operation including heating and deicing.

n. Chemical and Volume Control System

Fill, vent, and operation of charging, letdown, and seal injection; operation of excess letdown, letdown degasifier, demineralizers, filters, Pressurizer Level Control System, makeup, and boric acid subsystems; establishing cover gases.

o. Auxiliary Building (and Other Safety-related) Heating and Ventilation

Heating and ventilation for the following areas: Service Water Pumphouse and Cooling Tower, Primary Auxiliary Building, Diesel Generator Building, Fuel Storage Building, Emergency Feedwater Pumphouse.

p. Control Building Heating and Ventilation

Heating, cooling (where applicable), and ventilation for the following areas: control room, cable spreading area, emergency switchgear area, and electrical.

q. Radwaste Building Heating and Ventilation

Heating and ventilation for the Waste Processing Building

r. Instrument Air System

Operations of the Compressed Air and Containment Compressed Air Systems.

s. Electrical Systems

(1) Offsite (access circuits)

Operation of main generator, unit auxiliary, and reserve auxiliary transformer auxiliaries.

(2) Onsite

(a) Emergency Power

Operations of battery chargers, diesel generators and auxiliaries.

(b) AC Systems

Operations of 13.8 kV, 4.16 kV, 480V, and 120V; operation of lighting systems.

(c) DC Systems

Operations of 125V DC systems and inverters; operation of emergency lighting.

t. Nuclear Instrument System

NIS switch alignment; NIS visual/audio count rate system operations.

u. Reactor Control and Protection System

See Reactor Coolant System and Control Rod Drive System.

v. Hydrogen Recombiner

Operations of hydrogen analyzer and hydrogen recombiner.

3. Procedures for Abnormal Conditions

Loss of refueling cavity water, letdown, charging, RHR during shutdown, vacuum, instrument and containment air, control room makeup air, and plant computer; malfunctions of reactor coolant pump, control rod drives, rod position indication, PCCW, Service Water Cooling Tower, turbine generator; RC leakage, condenser leakage, fire system break, oil spill; response to fire, rapid boration, RC high activity and chemistry out of specification, steam generator blowdown; safe shutdown, severe weather conditions.

4. Emergency Operating Procedures

   a. Emergency Response Procedures

   Reactor trip, safety injection, and rediagnosis; natural circulation cooldown scenarios; loss of primary or secondary coolant scenarios and recoveries; steam generator tube rupture scenarios and recoveries.

   b. Emergency Contingency Actions

   Loss of AC power and recovery scenarios; loss of coolant outside containment and loss of emergency coolant recirculation; steam generator depressurization and tube rupture.

   c. Functional Restoration Procedures

   Responses to nuclear power generation/ATWS, and loss of core shutdown; responses to inadequate, degraded, and saturated core cooling; responses to loss of secondary heat sink, steam generator over pressure, steam generator high and low level, loss of normal steam release capabilities; response to imminent or anticipated pressurized thermal shock; responses to high containment pressure, containment flooding, or containment high radiation; responses to high or low pressurizer level and voids in reactor vessel.

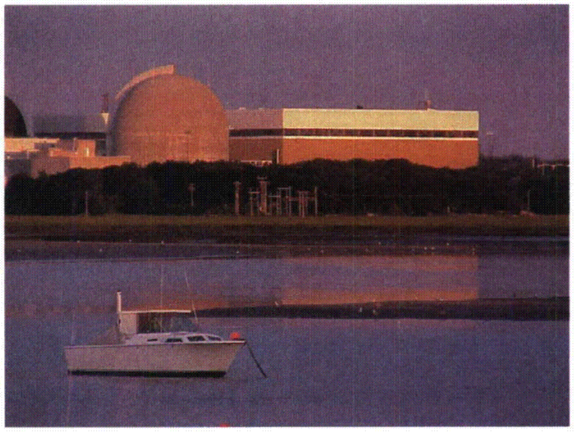
   d. Critical Safety Functions (Status Trees)

   Subcriticality, core cooling, heat sink, integrity, containment, inventory.

SEABROOK STATION
UPDATED FINAL SAFETY ANALYSIS REPORT

CHAPTER 13
CONDUCT OF OPERATIONS

FIGURES



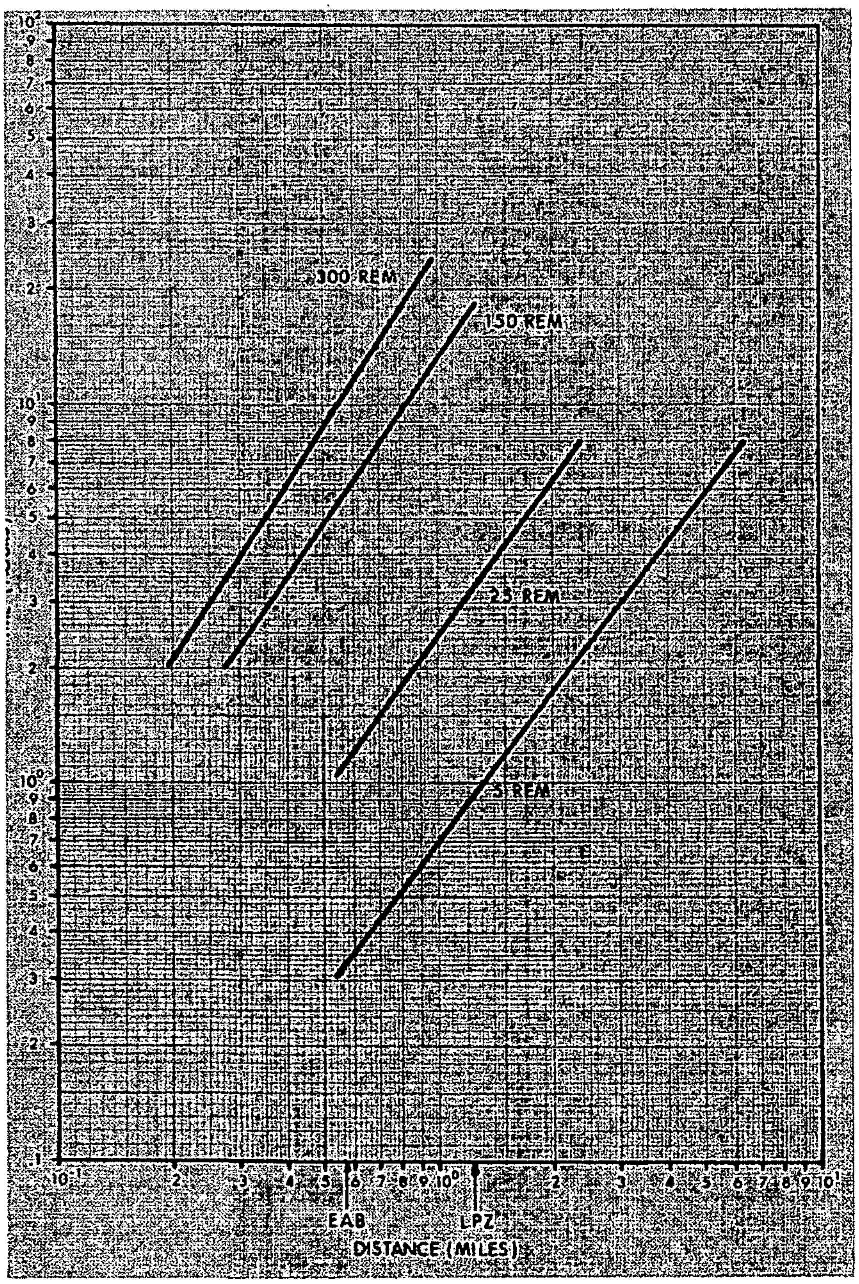
Deleted

Deleted

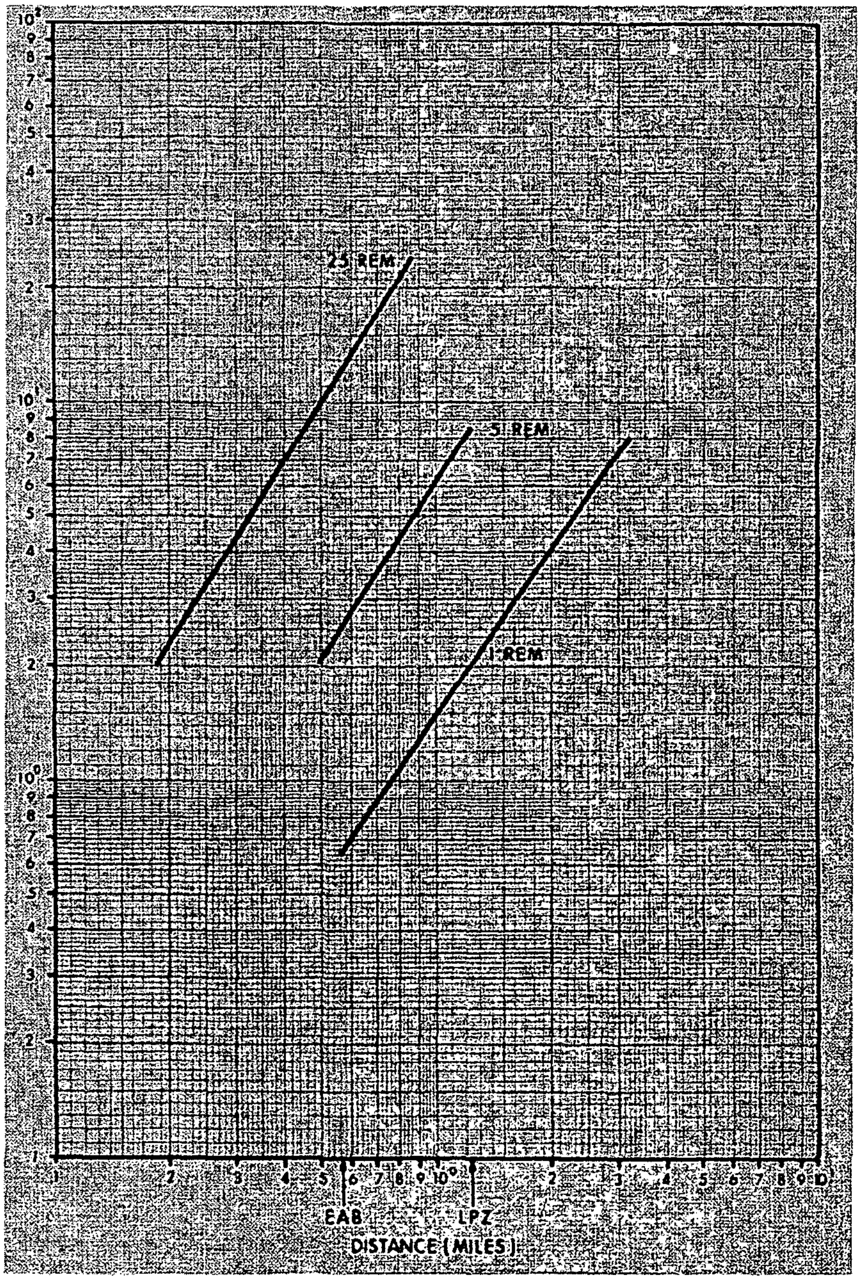
Deleted

Deleted

See 1-NHY-500090



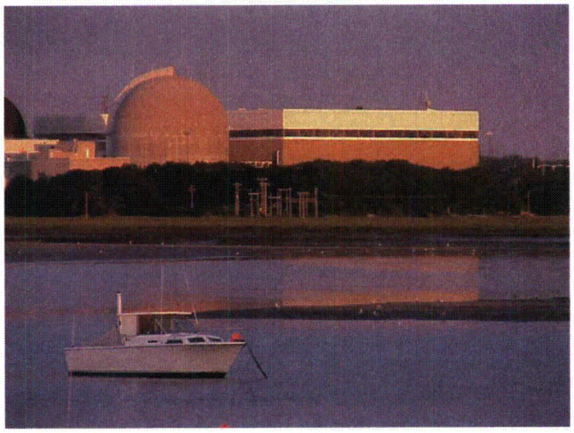
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Thyroid Dose Time vs. Distance LOCA Releases
	Figure 13-A-1



SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Whole Body Dose vs. Distance LOCA Releases	
		Figure 13-A-2

SEABROOK STATION
UPDATED FINAL SAFETY ANALYSIS REPORT

CHAPTER 14
INITIAL TEST PROGRAM



14.1 SPECIFIC INFORMATION TO BE INCLUDED IN PSAR

Not applicable to Updated FSAR.

14.2 SPECIFIC INFORMATION TO BE INCLUDED IN UPDATED FSAR

14.2.1 Summary of Test Program and Objectives

A comprehensive initial test program was conducted at Seabrook Station to demonstrate that plant systems, structures, and components performed in a manner that did not endanger the health and safety of the public. The principle objectives of this program are to provide, to the extent practical, assurance of the following:

a. The plant has been properly designed and constructed and is capable of operating safely at performance levels specified in the Updated FSAR

b. The plant operating and emergency procedures have been verified by trial use to be adequate

c. The plant operating and technical personnel are knowledgeable about the plant equipment and procedures and are prepared to operate the facility in a safe manner.

The initial test program included a preoperational test phase and an initial startup test phase. Preoperational testing consisted of individual system and integrated system tests performed prior to (and in some cases after) initial core load on essentially completed systems and structures. These tests demonstrated, to the extent practical, the capability of systems, structures, and components to meet performance requirements.

Initial startup testing consisted of those single and multi-system activities scheduled to be performed during and following fuel loading. This included precritical tests, initial criticality, low-power tests, and power ascension tests. This testing demonstrated that the plant will operate in accordance with design and the ability of the plant to respond properly to anticipated transients.

14.2.2 Organization and Staffing

The Startup Test Department managed and provided overall direction for the initial program. The Startup Test Department consisted of personnel assigned to the plant site with specialties in areas such as primary systems, secondary systems, electrical systems, and plant operations. These individuals were assigned overall responsibility for various aspects of the test program within their areas of expertise. During the performance of system preoperational tests and initial startup tests, the Startup Test Department personnel directed plant operations personnel during test activities and were responsible for the acquisition, review and evaluation of relevant data.

Table 14.2-1 is a responsibility/authority matrix showing the various organizations involved with each portion of the Seabrook initial test program.

A definition of each of the major responsibilities is provided to clarify its specific intent. This table also presents the organizations responsible for the preparation, review and approval of Preoperational, Acceptance, Startup and Special Test procedures. The responsible design organizations or vendors provided technical support, as requested by their respective onsite organizations, and either reviewed or specified the acceptance criteria used in these test procedures. The interrelationship of the various organizations during testing activities is discussed in Subsections 14.2.4 and 14.2.5.

To ensure a comprehensive overview of the preoperational test program by the appropriate organizations, a Joint Test Group (JTG) was formed consisting of site representatives of the Startup Test Department, Seabrook Station Operations Staff, and the Nuclear Services Division (YNSD) of Yankee Atomic Electric Company (YAEC). The Startup Manager acted as chairman of the Joint Test Group and had final responsibility for approval of test procedures and test results. When necessary, personnel from other organizations were invited to attend the meetings of the JTG for information, coordination, or technical advice. The Nuclear Steam Supply System vendor (Westinghouse), the Architect-Engineer (UE&C), and Construction Manager (UE&C) provided technical assistance in their areas of specialty as required throughout the test program.

The JTG was responsible for the following activities:

a. Review and approval of preoperational test procedures

b. Review and approval of changes to preoperational test procedures

c. Review and approval of the results of preoperational tests.

At the time of the start of initial fuel loading, the JTG was dissolved and the Station Operations Review Committee (SORC) assumed the responsibilities stated above during the initial startup testing. During this portion of the program, the appropriate vendor and design organizations provided technical assistance during the initial procedure technical review by the Startup Test Department.

All personnel authorized to direct testing during the test program and to approve the procedures used in these tests were appropriately qualified in accordance with the requirements of Regulatory Guide 1.58 (Revision 1, 9/80) as further clarified in Section 1.8. Personnel authorized to direct preoperational and startup tests (Phases 2 through 6) also met the additional requirements of a Bachelor's Degree in Engineering or related science with a minimum of one year's experience acquired in testing, operation, and maintenance of nuclear power generating facilities for the direction of preoperational tests and a minimum of two years' experience for the direction of startup tests. For personnel who did not possess the formal education, this requirement was waived if other factors provided sufficient demonstration of ability. Personnel assigned to the Startup Test Department also received additional training in the administration and requirements of the test program. The qualifications of the station operating and technical staff are discussed in Section 13.1.

14.2.3 Test Procedures

The initial test program was conducted using written procedures for each individual test. Tests of systems and equipment performed prior to (or in some cases after) initial core load are designated as either Preoperational Tests (PT) or Acceptance Tests (AT). Preoperational Tests are subdivided into either of the following categories:

a. Individual systems tests demonstrate the proper operation of plant systems and equipment which perform a safety-related function.

b. Integrated systems tests involve the integrated operation of plant systems and equipment to demonstrate or verify a safety-related function.

Acceptance Tests demonstrated the proper operation of nonsafety-related plant systems and equipment.

Tests performed as part of or subsequent to loading of fuel into the reactor core were designated as Startup Tests (ST). In addition, Special Test Procedures (STP) not in the original scope of the test program were used for situations that required the performance of a test for investigative or data collection purposes.

Each test specified above contained as a minimum, the following sections:

a. Test Objectives

b. Prerequisites

c. Special Precautions

d. Initial Conditions (including environmental)

e. Test Instructions

f. Final Conditions

g. Acceptance Criteria.

The Test Instructions section of the test provided data blanks or reference data sheets which specifically identified the data recorded in each test. Means were provided to identify the individuals who witness or record data during each test and the instrumentation used for data collection. Administrative procedures were provided to specify proper methods for collection and retention of test data.

Table 14.2-1 shows the organizations responsible for the preparation, review and approval of Preoperational, Acceptance, Startup and Special Test procedures. The responsible design organizations or vendors provided technical support, as requested by their respective onsite organizations, and either reviewed or specified the acceptance criteria used in these test procedures.

14.2.4 Conduct of the Test Program

The preoperational test program was administered in accordance with the Preoperational Test Program Description which was prepared by the Startup Test Department and approved by the Joint Test Group participating organizations. Where necessary, due to certain unique activities associated with testing, administrative procedures were prepared by the Startup Test Department, reviewed by the Joint Test Group, and the Startup Manager had final responsibility for approval. Otherwise, station administrative procedures were used as applicable during the initial test program.

The initial startup program was administered in accordance with a startup procedure which was prepared by the Startup Test Department and approved by the Station Operations Review Committee, with the Station Manager having final approval responsibility. Normal station administrative procedures were used during the initial startup program.

Prior to the performance of a system preoperational or acceptance test, a test engineer (or engineers) was assigned by the Startup Test Department to direct the test. For startup tests, Startup Test Department engineers or appropriately qualified station staff technical personnel were assigned test director responsibility. These individuals were responsible for ensuring that prerequisites were completed, precautions complied with and initial conditions established. They then directed the station operating personnel in the performance of the test and assured all applicable data was recorded. Station operating personnel were responsible for the safe and proper operation of the plant and its associated equipment throughout the test program. The Shift Supervisor took whatever action necessary including, but not limited to, stopping any test and placing plant equipment in a safe condition.

All field changes to preoperational and acceptance test procedures were approved by the Shift Test Director prior to performance. The JTG reviewed all such field changes within fourteen days of implementation. All changes to startup test procedures were approved in accordance with Technical Specification requirements.

All plant modifications which were initiated as a result of system pre-operational or acceptance tests were controlled in accordance with the procedure for modifications during plant construction. Any such modifications or repairs were retested to the requirements of the test procedure. Subsequent to the completion of the system preoperational test, all modifications or repair activities were performed and retested in accordance with the normal station administrative procedures for modifications or maintenance as applicable.

14.2.5 Review, Evaluation and Approval of Test Results

Upon completion of each preoperational, acceptance, or startup test, the responsible test engineers reviewed the test data for completeness, performed any evaluations or calculations required, and compared the results to the stated acceptance criteria. Any unresolved or incomplete items, including acceptance criteria, were described on a summary list of test exceptions. The test results then were submitted to the Joint Test Group or Station Operations Review Committee, as applicable, for test result review. Upon satisfactory review by the Joint Test Group or Station Operations Review Committee, the test results were approved by the Startup Manager or the Station Manager.

Prior to the start of fuel loading, a final review was made by the Joint Test Group of the preoperational test program to ensure all required preoperational and acceptance tests were conducted and test results approved.

If during the course of the preoperational test program, it was necessary to delay a portion of a preoperational test, such tests were incorporated into the startup test program if adequate justification was present for delaying the test beyond core load. At this time, only AT-17, waste solidification system test, may be performed subsequent to core loading. This may be required if a permanent Waste Solidification System is not designed and installed at the time of fuel loading. This system was tested subsequent to installation independent of the startup program.

Prior to the start of each major phase of the initial startup program identified in Table 14.2-2, the Station Operations Review Committee performed a preliminary review of all prerequisite testing to ensure that it was satisfactorily completed to the extent necessary to perform the next phase of the startup program. The committee used a prerequisite list which was approved prior to the start of any test in the subsequent phase of testing.

14.2.6 Test Records

A copy of all Preoperational Tests, Acceptance Tests, Startup Tests, Special Test Procedures, and all relevant data recorded during the conduct of the tests will be maintained for the life of the station in accordance with station procedures for record retention.

14.2.7 Conformance of Test Programs with Regulatory Guides

The Regulatory Guides listed below were followed, to the degree indicated, during the conduct of the Seabrook Station initial test program.

Regulatory Guide 1.8, Rev. 1-R
Personnel Selection and Training

The initial personnel selection and training program met the requirements of Regulatory Guide 1.8 (1977 edition). For current status, see Sections 13.1 and 13.2.

Regulatory Guide 1.20, Rev.2
Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing

The Westinghouse position on Regulatory Guide 1.20, Rev. 2, is discussed in Subsection 3.9(N).2.4.

Regulatory Guide 1.33, Rev. 2
Quality Assurance Program Requirements (Operation)

The quality assurance program for operation complies with the requirements of this Regulatory Guide. For further discussion, see Section 17.2.

Regulatory Guide 1.41, Rev. 0
Preoperational Testing of Redundant Electric Power Systems to Verify Proper Load Group Assessments

Seabrook Station conforms with the recommendations of Regulatory Guide 1.41.

Regulatory Guide 1.52, Rev.2
Design, Testing and Maintenance Criteria for Engineered Safety Feature Atmosphere Cleanup System Air Filtration and Absorption Units of Light Water Cooled Nuclear Power Plants

A detailed discussion on the degree of conformance to Regulatory Guide 1.52 is found in Subsection 6.5.1.

Regulatory Guide 1.68, Rev. 2
Initial Test Programs for Water-Cooled Nuclear Power Plants

The initial test program for Seabrook Station was conducted in accordance with the intent of Regulatory Guide 1.68 except for the items specified below:

a. During the preoperational test program, no practical method existed to vary system voltage to obtain maximum and minimum design voltages. The intent of the requirement to demonstrate that the emergency loads can start and operate with the maximum and minimum design voltage available was met by testing the emergency loads under plant light load conditions to simulate the maximum practically obtainable voltage and under plant heavy load conditions to simulate the minimum practically obtainable voltage. The results of this testing were compared to the station voltage study to verify the adequacy of the analytical model (Appendix A, Subsection 1.g.2).

b. During the power ascension testing phase, tests were scheduled so that the safety of the plant was not dependent on the performance of an untested system or feature. Power ascension testing was performed at power plateaus of approximately 30%, 50%, 75% and 100%. It was required that testing be performed at 30% rather than 25% because individual system stability is increased at 30% (e.g., Feedwater System). This allows comparison of steady-state conditions with the design at low power. Westinghouse-supplied plants have historically conducted tests at 30% and, therefore, generic data was available for review and comparison.

c. Throughout core loading and precritical tests, the shutdown margin was verified by periodic sampling of core coolant and verification that boron concentration was maintained at or above the Technical Specification concentration limit for refueling conditions (Appendix A, Section 2.a).

d. Control rod runback and partial scram features are not used in the Seabrook Station design and, therefore, were not tested during power escalation (Appendix A, Section 5.j).

e. A demonstration of the capability of systems and components to remove residual heat or decay heat from the Reactor Coolant System was performed during power ascension testing only if not performed during hot functional or low power tests (Appendix A, Section 5.1).

f. The failed fuel detection system is not applicable to the Seabrook design and, therefore, was not tested during power escalation (Appendix A, Section 5.q).

g. The integrated control system and the Reactor Coolant Flow Control System are not applicable to the Seabrook Station design and therefore, were not tested during power escalation. The Startup and Emergency Feedwater Control Systems and the Steam Pressure Control Systems are only used in the hot shutdown, hot standby or low power operating modes. These systems cannot be tested during power ascension (Appendix A, Section 5.s).

h. A demonstration of the dynamic response of the plant to a loss of or bypassing of a feedwater heater(s) was not performed. As shown in Subsection 15.1.1, the transient resulting from the most severe case of feedwater temperature reduction initiated by a single failure or operator error is similar to, but of a lesser magnitude than the excessive load increase (load swing). The load swing test was performed at several major plateaus.

i. As shown in Subsections 15.2.3 and 15.2.4, dynamic response of the plant to a MSIV closure is bounded by the response of the plant to the turbine trip event. Plant response to a turbine trip was demonstrated during performance of ST-38, unit trip from 100 percent power (Appendix A, Subsection 5.m.m).

j. The ability of the movable incore neutron flux instrumentation to detect control rod misalignments was not performed. The excore neutron flux instrumentation is not designed to detect a local condition such as a misaligned RCCA, but rather a more global anomalous core condition. The movable incore flux instrumentation is not intended to specifically detect a misaligned control rod, but may be able to confirm an RCCA misalignment initially detected by the Rod Position Indication System. The individual Rod Position Indication System is the primary means for detecting RCCA misalignments. Since at 100 percent power the control rods are essentially withdrawn, individual rod worth is such that the ability of the movable incore instrumentation to detect a control rod misalignment is limited. The basis for this deletion is that the original requirement was imposed to demonstrate alternative instrumentation capabilities in detecting a misaligned control rod. With the advent of the Digital Rod Position Indication (DRPI) System, the need for accurate and sensitive alternative indications has been essentially eliminated. In any case, the distribution and number of the incore and excore flux instrumentation has not been changed and is identical to all Westinghouse four-loop plants since Indian Point 2. Since that time, the capability and sensitivity of the excore and incore flux instrumentation has been demonstrated numerous times (Appendix A, Section 5.1).

k. Vibration levels of the Reactor Coolant System and piping reaction to transient conditions are measured during hot functional testing (Appendix A.2.f).

l. Evaluation of rod scram times for scrams that occur during power ascension was not performed since no practical method for obtaining this data exists for a Westinghouse PWR (Appendix A, Section 5.h).

m. The static rod drop test was not performed at Seabrook. Performance of this test at other facilities has resulted in abnormally high power tilts and large xenon oscillations and may increase the risk of fuel failure. Performance of this test at plants similar to Seabrook has provided ample data to demonstrate that Westinghouse computer codes are able to adequately predict core thermal and nuclear parameters for RCCA misalignments up to and including full insertion of a single high worth rod. In addition, following performance of this test at Catawba, INPO has recommended that utilities delete this test from their startup programs (Appendix A, Section 5.f).

n. The pseudo-rod ejection test was not performed at greater than 10% power at Seabrook. Performance of this test may result in violation of the Technical Specification limits on peaking factor. Since the accident analysis for Seabrook shows the at-power ejected rod worth and power peaking factors are bounded by the zero power case, the calculation model was verified during the pseudo-rod-ejection test at zero power (Appendix A, Section 5.e).

o. Control rod scram times were measured at hot full-flow conditions only. Testing at no-flow conditions would require unnecessary and undesirable cycling of the reactor coolant pumps to be in compliance with the Technical Specifications. In addition, the hot full-flow condition is the limiting condition required by Technical Specifications (Appendix A, Section 2.b).

<u>Regulatory Guide 1.79, Rev.1</u>
<u>Preoperational Testing of Emergency Core Cooling Systems for Pressurized Water Reactors</u>

The initial test program for Seabrook Station was conducted in accordance with the intent of Regulatory Guide 1.79 except for the following:

a. Subsection C.1.c.(2) specifies that an opening test of the accumulator isolation valves be performed at the maximum differential pressure that the valve will experience using both normal and emergency power supplies. Since the valve operational capability is independent of the source of power and the valve motors are a small fraction of the rating of the emergency power supply, the valves were cycled at maximum differential pressure using the normal power supply only.

b. Subsection C.1.a.(2) specifies that a flow test at hot operating conditions be initiated by actuation of the safety injection signal. Since the intent of the test is to verify the ability to deliver cooling water to the vessel under simulated accident conditions, manual actuation of pump operation is considered sufficient to initiate the system and will permit rapid termination of the injection to minimize thermal shock effects. If a safety injection actuation signal is used, the pumps cannot be readily de-energized due to circuit time delays, resulting in a greater thermal shock to the primary system. Integrated system response to an actuation signal was demonstrated in other tests.

<u>Regulatory Guide 1.80, Rev. 0</u>
<u>Preoperational Testing of Instrument Air Systems</u>

The initial test program for Seabrook Station was conducted in accordance with the intent of Regulatory Guide 1.80 except for the following:

a. The loss-of-instrument-air supply test described in Sections C.8, C.9 and C.10 was performed only on the portion of the Instrument Air System which performs a safety-related function and uses instrument air in the performance of that function.

b. Section C.11 requires that the results of preoperational testing of the instrument air system be included in the startup report. This requirement is in conflict with the reporting requirements of Section C.9 of Regulatory Guide 1.68, Rev. 2. The guidelines contained in Regulatory Guide 1.68 were followed for handling the results of the instrument air preoperational testing.

Regulatory Guide 1.108, Rev. 1
Periodic Testing of Diesel Generators Used as Onsite Electric Power Systems at Nuclear Power Plants

Seabrook Station is generally in conformance with Regulatory Guide 1.108.

The detailed discussion on this guide is found in Section 1.8.

Regulatory Guide 1.128, Rev. 1
Installation Design and Installation of Large Storage Batteries for Nuclear Power Plants

Seabrook Station is generally in conformance with Regulatory Guide 1.128.

The detailed discussion on this guide is found in Subsections 8.3.2 and 8.3.3.

Regulatory Guide 1.140, Rev. 1
Design, Testing and Maintenance Criteria for Normal Ventilation Exhaust System Air Filtration and Adsorption Units of Light Water-Cooled Nuclear Power Plants

A discussion of the degree of conformance to this regulatory guide is found in Section 1.8. Initial testing of the applicable filtration systems was in accordance with the recommendations contained in this regulatory guide.

14.2.8 Utilization of Reactor Operating and Testing Experiences in Development of the Test Program

The Startup Test Department performed a survey of PWR operating experiences, encompassing approximately similar power plants over at least the two previous years. The survey identified operating problem areas or categories of abnormal occurrences that are repeatedly being experienced by other facilities. This information was incorporated appropriately into the Seabrook Startup Program.

14.2.9 Trial Use of Plant Operating and Emergency Procedures

The procedures used to conduct the preoperational test program referenced the station operating, emergency and surveillance procedures whenever possible. During initial startup, plant operating and emergency procedures were used almost exclusively to operate the plant and its systems. Whenever corrections to station procedures were identified during testing, the corrections were evaluated and the procedures revised accordingly.

A description of station procedures is provided in Section 13.5.

14.2.10 Initial Fuel Loading and Initial Criticality

The following describes the general approach used to prepare for and perform initial fuel loading and initial criticality. Detailed procedures prepared and approved in accordance with Table 14.2-1 governed the actual work activities.

14.2.10.1 Initial Fuel Loading

Initial fuel loading did not begin until all prerequisite system tests and operations were completed to the satisfaction of the Station Operations Review Committee.

Fuel handling tools and equipment were checked out and dry runs conducted in the use and operation of the tools and equipment.

The containment integrity was established as required by Station Technical Specifications for the refueling mode.

The reactor vessel and associated components were ready to receive fuel. Water level was maintained above the bottom of the nozzles and recirculation maintained to assure a uniform boron concentration. Boron concentration can be increased via emergency boration or the addition of borated water directly to the open vessel. The refueling cavity remained dry during initial fuel loading activities.

The overall responsibility and direction for initial fuel loading was exercised by the station staff. The loading was directly supervised by a Senior Licensed Operator having no concurrent duties. The process was directed from the operating floor of the containment structure. Procedures for the control of personnel access and the maintenance of containment security were established prior to fuel loading.

The as-loaded core configuration is specified as part of the core design studies conducted in advance and is not expected to change. In the event that mechanical damage occurs to a fuel assembly and no spare is available onsite, an alternate core loading scheme whose characteristics closely approximate that of the initially prescribed pattern will be determined.

The core was assembled in the reactor vessel. The fuel assemblies were submerged in reactor grade water containing enough dissolved boric acid to maintain a calculated core effective multiplication factor within Technical Specification limits. Core moderator chemistry conditions (particularly boron concentration) were prescribed in the core loading procedure and were verified at a specified frequency by chemistry sample and analysis.

Core loading instrumentation consists of two permanently installed source range (pulse type) channels and two temporary incore source range channels. A third temporary channel may also be used as a spare. The temporary channels are monitored in the containment structure. Both permanent channels can display the neutron flux level on a strip chart recorder. One permanent channel is also equipped with an audible count rate indicator; the temporary channels indicate on scalers with a minimum of one channel recorded on a strip chart recorder. Minimum count rates of one-half count per second, attributable to core neutrons, are required on at least two of the four available source range channels at all times following installation of the initial nucleus of eight fuel assemblies. A response check of the source range channels to a neutron source was performed within eight hours prior to the start of core loading (or resumption of loading if delayed for more than eight hours).

At least two artificial neutron sources were introduced into the core at appropriate specified points in the core loading program to ensure a detector response of at least one-half count per second attributable to neutrons.

Fuel assemblies, together with inserted components (control rod assemblies, burnable poison assemblies, neutron source assemblies, or thimble plugging devices), were placed in the reactor vessel one-at-a-time according to a previously approved sequence which was developed to provide reliable core monitoring during fuel loading activities. The core loading procedure documents included check sheets which specified and verified the sequential movement of each fuel assembly from its initial position in the storage racks to its final position in the core. Multiple checks were made of component serial numbers and types to guard against possible inadvertent exchanges or substitutions of components. Visual inspections were made to verify proper seating and orientation of fuel assemblies and components. Fuel assembly status boards were maintained throughout the core loading operation.

An initial nucleus of eight fuel assemblies, one of which contains a neutron source, is considered the minimum source-fuel nucleus which permits meaningful inverse count rate monitoring. This initial nucleus is determined by calculation to be markedly subcritical under the conditions of loading.

Each subsequent fuel addition is accompanied by neutron count rate monitoring to determine that the just-loaded fuel assembly does not excessively increase the count rate and that the extrapolated inverse count rate ratio is not decreasing for unexplained reasons. The results of each loading step were evaluated before the next sequential step was started. The final, as-loaded, core configuration was subcritical ($k_{eff} \leq 0.95$) under the required loading conditions.

Criteria for safe loading require that loading operations stop immediately if any of the below conditions exist:

a. An unanticipated increase in the neutron count rates by a factor of 2 occurs on all responding nuclear channels during any single loading step after the initial nucleus of eight fuel assemblies are loaded

b. An unanticipated increase in the count rate by a factor of 5 occurs on any individual responding nuclear channel during any single loading step after the initial nucleus of eight fuel assemblies are loaded

c. An unexplained decrease in boron concentration greater than 20 ppm from nominal as determined from samples of reactor coolant water.

An alarm in the Containment and control room is connected to the source range channels with an alarm setpoint equal to or less than five times the current count rate. This alarm automatically alerts the personnel performing loading operations that a high count rate condition exists and requires an immediate stop of all operations until the condition is evaluated. If the alarm is actuated during core loading, preselected personnel are permitted to remain in the Containment to evaluate the cause and determine further action.

In addition to the above, fuel loading procedures specify the condition of fluid systems to prevent inadvertent dilution of the reactor coolant boron concentration; define the means of fuel movement and handling to prevent the possibility of mechanical damage; define the criteria for stopping fuel loading, containment evacuation, and emergency boration; define the conditions that must exist for fuel loading to proceed; define responsibility and authority of personnel involved in the operation; and provide for fuel and core component accountability and status.

14.2.10.2 Initial Criticality

Upon completion of fuel loading, the reactor upper internals and the pressure vessel head were installed. Mechanical and electrical tests were performed on the control rod drive mechanisms. These tests included a complete operational checkout of the mechanisms and calibration of the individual Rod Position Indication System.

Tests are performed on the reactor trip circuits to verify manual trip operation of control assemblies. The control assembly drop times are measured for each control rod assembly at hot full flow conditions.

During control rod drive mechanism testing, the boron concentration in the coolant-moderator is maintained so that the shutdown margin requirements specified in the Technical Specifications are met. During individual control assembly or control bank movement, source range instrumentation is monitored for unexpected changes in core reactivity. A functional check is made of the moveable Incore Detector System and a leak test of the Reactor Coolant System is performed. Just prior to the approach to criticality, a functional test of the nuclear instrumentation is conducted, including verification that the high flux scram setpoint is set at a low value.

Initial criticality was achieved with the reactor at normal operating temperature and pressure by a combination of shutdown and control bank withdrawal and reactor coolant system boron reduction. Inverse count rate ratio monitoring, using data from the normal plant source range instruments, is used as an indication of the proximity and rate of approach to criticality. Inverse count rate ratio data are plotted as a function of rod bank position during rod motion and as a function of primary water addition during reactor coolant system boron concentration reduction.

Initially, the shutdown and control banks are withdrawn in the normal withdrawal sequence leaving the last withdrawn control bank inserted far enough in the core to provide effective control when criticality is achieved.

The boron concentration of the Reactor Coolant System is then reduced by the addition of primary water. Criticality is achieved during boron dilution or by subsequent rod withdrawal following boron dilution. The rate of primary water addition and, therefore, the rate of approach to criticality may be reduced as the reactor approaches criticality to ensure that effective control is maintained. Throughout this period, samples of reactor coolant are obtained and analyzed for boron concentration.

Written procedures specify the plant conditions, precautions and specific instructions for the approach to criticality and for limiting the period to more than thirty seconds once criticality is achieved.

14.2.11 Test Program Schedule

The initial test program consisted of a preoperational test phase and a startup test phase. The preoperational phase of testing of individual plant systems began after construction of the system was essentially complete and construction verification tests (hydrostatic tests, control circuits checks, etc.), system flushing, and preliminary system operational checks (instrument calibration, pump and motor operation, valve checks, etc.) were completed. Each system preoperational or acceptance test demonstrated, to the extent practical, the ability of the system and equipment to perform its design function in accordance with the Updated FSAR requirements.

The individual system preoperational and acceptance tests proceeded concurrently as individual system construction and preliminary testing was completed. When the appropriate systems were turned over to the station staff, integrated system preoperational tests were performed. The principal milestones during this phase of the program were the reactor vessel hydrostatic test and integrated hot functional tests. Tests of other systems were scheduled as appropriate to support these events. Subsection 14.2.12 provides more detailed information on each test performed during the preoperational phase of the program.

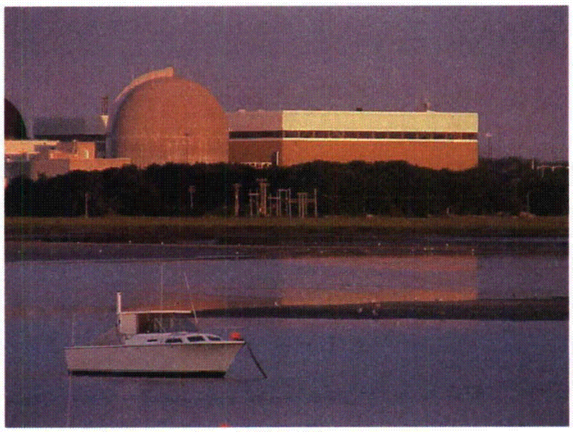
The startup test phase commenced at initial fuel loading. Initial fuel loading and initial criticality are discussed in Subsection 14.2.10. Subsequent to initial criticality, low power reactor physics tests were performed. During these tests, measurements were performed to verify the calculated values of control rod bank reactivity worths, isothermal temperature coefficients, and differential boron concentrations as a function of control rod configuration.

When the reactivity control characteristics of the reactor were verified by the low power tests, a program of power level escalation brought the unit to its full-rated power level. During the power escalation, pre-determined tests were conducted to verify that the reactor and unit perform as expected at 30%, 50%, 75% and 100% power. Tests were scheduled so that the safety of the plant was not dependent on the performance of an untested system or feature. Subsection 14.2.12 provides more detailed information on each test that was performed during low power testing and power escalation.

SEABROOK STATION
UPDATED FINAL SAFETY ANALYSIS REPORT

CHAPTER 14
INITIAL TEST PROGRAM

TABLES



Test procedures used during the preoperational test phase were available for review by NRC regional personnel approximately 60 days prior to their use. Startup test procedures were available for review approximately 60 days prior to fuel load.

14.2.12 Individual Test Descriptions

Included in this section are test abstracts for individual tests which were conducted during the initial test program to verify the performance capabilities of structures, systems, and components that:

a. Are used for shutdown and cooldown of the reactor under normal plant conditions and for maintaining the reactor in a safe condition for an extended shutdown period

b. Are used for shutdown and cooldown of the reactor under transient (infrequent or moderately frequent events) conditions and postulated accident conditions and for maintaining the reactor in a safe condition for an extended shutdown period following such conditions

c. Are used for establishing conformance with safety limits or limiting conditions for operation that are included in the facility Technical Specifications

d. Are classified as Engineered Safety Features or that will be relied upon to support or ensure the operation of Engineered Safety Features within design limits

e. Are assumed to function, or for which credit is taken in the accident analysis of the facility, as described in the Updated FSAR

f. Are used to process, store, control, or limit the release of radioactive materials.

It is anticipated that changes may occur to these abstracts as the test program develops. As a result, tests or portions of tests described by particular abstracts may be combined or divided into smaller test segments when the detailed test procedures are written.

Table 14.2-3 provides an index of the Preoperational Tests (PT) performed on safety-related systems and equipment during the initial test program.

Table 14.2-4 provides an index of the Acceptance Tests (AT) performed on nonsafety-related systems and equipment during the initial test program.

Table 14.2-5 provides an index of the Startup Tests (ST) performed during the initial startup program.

TABLE 14.2-1 INITIAL TEST PROGRAM RESPONSIBILITY/AUTHORITY MATRIX

	Preoperational Test Program		Initial Startup Program		
Activity	Individual System Tests	Integrated Systems Tests	Core Load	Criticality & Physics Tests	Power Escalation Tests
Test Program Management	STD	STD	STD	STD	STD
Test Procedure Preparation	STD	STD	STD or SS	STD or SS	STD or SS
Test Procedure Approval	JTG	JTG	STD, SORC	STD, SORC	STD, SORC
Test Coordination & Direction	STD	STD	STD or SS	STD or SS	STD or SS
Systems & Equipment Operations	STD or SS	SS	SS	SS	SS
Systems & Equipment Maintenance	STD or SS	SS	SS	SS	SS
Test Completion Approval	JTG	JTG	STD, SORC	STD, SORC	STD, SORC
Technical Support	NSD, NSS, AE, TG	NSD, NSS, AE, TG	NSD, NSS, AE	NSD, NSS, AE	NSD, NSS, AE, TG

Definitions

Test Program Management

"Test Program Management" defines the organization responsible for coordinating and sequencing of the initial test program activities.

Test Procedure Preparation

"Test Procedure Preparation" defines the organization responsible for preparation of the test procedure initial draft, coordination of the procedure review, and resolution of comments.

Test Procedure Approval

"Test Procedure Approval" defines the organization that will review and approve test procedures prior to their performance.

Test Coordination and Direction

"Test Coordination and Direction" defines the organization that will coordinate the activities prior to, during and after each test. A test director will insure that the test is properly conducted and all relevant data is properly recorded. Upon completion of the test, the data will be analyzed for completion review and approval.

Systems and Equipment Operations

"Systems and Equipment Operations" defines the organization responsible for the operation of the plant equipment during each phase of the test program.

System and Equipment Maintenance

"System and Equipment Maintenance" defines the organization responsible for the maintenance of plant equipment during each phase of the test program.

Test Completion Approval

"Test Completion Approval" defines the organizations that will review and approve the results of completed test procedures.

Definitions

Technical Support

"Technical Support" defines the offsite organizations that will be used to provide technical input for the initial test program, as required.

Legend:

STD	Startup Test Department - New Hampshire Yankee
JTG	Joint Test Group - JTG shall review, the Startup Manager shall approve.
NSS	Nuclear Steam Supply Vendor - Westinghouse Electric Corporation
AE	Architect-Engineer and Construction Manager - United Engineers & Constructors
SS	Station Staff - New Hampshire Yankee
NSD	Nuclear Services Division Yankee Atomic Electric Company
TG	Turbine Generator Vendor General Electric Company
SORC	Station Operations Review Committee - SORC shall review, the Station Manager shall approve.

TABLE 14.2-2 MAJOR ACTIVITIES REQUIRING STATION OPERATING REVIEW COMMITTEE APPROVAL PRIOR TO START

Fuel Loading

Initial Criticality

30% Power Ascension

50% Power Ascension

75% Power Ascension

100% Power Ascension

TABLE 14.2-3 PREOPERATIONAL TEST ABSTRACTS

Preoperational Test Abstracts

1. REACTOR COOLANT PUMPS

Objective

To verify proper operation of the reactor coolant pumps and to establish baseline data for pump operations.

Plant Conditions Prerequisites

Prior to and during hot functional testing.

Test Method

Instructions will be given specifying the required operations for the initial run of the reactor coolant pumps. Interlocks and controls will be tested. Pump operating data will be recorded. Additional operating data will be obtained during hot functional testing.

Acceptance Criteria

Reactor coolant pump controls and interlocks operate in accordance with the requirements of Updated FSAR Subsection 5.4.1, and baseline pump and motor data is collected.

Preoperational Test Abstracts

2. PRESSURIZER RELIEF TANK

Objective

To verify that the pressurizer relief tank provides adequate control of the discharges from the pressurizer relief valves.

Plant Conditions/Prerequisites

Prior to and during hot functional testing.

Test Method

The operation of the pressurizer relief tank will be demonstrated by performing operability checks of the tank and associated instrumentation and auxiliary equipment. During hot functional tests, the ability of the system to receive and cooldown a discharge from the power-operated relief valves will be verified.

Acceptance Criteria

The pressurizer relief tank operates in accordance with the requirements of Updated FSAR Subsections 5.4.11.2 and 5.4.11.4.

Preoperational Test Abstracts

3. REACTOR COOLANT AND ASSOCIATED SYSTEMS PIPING VIBRATION TEST

Objective

To demonstrate that vibration levels in selected ASME Code Class 1, 2 and 3 systems, Seismic Category I systems, and other high energy piping systems located in Seismic Category I structures are acceptable.

Plant Conditions/Prerequisites

Prior to initial core loading. The specific conditions required for each system will be specified by the test procedure.

Test Method

Selected lines will be instrumented and the amplitude of the vibrations measured for various operational modes. Noninstrumented piping will be inspected during system operation to ensure vibration levels are within acceptable limits.

Acceptance Criteria

The reactor coolant and associated system piping vibration does not exceed the requirements of Section III of the ASME Code, paragraph NB-3622.3.

Preoperational Test Abstracts

4. REACTOR COOLANT AND ASSOCIATED SYSTEMS THERMAL EXPANSION AND RESTRAINT TEST

Objective

To verify that the reactor coolant and other selected plant systems are free to expand during plant heatup and contract during plant cooldown.

Plant Conditions/Prerequisites

During heatup and cooldown for hot functional testing. Preservice inspection has been completed for hydraulic snubbers.

Test Method

Baseline position data will be taken at selected points on components and piping at cold plant conditions. During heatup to normal operating temperatures, expansion data will be taken at specified temperature plateaus at these selected points. An inspection will be performed to detect any points of interference which will be corrected prior to continuing the heatup. Hydraulic snubbers will also be visually inspected during heatup and cooldown to demonstrate operability. Following the cooldown, a final check of piping and component baseline positions will be obtained.

Acceptance Criteria

During heatup and cooldown of the Reactor Coolant System, the system piping is free to expand and contract in accordance with Updated FSAR Subsection 3.9(B).2.1b.

Preoperational Test Abstracts

5. CVCS - CHARGING AND LETDOWN

Objective

To demonstrate the charging, letdown, boric acid transfer, and associated purification functions of the Chemical and Volume Control System (CVCS).

Plant Conditions/Prerequisites

Prior to and during hot functional testing.

Test Method

Prior to hot functional tests, system components and their associated control systems will be operationally checked to the extent practical. During hot functional tests, the ability of the positive displacement pump and the centrifugal charging pumps to deliver water into the RCS will be demonstrated. The letdown capabilities will also be demonstrated and the various control systems will be operationally checked. Operations will be conducted to demonstrate the various modes of boration and dilution. The Boric Acid Transfer System will be functionally tested in its various operational modes. Operation of the purification system will be demonstrated by verification of flow and pressure drops across the demineralizers. Relevant system pressure, flow, and temperature data will be recorded.

Acceptance Criteria

The Chemical and Volume Control System charging and letdown operates in accordance with Updated FSAR Subsection 9.3.4.

Preoperational Test Abstracts

6. CVCS - BORON THERMAL REGENERATION SYSTEM

Objective

To demonstrate the operational capability of the Boron Thermal Regeneration System (BTRS).

Plant Conditions/Prerequisites

Prior to and during hot functional testing.

Test Method

Prior to hot functional testing, system components will be operationally checked to the extent practical. During hot functional testing, the system will be operated in its various operational modes and relevant pressure, temperature and flow data recorded.

Acceptance Criteria

The Boron Thermal Regeneration System operates in accordance with the requirements of Updated FSAR Subsection 9.3.4.2d.

Preoperational Test Abstracts

7. RESIDUAL HEAT REMOVAL SYSTEM

Objective

To demonstrate the operational capability of the Residual Heat Removal System, and to establish baseline data for pump operation.

Plant Conditions/Prerequisites

Prior to hot functional testing.

Test Methods

The Residual Heat Removal (RHR) System will be tested to verify controls and interlocks and to determine system operating characteristics. Operability of relief valves will be verified. Additional testing will be performed during the integrated plant cooldown from hot functional tests.

Acceptance Criteria

The Residual Heat Removal System operates in accordance with the requirements of Updated FSAR Subsection 5.4.7.

Preoperational Test Abstracts

8. ECCS PERFORMANCE TEST

Objective

To demonstrate the capability of the Emergency Core Cooling Systems to pump water from the refueling water storage tank into the reactor vessel through various combinations of pumps and injection lines.

Plant Conditions/Prerequisites

Prior to initial core loading with the reactor vessel open.

Test Method

A series of flow tests will be run using the centrifugal charging pumps, safety injection pumps, and RHR pumps to verify proper flow rates and to perform any required flow balancing during pumping from the RWST to the reactor vessel. The draw-down characteristics of the RWST and SAT will be demonstrated during these operations. Appropriate data will be obtained to determine pump headflow characteristics. The ability of the RHR pumps to supply water to the SI and the centrifugal charging pumps will be demonstrated. The operability of the RWST and the SAT (ECCS water sources) will be demonstrated.

Margin between pump motor current trip points and current values at full design flow conditions will be demonstrated.

Acceptance Criteria

The Emergency Core Cooling System operates in accordance with the requirements of Updated FSAR Subsection 6.3.2.1.

Preoperational Test Abstracts

9. ECCS HOT FUNCTIONAL TEST

Objective

To demonstrate the capability of the Emergency Core Cooling Systems to pump into the Reactor Coolant System at operating conditions, and to verify that the accumulator check valves operate properly at higher temperatures.

Plant Conditions/Prerequisites

During hot functional testing and during cooldown from hot functional testing.

Test Method

Water from the RWST will be injected into the Reactor Coolant System utilizing the centrifugal CVCS pumps to the extent necessary to verify check valve operation and to obtain rated pump flow. The duration of injection will be limited to minimize thermal shock effects. During the cooldown from hot functional tests, this test will be performed using the safety injection pumps and the accumulators. Following each injection, the ability of the check valves to reseat will be verified.

Acceptance Criteria

Emergency core cooling water is injected into the - primary system by each subsystem at its design operating limit in accordance with the Updated FSAR Subsection 6.3.2.1.

The associated system check valves are tested for leakage in accordance with Technical Specification 4.0.5.

Preoperational Test Abstracts

10. SAFETY INJECTION ACCUMULATOR BLOWDOWN TEST

Objective

To demonstrate proper system actuation and flow rate for the test conditions, and to demonstrate isolation valve operability.

Plant Conditions/Prerequisites

Prior to initial core loading with the reactor vessel open.

Test Method

The accumulators will be filled to their normal operating level and pressurized to a specified pressure. The accumulators will be discharged one at a time into the vessel and data will be collected to determine the rate of discharge. The accumulators will again be filled and pressurized to the maximum expected accumulator precharge pressure. The accumulator isolation valves will be opened under the maximum differential pressure condition.

Acceptance Criteria

Safety injection accumulator response is in accordance with the Updated FSAR Section 6.3 design requirements, and the isolation valves are capable of opening with maximum cover pressure specified in Technical Specification 3.5.1.

Preoperational Test Abstracts

11. CONTAINMENT RECIRCULATION SUMP OPERABILITY DEMONSTRATION

Objective

To verify the operability of the ECCS sump.

Plant Conditions/Prerequisites

Prior to initial core loading.

Test Method

The ECCS sump will be filled. An RHR and a containment spray pump will be operated at post-LOCA recirculation flow rates and recirculated back to the sump. Appropriate pressure and flow data will be recorded to verify net positive suction head characteristics.

Acceptance Criteria

The containment building sump provides fluid suction pressure greater than the net positive suction head required (as specified in the certified pump curves) to the RHR and CBS pumps at post-LOCA conditions.

Preoperational Test Abstracts

12. CONTAINMENT SPRAY SYSTEM

Objective

To verify the proper operation of the Containment Spray System.

Plant Conditions/Prerequisites

Prior to initial core loading.

Test Method

Tests will be performed to verify proper operation of all containment spray system components and to determine pump head-flow characteristics. Air flow tests of the containment spray nozzles will verify that the nozzles are not plugged.

Tests will be performed to verify proper operation of the Refueling Water Storage Tank (RWST) Heating System.

Acceptance Criteria

The Containment Spray System operates in accordance with safety analysis requirements of Updated FSAR Subsection 6.2.2.2.

Preoperational Test Abstracts

13. MAIN STEAM LINE ISOLATION VALVES

Objective

To verify proper operation of the main steam line isolation valves.

Plant Conditions/Prerequisites

Prior to and during hot functional testing.

Test Method

Operation of the main steam line isolation and bypass valves will be demonstrated at cold plant conditions including the response to a main steam line isolation signal. During hot functional tests, valve operation will be demonstrated and the closure time measured.

Acceptance Criteria

The Main Steam Line Isolation System operates in accordance with the requirements of the Updated FSAR Subsection 5.4.5 and valve closure times meet Updated FSAR Table 16.3-4 requirements for main steam isolation valves and main steam isolation valve bypass valves.

Preoperational Test Abstracts

14. EMERGENCY FEEDWATER SYSTEM

Objective

To demonstrate proper operation of the Emergency Feedwater System.

Plant Conditions/Prerequisites

Prior to and during hot functional testing.

Test Method

Prior to hot functional tests, emergency feedwater pump and feedwater isolation valve control and operability checks will be performed to the extent practical.

During hot functional tests, each emergency feedwater pump will be operated to verify pump bead-flow characteristics and to demonstrate the capability to feed the steam generators while at pressure. System response to ESF actuation signals, including the operation of the feedwater isolation valves, will be demonstrated.

A 48-hour endurance run and subsequent restart will be performed on each emergency feedwater pump to demonstrate long-term reliability.

At least five consecutive, successful, cold, pump starts for each emergency feedwater pump will be demonstrated.

Steam generator feeding capabilities using the Emergency Feedwater System under a simulated loss of offsite and onsite AC power condition will be demonstrated.

A flow instability test will be performed to demonstrate a "water hammer" will not occur in system components, piping, or inside the steam generators during normal system startup and operation.

The operation of the associated ventilation system will be verified during this test.

The operation of the condensate storage system will be demonstrated.

Preoperational Test Abstracts

14. EMERGENCY FEEDWATER SYSTEM (Continued)

Acceptance Criteria

The Emergency Feedwater System operates in accordance with the requirements of Updated FSAR Section 6.8.

The emergency feedwater pump can operate for the 48-hour endurance run with a subsequent restart without exceeding the operational limitations listed in the plant operating procedures.

Feedwater isolation valve closure times meet the requirements noted in the Technical Requirements Manual.

Steam generator feeding capabilities under a simulated loss of offsite and onsite AC power condition is demonstrated.

Preoperational Test Abstracts

15. SERVICE WATER SYSTEM

Objective

To demonstrate the proper operation of the Service Water System.

Plant Conditions/Prerequisites

During hot functional testing, and prior to initial core load.

Test Method

During hot functional testing, the ability to maintain required component temperatures will be verified. Cooling tower performance will be verified by a combination of air flow tests and capability analysis. Prior to initial core load, control system functional tests will be performed. Pump and overall system performance data will be obtained using both the ocean and the cooling tower as the source of cooling water. The operation of the associated ventilation systems for the Service Water Pumphouse and Cooling Tower will be demonstrated.

The overflow is not tested because of the potential to undermine paving and wash out earthen fill. System design calculations were bench-marked against actual test data so that the calculated flow out the overflow is valid.

Acceptance Criteria

The Service Water System operates in accordance with the requirements of the Updated FSAR Subsections 9.2.1 and 9.2.5.

Each system flow train supplies cooling water to both safety and nonsafety-related loops in the normal plant configuration and to safety-related loops in the accident configuration utilizing either the ocean or cooling tower.

The cooling tower performance test results demonstrate the dissipation of the heat loads specified in Table 9.2-13 of the Updated FSAR.

Cooling tower makeup water equipment meets Updated FSAR Subsection 9.2.5.3c criteria.

Preoperational Test Abstracts

16. • PRIMARY COMPONENT COOLING WATER SYSTEM

Objective

To demonstrate the proper operation of the PCCW System.

Plant Conditions/Prerequisites

Prior to and during hot functional testing.

Test Method

Prior to hot functional testing, system component operability checks and control system functional tests will be performed. During hot functional tests, data will be taken to verify that adequate cooling is being provided to PCCW components.

Acceptance Criteria

The Primary Component Cooling Water System operates in accordance with the requirements of the Updated FSAR Subsection 9.2.2.

The system supplies cooling water to both the safety and nonsafety-related loops in the normal plant configuration and to safety-related loops in the accident configuration.

Preoperational Test Abstracts

17. SPENT FUEL POOL COOLING SYSTEM

Objective

To demonstrate the proper operation of the Spent Fuel Pool Cooling System.

Plant Conditions/Prerequisites

Prior to initial core loading.

Test Method

Spent fuel pool cooling and cleanup system equipment operability checks, flow verification tests and control system functional tests will be performed in the various system operational modes to demonstrate proper system performance.

Antisyphon devices, high radiation alarms, and low water level alarms will be demonstrated.

Leak tests of sectionalizing devices will be demonstrated operable.

Acceptance Criteria

The Spent Fuel Pool Cooling System operates in accordance with the requirements of Updated FSAR Subsection 9.1.3.

Normal and alternate system design flow paths are demonstrated.

Preoperational Test Abstracts

18. EXCORE NUCLEAR INSTRUMENTATION

Objective

To verify the calibration of the excore nuclear instrumentation.

Plant Conditions/Prerequisites

Prior to initial core loading.

Test Method

The excore nuclear instrument channels will be calibrated and functionally checked to verify alarm and trip setpoints and the operation of auxiliary equipment. The response of the source range detectors to a neutron source will be verified.

Acceptance Criteria

The reactor trip setpoints and interlocks generated by the Nuclear Instrumentation System have been verified at the values specified in the Technical Specifications Table 2.2-1.

The control and indication functions of the Nuclear Instrumentation System have been demonstrated to be in accordance with the Updated FSAR Section 7.2. The overall time-response of the nuclear instrumentation channels has been demonstrated at the values specified in Updated FSAR Table 16.3-1.

Preoperational Test Abstracts

19. REACTOR PROTECTION SYSTEM AND ENGINEERED SAFETY FEATURES

Objective

To verify proper operation and response time of the Reactor Protection System and the engineered safety features (ESF) actuation logic.

Plant Conditions/Prerequisites

Prior to initial core loading.

Test Method

The operation of the Reactor Protection System will be verified for all conditions of logic using outputs or simulated outputs from each of the RPS sensors through to tripping of the reactor trip breakers. Individual protection channels will be tested to check design redundancy and to demonstrate safe failure on loss of power.

The Reactor Trip System response time shall be the time interval from when the monitored parameter exceeds its trip setpoint at the channel sensor until loss of stationary gripper coil voltage.

The operation of the ESF logic will be verified for all modes of operation using outputs or simulated outputs from each of the sensors through to the output of the slave relays. Individual ESF channels will be tested to verify design redundancy. The response time of required ESF signals will be determined from the sensor output to equipment actuation.

Acceptance Criteria

The Reactor Protection System has been verified to operate in accordance with the design requirements dictated by Updated FSAR Section 7.2.

The reactor protection time response has been verified to meet the requirements specified in Updated FSAR Table 16.3-1. All reactor protection system trip setpoints and interlocks have been demonstrated at the values specified in Technical Specifications Table 2.2-1.

The engineered safety features have been demonstrated to operate in accordance with the design requirements of Updated FSAR Section 7.3. The engineered safety features time response has been verified to meet the requirements specified by Updated FSAR Table 16.3-2.

Preoperational Test Abstracts

21. PRIMARY CONTAINMENT ISOLATION VALVES

Objective

To verify that the primary containment isolation valves respond properly to their respective isolation signals.

Plant Conditions/Prerequisites

Prior to primary containment leak rate test.

Test Method

Primary containment isolation valve controls and interlocks will be tested to verify the following:

a. That on a Phase A containment isolation test signal, each Phase A isolation valve actuates to its isolation position.

b. That on a Phase B containment isolation test signal, each Phase B isolation valve actuates to its isolation position.

c. That on a containment purge and exhaust isolation test signal, each purge and exhaust valve actuates to its isolation position.

d. That on a feedwater isolation test signal, each feedwater isolation valve actuates to its isolation position.

e. That the isolation time of each automatic containment isolation valve is within its limit.

Acceptance Criteria

The primary containment isolation valves operate in accordance with safety analysis requirements for "Containment Isolation System," Updated FSAR Subsection 6.2.4.

Preoperational Test Abstracts

22. CONTAINMENT ENCLOSURE VENTILATION SYSTEMS

Objective

To demonstrate proper operation of the Containment Enclosure Ventilation Systems.

Plant Conditions/Prerequisites

Prior to and during hot functional testing.

Test Method

Containment enclosure ventilation operability tests, and air flow verification checks will be performed prior to and during hot functional testing. Data will be recorded, during hot functional testing, to verify that area temperatures are satisfactory.

Acceptance Criteria

Satisfactory demonstration of equipment to maintain area temperatures per design, as stated in Updated FSAR Subsection 9.4.6.

Preoperational Test Abstracts

23. CONTAINMENT ENCLOSURE EXHAUST SYSTEM

Objective

To demonstrate proper operation of the Containment Enclosure Exhaust System.

Plant Conditions/Prerequisites

Prior to initial core loading.

Test Method

Containment enclosure exhaust equipment operability checks and control system functional tests will be performed to verify their ability to:

a. Provide isolation from Primary Auxiliary Building "normal exhaust system," switching to emergency exhaust air cleaning mode following a LOCA.

b. Maintain negative pressure within containment enclosure areas with respect to the outside by the Containment Enclosure Exhaust Filter System.

Containment enclosure filters will be installed.

In-place filter verification testing will be conducted to satisfy Regulatory Guide 1.52. This testing will be performed in Item 44.

Acceptance Criteria

Containment enclosure exhaust system operational performance tests satisfy design criteria, as described in Updated FSAR Subsections 6.5.1 and 9.4.6.

Preoperational Test Abstracts

24. CONTAINMENT ENCLOSURE LEAK RATE TEST

Objective

To demonstrate the containment enclosure leakage is less than design.

Plant Conditions/Prerequisites

Prior to initial core load.

Test Method

The containment enclosure area will be leak tested by demonstrating equipment ability to provide negative pressure.

Both containment enclosure exhaust filter subsystems will be tested to verify that independently they can establish and maintain the containment enclosure area at, or greater than, 0.25 inches water negative pressure within the required time following a containment isolation signal.

Acceptance Criteria

Containment enclosure area leakage meets design and safety analysis requirements as set forth in Updated FSAR Subsection 6.5.1.

Preoperational Test Abstracts

25. CONTAINMENT COMBUSTIBLE GAS CONTROL SYSTEM

Objective

To demonstrate the proper operation of the Containment Combustible Gas Control System.

Plant Conditions/Prerequisites

Prior to initial core loading.

Test Method

Containment combustible gas control system operability checks, flow verification tests, and control system functional tests will be performed to demonstrate proper system performance.

Acceptance Criteria

The Containment Combustible Gas Control System operates in accordance with the requirements of Updated FSAR Subsection 6.2.5.

Preoperational Test Abstracts

26. CONTAINMENT AIR RECIRCULATION SYSTEM

Objective

To demonstrate proper operation of the Primary Containment Air Recirculation System.

Plant Conditions/Prerequisites

Prior to initial core loading.

Test Method

Primary containment air recirculation system equipment operability checks and control system functional checks will be performed to verify:

a. Air flow for required containment air mixing.

b. Recirculation fans auto start on containment pressure high-high (P) signal.

c. Redundant systems will function properly in either "recirculation" or "filter" mode.

d. Filters will be installed and in-place filter verification tests will be conducted in accordance with Regulatory Guide 1.140.

Acceptance Criteria

Containment air recirculation systems demonstrated to operate satisfactorily in accordance with design requirements set forth in Updated FSAR Subsection 9.4.5.

Preoperational Test Abstracts

27. FUEL STORAGE BUILDING VENTILATION SYSTEM

Objective

To demonstrate proper operation of the Fuel Storage Building ventilation system.

Plant Conditions/Prerequisites

Prior to receipt of new fuel.

Test Method

Fuel storage building ventilation equipment operability check and control system functional test will be performed to verify:

a. Proper air flows in both normal and fuel handling modes.

b. Normal exhaust isolation and Fuel Storage Building established and maintained at 0.25 inches water negative pressure when in fuel handling mode.

c. Filters will be installed, and in-place filter verification testing conducted to satisfy Regulatory Guide 1.52.

Acceptance Criteria

Fuel storage building ventilation system tests satisfactorily demonstrate ability to meet, or exceed, design requirements per Updated FSAR Subsections 9.4.2 and 6.5.1.

Preoperational Test Abstracts

28. CONTROL ROOM HVAC

Objective

To demonstrate proper operation of the control room heating, ventilating, and air conditioning system.

Plant Conditions/Prerequisites

Prior to initial core loading.

Test Method

The control room complex, excluding computer room HVAC equipment, HVAC subsystems will be demonstrated by equipment operability checks and control(s) functional tests.

Air flow verification tests will be performed.

Makeup air filters will be installed, and in-place filter verification testing will be done to satisfy Regulatory Guide 1.140.

"Control room envelope" boundary seals will be verified by demonstrating the ability to maintain the "envelope" at the required positive pressures.

Emergency air cleanup subsystem automatic initiation on "S" safety injection signal, and makeup air isolation on high radiation signal will be verified.

Acceptance Criteria

The control room complex ventilation subsystems will be satisfactorily demonstrated to meet or exceed safety and design requirements, as stated in Updated FSAR Section 6.4 and Subsection 9.4.1.

Preoperational Test Abstracts

29. EMERGENCY SWITCHGEAR VENTILATION

Objective

To demonstrate the proper operation of the emergency switchgear and cable spreading area ventilation systems.

Plant Conditions/Prerequisites

Prior to initial core loading.

Test Method

The emergency switchgear and cable spreading areas ventilation systems shall have equipment operability checks and control system functional tests performed.

Tests will include battery room(s) and electrical tunnel(s) ventilation subsystems.

Air flow verification tests will be performed on all subsystems. In particular, equipment will be demonstrated capable of:

a. Maintaining battery rooms at slightly negative pressure to prevent hydrogen exfiltration.

b. 4-kV switchgear areas at a slightly positive pressure to prevent infiltration of dirt and dust.

Acceptance Criteria

Operation of emergency switchgear and cable spreading area ventilation, including all subsystems, shall be demonstrated satisfactory under normal and emergency conditions per design criteria in Updated FSAR Subsections 9.4.9 and 9.4.10.

Preoperational Test Abstracts

30. AC ELECTRICAL DISTRIBUTION

Objective

To demonstrate the capability of the offsite power system to serve as a source of power to the emergency buses.

Plant Conditions/Prerequisites

During hot functional testing and prior to initial core loading.

Test Method

During hot functional testing the analytical techniques and assumptions used in the voltage analyses will be verified by actual measurement. These tests will meet the requirements of NRC Branch Technical Position, PSB 1, Adequacy of Station Electrical Distribution System Voltage.

Subsequent to hot functional testing, tests will be performed to demonstrate the transfer capabilities between the UATs and the RATS. Undervoltage protection will also be demonstrated at this time.

Acceptance Criteria

The AC electrical distribution system operates in accordance with the requirements of Updated FSAR Section 8.3.

Preoperational Test Abstracts

31. 125V DC DISTRIBUTION SYSTEM

Objective

To demonstrate the proper operation of the 125V DC Distribution System.

Plant Conditions/Prerequisites

Prior to loss of offsite power tests.

Test Methods

Tests will be performed to demonstrate operation of instrumentation and alarms, and that actual total system amperage loads are in agreement with design loads. A discharge test of each battery bank will be conducted. System interlocks will be verified to demonstrate proper operation under accident conditions. The independence of redundant power supplies and load groups will be verified.

Tests will be performed on selected circuits to ensure proper operation of the safety related DC loads at the minimum battery terminal voltage.

Acceptance Criteria

The DC power system operates in accordance with the requirements of Updated FSAR Subsection 8.3.2.

Preoperational Test Abstracts

32. 120V AC VITAL INSTRUMENT POWER SUPPLY

Objective

To demonstrate the proper operation of the 120V AC vital instrumentation power supply.

Plant Conditions/Prerequisites

Prior to loss of offsite power tests.

Test Methods

Full-load tests for the uninterruptible power supplies to the vital buses will be conducted using normal and emergency sources of power supplies to the bus. System interlocks will be verified to demonstrate proper operation. The independence of the redundant power supplies and load groups will be verified.

Acceptance Criteria

The 120V AC Vital Instrument Power Supply System operates in accordance with Updated FSAR Subsection 8.3.1.1d.

Preoperational Test Abstracts

33. DIESEL GENERATORS

Objective

To demonstrate reliability and extended full-load carrying capability of the emergency diesel generator units.

Plant Conditions/Prerequisites

Prior to core loading. Where possible, diesel generator reliability testing will be completed prior to loss of offsite power test.

Test Method

System and component operability checks will be performed on the diesel engine support and ventilation systems.

Protective features and interlocks will be demonstrated operable.

Full load-carrying capability for an interval of not less than 24 hours will be demonstrated operable in accordance with Regulatory Guide 1.108 position 2.a(3).

Diesel generator load shedding will be demonstrated operable in accordance with Regulatory Guide 1.108 position 2.a(4).

Diesel generator functional capability at full-load temperature will be demonstrated operable in accordance with Regulatory Guide 1.108 position 2.a(5) with exception as described in Updated FSAR Section 1.8.

Diesel generator reliability will be demonstrated by performing 35 consecutive valid starts per diesel generator in accordance with Regulatory Guide 1.108 position 2.a(9).

Acceptance Criteria

The diesel generators operate in accordance with the requirements of Updated FSAR Subsection 8.3.1.1e and complete the testing specified above.

Preoperational Test Abstracts

34. FUEL HANDLING AND TRANSFER EQUIPMENT

Objective

To demonstrate the proper operation of fuel handling equipment.

Plant Conditions/Prerequisites

Prior to storage of new fuel and initial core loading, as applicable. Dynamic and static load testing of cranes, hoists, and associated lifting and rigging equipment, including the fuel cask handling has been completed. Static testing has been performed at 125% of rated load and full operational testing has been performed at 100% of rated load.

Test Method

Tests will be performed prior to core loading to demonstrate the functional operability, controls and protective interlocks of the fuel handling and transfer equipment used for handling spent fuel. Components required for new fuel storage will be checked prior to the receipt of new fuel.

Acceptance Criteria

The fuel handling and transfer equipment operate in accordance with the requirements of Updated FSAR Subsections 9.1.1 and 9.1.2.

Preoperational Test Abstracts

35. REACTOR COOLANT SYSTEM HYDROSTATIC TEST

Objective

To perform a cold hydrostatic test of the Reactor Coolant System.

Plant Conditions/Prerequisites

Prior to hot functional testing.

Test Method

Prior to pressurization, the Reactor Coolant System will be heated above the minimum temperature for pressurization. A hydrostatic test of the Reactor Coolant System and adjoining unisolable piping will be conducted to the requirements of ASME B&PV Code, Section III. The pressure will be increased in increments and at each increment inspections will be made for leakage. Leaky valve seats or mechanical joints will not be a basis for rejecting the test.

Acceptance Criteria

The hydrostatic test meets the requirements of ASME B&PV Section III.

Preoperational Test Abstracts

36. PRIMARY CONTAINMENT STRUCTURAL INTEGRITY TEST

Objective

To perform a structural integrity verification of the containment structure.

Plant Conditions/Prerequisites

Prior to initial core loading.

Test Method

A structural acceptance test will be performed in accordance with Regulatory Guide 1.18.

The Containment will be pressurized to 115% (60 psig) of the design pressure (52 psig), with inside and outside temperatures monitored and controlled.

Structural responses to test conditions will be measured and compared to predicted responses, to verify that structural behavior is as analytically anticipated.

Acceptance Criteria

The reactor containment structure meets structural integrity design requirements, as defined by regulatory guides and codes described in Updated FSAR Sections 3.1, 3.8 and 6.2.

Preoperational Test Abstracts

37. PRIMARY CONTAINMENT LEAK RATE TESTS

Objective

To perform the initial primary containment leak rate tests.

Plant Conditions/Prerequisites

Prior to initial core loading.

Test Method

Type A, B and C primary containment leak rate tests will be performed in accordance with the requirements of 10 CFR 50, Appendix J.

Prior to the Type A test, Type C (containment isolation valve leakage rate test) tests will be performed at a pressure not less than Pa (46.1 psig). Valve leakage rates will be recorded and verified within allowable design limits.

Prior to the Type A test and concurrent with Type C tests, Type B (containment penetration leakage rate test) tests will be performed on containment airlocks, hatches, electrical penetration and fuel transfer tube, at a pressure not less than Pa.

Type C and B leakage rate results will be totaled and verified within allowable design limits.

On completion of all prerequisite testing, the Containment will be pressurized to Pa; while pressure, temperature and dew point will be controlled, recorded and allowed to stabilize. Test conditions will be maintained for a minimum 24 hour period and, utilizing the perfect gas law, leakage rate in percent per day will be computed from the changes in containment air mass.

At the end of the prescribed test period, an instrument accuracy verification test will be performed on the Containment to verify test instrumentation and test results accuracy.

Acceptance Criteria

Type A, B and C leak rates and instrumentation accuracy is verified to be within allowable design limits as set forth in the Updated FSAR Section 6.2.

Preoperational Test Abstracts

38. ESF INTEGRATED ACTUATION TEST

Objective

To demonstrate proper plant system response to ESF actuation signals.

Plant Conditions/Prerequisites

Prior to initial core loading and subsequent to hot functional testing.

Test Method

Simulated ESF signals will be introduced and the integrated plant response will be monitored to verify proper pump, valve, and diesel generator actuation.

Acceptance Criteria

The ESF actuation signals operate in accordance with safety analysis requirements of the Updated FSAR Subsections 6.2.2 and 6.3.2.

Preoperational Test Abstracts

39. LOSS OF OFFSITE POWER TESTS

Objective

To demonstrate the proper response of plant systems to a loss of offsite power.

Plant Conditions/Prerequisites

Prior to initial core loading.

Test Method

Startup operation of the diesel generator units will be demonstrated by simulating loss of all AC voltage in accordance with Regulatory Guide 1.108 position 2.a(l).

Proper operation for design-accident-loading-sequence to design-load requirements will be demonstrated in accordance with Regulatory Guide 1.108 position 2.a(2). This testing will be conducted with one diesel generator unit at a time. The bus not being tested will be monitored to verify absence of voltage. Load testing of the batteries will also be demonstrated.

The ability to (a) synchronize with offsite power, (b) transfer load to offsite power, (c) isolate the diesel generator unit, and (d) restore it to standby status will be demonstrated in accordance with Regulatory Guide 1.108 position 2.a(6).

The capability of the diesel generator unit to supply emergency power within the required time is not impaired during periodic testing and will be demonstrated in accordance with Regulatory Guide 1.108 position 2.a(8).

Testing will be conducted in which both diesel generator units will be started simultaneously to help identify certain common failure modes undetected in single diesel generator unit tests in accordance with Regulatory Guide 1.108 position 2.b.

Acceptance Criteria

Diesel generator operation and circuit breaker sequencing are in accordance with the Updated FSAR design requirements of Section 8.3.

Preoperational Test Abstracts

40. INTEGRATED HOT FUNCTIONAL TESTS

Objective

To verify the proper operation of various primary and secondary instrumentation, controls and components at normal operating temperatures and pressures and to provide general guidance for the conduct of hot functional testing.

Plant Conditions/Prerequisites

Prior to initial criticality.

Test Method

General guidelines to conduct the hot functional test program will be provided. Following plant heatup, the reactor coolant temperature and pressure will be maintained at normal operating values. A series of tests, which are listed below, will be performed to verify system operation.

a. Demonstration of the Pressurizer Pressure Control System ability to maintain RCS pressure.

b. Demonstration of the Pressurizer Level Control System ability to maintain pressurizer level.

c. Demonstration of the RCS leak detection capability.

d. Verification of steam generator level instrumentation operability.

e. Verification of selected primary and secondary plant instrumentation operability.

f. Demonstration of the remote shutdown panel to maintain the plant in a hot shutdown condition.

g. Initial roll of the turbine generator with main steam, including verification of turbine stop, reheat and intercept valve operation.

h. Verification of pressurizer and main steam safety valve setpoints.

i. Demonstration of condenser steam dump valve operation.

j. Demonstration of containment penetration cooling capacity.

Preoperational Test Abstracts

40. INTEGRATED HOT FUNCTIONAL TESTS (Continued)

In addition to the above, other tests specified in Tables 14.2-3 and 14.2-4 as being performed during hot functional testing will be performed at this time. After the completion of at temperature testing, the plant will be cooled down.

Acceptance Criteria

The plant has been operated at hot condition in accordance with normal plant operating procedures and the following systems operate in accordance with the requirements of the Updated FSAR or Technical Specifications listed below:

System

Pressurizer pressure control	Updated FSAR Subsection 7.7.1
Pressurizer level control	Updated FSAR Subsection 7.7.1
RCS leak detection capability	Updated FSAR Subsection 5.2.5
Remote shutdown panel	Updated FSAR Subsection 7.4.1.3
Turbine generator	Updated FSAR Section 10.2
Pressurizer safety valve lift setting	TS 3.4.2.1
Steam line safety valve lift setting	TS Table 3.7-2

Preoperational Test Abstracts

41. INTEGRATED PLANT HEATUP FOR HOT FUNCTIONAL TESTS

Objective

To demonstrate the ability to bring the plant to normal operating temperature and pressure from a cold shutdown condition.

Plant Conditions/Prerequisites

The plant is at cold shutdown conditions following the performance of the primary hydrostatic test.

Test Method

The plant will be brought to normal operating pressure and temperature using reactor coolant pump heat. The test instructions will be based upon normal plant operating procedures to verify their methods. At specific points, the heatup will be terminated to allow the performance of specified hot functional tests.

Demonstration of the steam line atmospheric dump valves will be conducted during plant heat up.

Acceptance Criteria

The plant has been brought to normal operating temperature and pressure in accordance with normal plant operating procedures.

Preoperational Test Abstracts

42. INTEGRATED PLANT COOLDOWN FROM HOT FUNCTIONAL TESTS

Objective

To demonstrate the ability to bring the plant from normal operating temperature and pressure to cold shutdown conditions.

Plant Conditions/Prerequisites

The plant is at normal temperature and pressure following the completion of hot functional testing.

Test Method

The plant will be brought to hot shutdown conditions using steam dumps and the Residual Heat Removal System, from outside the control room. After initiation of residual heat removal system cooling, the plant will be further cooled an additional 50°F. After the additional 50°F cooldown, control will be transferred back to the control room. During operation of the Residual Heat Removal System, cooldown rates will be monitored and controlled, and data will be collected to verify its heat removal capability. The cooldown limitations of Technical Specification 3.4.9.1 will not be exceeded. At specific points, the cooldown will be terminated to allow the performance of specified hot functional tests.

Acceptance Criteria

The plant has been brought to cold shutdown conditions in accordance with normal plant operating procedures.

Preoperational Test Abstracts

43. REACTOR POST-HOT FUNCTIONAL INSPECTION

Objective

To provide a sequence of operations to be followed after hot functional tests to disassemble, clean, and inspect the reactor vessel and internals.

Plant Conditions/Prerequisites

After completion of hot functional testing and prior to initial core loading.

Test Method

Instructions will be given describing the required steps to disassemble, inspect, and clean the reactor vessel and its internals.

Acceptance Criteria

The reactor vessel is cleaned to the requirements of plant procedures and the internals are inspected in accordance with Updated FSAR Subsection 3.9(N).2.4.

Preoperational Test Abstracts

44. PLANT VENTILATION SYSTEMS FILTER TESTING

Objective

To demonstrate the proper operation of in-place plant filters.

Plant Conditions/Prerequisites

Prior to initial core loading.

Test Method

In-place HEPA filters will be visually inspected, and airflow capacity measured. A leak test will be conducted and the associated fan vibration measured.

Acceptance Criteria

In-place filters have been installed properly and have not been damaged. No circuitous flow paths which would compromise the filters/absorbers exist.

Airflow capacity meets applicable fan design requirements and airflow distribution meets ANSI N510-1980.

Duct heaters perform as required by ANSI N510-1980.

Fan vibration meets the criteria of ANSI N509-1980.

Preoperational Test Abstracts

45. PRIMARY AUXILIARY BUILDING VENTILATION SYSTEM

Objective

To demonstrate proper operation of the Primary Auxiliary Building Ventilation System.

Plant Conditions/Prerequisites

Prior to initial core loading.

Test Method

Primary auxiliary building ventilation equipment and controls will be tested to demonstrate required functional operability.

Air flows will be verified on all subsystems (normal and filtered clean-up systems).

Filters will be installed and in-place filter verification tests conducted to satisfy Regulatory Guide 1.140. This testing will be performed in the Updated FSAR , Item 44.

Acceptance Criteria

Primary auxiliary building ventilation subsystems have been demonstrated to function per design for normal and emergency operational modes as described in Updated FSAR Section 6.5 and Subsection 9.4.3.

TABLE 14.2-4 ACCEPTANCE TEST ABSTRACTS

Acceptance Test Abstracts

1. FEEDWATER SYSTEM

Objective

To demonstrate the proper operation of the Feedwater System.

Plant Conditions/Prerequisites

Prior to and during hot functional testing.

Test Method

Prior to hot functional testing, the operability of the startup feedwater pump will be demonstrated. The control and performance characteristics of the turbine-driven feedwater pumps will be operationally checked to the extent practical using the auxiliary boiler as a source of steam. During hot functional tests, the operation of the turbine-driven feedwater pumps will be demonstrated using main steam.

Acceptance Criteria

The Feedwater System operates in accordance with the requirements of Updated FSAR Subsections 10.4.7 and 10.4.12 for the conditions prevalent during hot functional testing.

Acceptance Test Abstracts

2. EXTRACTION STEAM AND HEATER DRAIN SYSTEMS

Objective

To demonstrate the proper operation of the extraction steam and heater drain system equipment.

Plant Conditions/Prerequisites

Prior to initial core loading.

Test Method

Functional tests will be performed to verify, to the extent practical, the proper operation of equipment associated with the extraction steam and heater drain systems.

Acceptance Criteria

The extraction steam and heater drain systems performance, to the extent practical, is in accordance with Updated FSAR Subsections 10.2.2.3 and 10.4.7 in response to simulated input signals.

Acceptance Test Abstracts

3. CONDENSATE SYSTEM

Objective

To demonstrate the proper operation of the condensate system.

Plant Conditions/Prerequisites

Prior to and during hot functional testing.

Test Method

Tests will be performed to demonstrate the operational performance characteristics of the condensate pumps, the hotwell level control system, and the condensate makeup water system.

Acceptance Criteria

The condensate system operates in accordance with the requirements of Updated FSAR Subsection 10.4.7 at conditions prevalent up to and including hot functional testing.

Acceptance Test Abstracts

4. CONDENSER AIR REMOVAL SYSTEMS

Objective

To demonstrate the proper operation of the condenser air removal equipment.

Plant Conditions/Prerequisites

Prior to hot functional testing.

Test Method

Tests will be performed to verify the operation and operational characteristics of the condenser vacuum pumps and the water box priming pumps.

Acceptance Criteria

The Condenser Air Removal System operates in accordance with the requirements of the Updated FSAR Subsection 10.4.2.

Acceptance Test Abstracts

5. CHEMICAL ADDITION SYSTEM

Objective

To demonstrate the proper operation of the Secondary Plant Chemical Addition System.

Plant Conditions/Prerequisites

Prior to hot functional testing.

Test Method

Functional tests will be performed on the chemical addition pumps and associated instrumentation and controls to demonstrate proper operation.

Acceptance Criteria

The chemical addition system operates in accordance with the requirements of Updated FSAR Subsection 10.3.5.

Acceptance Test Abstracts

6. CIRCULATING WATER SYSTEM

Objective

To demonstrate the proper operation of the Circulating Water System.

Plant Conditions/Prerequisites

Prior to initial core loading.

Test Method

Functional testing of system components, instrumentation and controls will be performed to demonstrate operability. An overall system dynamic performance test will be run to determine specific system operating parameters and response during steady-state, transient, and backflush operational modes. Proper operation of the pumphouse ventilation system will be verified.

Acceptance Criteria

The Circulating Water System operates in accordance with the requirements of Updated FSAR Subsection 10.4.5.

Acceptance Test Abstracts

7. SECONDARY COMPONENT COOLING WATER SYSTEM

Objective

To demonstrate the proper operation of the Secondary Component Cooling Water (SCCW) System.

Plant Conditions/Prerequisites

Prior to and during hot functional testing.

Test Method

Prior to hot functional testing, system component operability checks and control system functional tests will be performed. During hot functional testing, data will be taken to the extent practical to verify that adequate cooling is being provided to SCCW components.

Acceptance Criteria

Secondary Component Cooling Water System operates in accordance with Updated FSAR Subsection 10.4.10.

Acceptance Test Abstracts

8. CVCS - LETDOWN DEGASIFIER

Objective

To demonstrate the proper operation of the letdown degasifier portion of the CVCS.

Plant Conditions/Prerequisites

Prior to initial core loading.

Test Method

Functional tests will be performed to demonstrate the operational characteristics and performance of the CVCS letdown degasifier.

Acceptance Criteria

The letdown degasifier operates in accordance with Updated FSAR Subsection 9.3.4.2e.11.

Acceptance Test Abstracts

9. REACTOR MAKEUP WATER SYSTEM

Objective

To demonstrate the proper operation of the Reactor Makeup Water System.

Plant Conditions/Prerequisites

Prior to initial core loading.

Test Method

Functional testing of system components, instrumentation, and controls will be performed to demonstrate the ability of this system to transfer water to other plant systems.

Acceptance Criteria

The Reactor Makeup Water System operates in accordance with Updated FSAR Subsection 9.2.7.

Acceptance Test Abstracts

10. SAMPLING SYSTEM

Objective

To verify the proper operation of the plant sampling systems and installed analysis equipment.

Plant Conditions/Prerequisites

This test will be performed during operation of the systems which are served by the sample system. The major portion of this test will be performed during hot functional testing.

Test Method

Samples will be drawn from each of the primary and secondary sample points to verify proper piping arrangement and function. Holdup times for the sample lines from reactor coolant loops 1 and 3 will also be verified. The installed chemical analysis equipment will be operationally checked to the extent practical.

Acceptance Criteria

Sample point identification and piping arrangement has been verified.

The sampling system operates in accordance with the design requirements of Updated FSAR Subsection 9.3.2.1.

Acceptance Test Abstracts

11. REACTOR COOLANT DRAIN SYSTEM

Objective

To demonstrate the proper operation of the Reactor Coolant Drain System.

Plant Conditions/Prerequisites

Prior to hot functional testing.

Test Method

The reactor coolant drain tank, pumps and associated components will be functionally tested to verify proper performance in the various system operating modes.

Acceptance Criteria

The Reactor Coolant Drain System operates in accordance with the design requirements of Updated FSAR Subsection 9.3.5.2.

Acceptance Test Abstracts

12. INSTRUMENT AND SERVICE AIR SYSTEMS

Objective

To demonstrate the proper operation of the Instrument and Service Air Systems.

Plant Conditions/Prerequisites

Prior to initial core loading.

Test Methods

Functional testing of system components, instruments, and controls for the plant and containment building air system will be performed to demonstrate operability. Testing of components which perform a safety-related function will be performed during the respective system preoperational test.

Acceptance Criteria

Instrumentation, controls, alarms and interlocks operate as required in accordance with Updated FSAR Subsection 9.3.1 in response to normal or simulated input signals.

Acceptance Test Abstracts

13. FIRE PROTECTION SYSTEM

Objective

To demonstrate the proper operation of the fire protection system.

Plant Conditions/Prerequisites

Prior to initial core loading.

Test Methods

Tests will be performed to demonstrate proper operation of fire protection subsystem equipment and controls as follows:

a. Capacity tests of the fire pumps

b. Actuation tests of the water and halon systems

c. Proper operation of the smoke and fire detection systems

d. Fire pumphouse heating and ventilation

e. Fire protection interlocks with other (i.e., HVAC) plant systems

Acceptance Criteria

Demonstration of fire protection system performance satisfies codes and regulations per design, as stated in Updated FSAR Subsection 9.5.1.

Acceptance Test Abstracts

14. RADIOACTIVE GASEOUS WASTE SYSTEM

Objective

To demonstrate the proper operation of radioactive gaseous waste system components.

Plant Conditions/Prerequisites

Prior to initial core loading.

Test Method

Operability tests will be performed on system components, instrumentation and controls to the extent practical to verify proper operation.

Acceptance Criteria

The radioactive gaseous waste system operates in accordance with Updated FSAR Section 11.3.

Flow paths to all system components have been demonstrated.

Acceptance Test Abstracts

15. LIQUID WASTE SYSTEM

Objective

To demonstrate the proper operation of liquid waste system components.

Plant Conditions/Prerequisites

Prior to initial core loading.

Test Method

Tests will be performed to the extent practical to verify the proper operation of the liquid waste system components, instrumentation, and controls. Isolation of liquid waste will be demonstrated.

Tests will be performed to verify the proper operation of the equipment and floor drainage system sump/tank high level alarms.

Acceptance Criteria

Flow paths to the liquid waste system components have been demonstrated.

The liquid waste system operates in accordance with Updated FSAR Section 11.2. The equipment and floor drainage system operates in accordance with the Updated FSAR Subsection 9.3.3.

Acceptance Test Abstracts

16. SPENT RESIN SLUICE SYSTEM

Objective

To demonstrate the proper operation of the resin sluice system equipment.

Plant Conditions/Prerequisites

Prior to initial core loading.

Test Method

Functional tests of system components, instrumentation, and controls will be performed to verify, to the extent practical, the proper operation of resin sluice system equipment.

Acceptance Criteria

The Spent Resin Sluice System operates in accordance with Updated FSAR Subsection 11.4.2.3a.

Flow paths to system components have been demonstrated.

Acceptance Test Abstracts

17. WASTE SOLIDIFICATION SYSTEM

Objective

To demonstrate the proper operation of waste solidification system equipment.

Plant Conditions/Prerequisites

Prior to initial core loading.

Test Method

Tests will be performed to the extent practical to verify proper operation of system components, instrumentation and controls. Solidification of test samples, representative of the expected wastes, will be performed to verify proper operation of the Waste Solidification System.

Acceptance Criteria

The Waste Solidification System operates in accordance with Updated FSAR Subsection 11.4.2.4.

Acceptance Test Abstracts

18. BORON RECOVERY SYSTEM

Objective

To demonstrate the proper operation of boron recovery system equipment.

Plant Conditions/Prerequisites

Prior to initial core loading.

Test Method

Tests will be performed to the extent practical to verify proper operation of system components, instrumentation, and controls for the Boron Recovery System.

Acceptance Criteria

Instrumentation, controls, alarms and interlocks operate as required in accordance with Updated FSAR Subsection 9.3.5 in response to normal or simulated input signals.

Flow paths to system components have been demonstrated.

Acceptance Test Abstracts

19. STEAM GENERATOR BLOWDOWN SYSTEM

Objective

To demonstrate proper operation of the Steam Generator Blowdown System.

Plant Conditions/Prerequisites

Prior to hot functional testing, and prior to initial core loading.

Test Method

Prior to hot functional tests, testing will be performed on system components, instrumentation and controls to the extent practical.

Prior to initial core loading, the overall system operation will be demonstrated.

Acceptance Criteria

Instrumentation, controls, alarms and interlocks operate as required in accordance with Updated FSAR Subsection 10.4.8 in response to normal or simulated input signals.

Flow paths to system components are demonstrated.

Acceptance Test Abstracts

20. WASTE PROCESSING BUILDING HVAC SYSTEM

Objective

To demonstrate the proper operation of the Waste Processing Building HVAC system.

Plant Conditions/Prerequisites

Prior to initial core loading.

Test Method

Tests will be performed to verify proper operation of system components, instrumentation and controls and to verify air flows. Filters will be installed and in-place filter testing conducted in accordance with Regulatory Guide 1.140.

Acceptance Criteria

Demonstration that Waste Processing Building HVAC performs in accordance with design, as stated in Updated FSAR Section 6.5 and Subsection 9.4.4.

Acceptance Test Abstracts

21. CONTAINMENT COOLING SYSTEMS

Objective

To demonstrate proper operation of the containment cooling systems.

Plant Conditions/Prerequisites

Prior to and during hot functional tests.

Test Method

Containment cooling and control rod drive mechanism cooling subsystems will be tested to demonstrate equipment and controls functional operability.

Individual and integrated air flow verification will be performed.

During hot functional tests, subsystems will be operated and sufficient data recorded and evaluated to ascertain ability to satisfy containment temperature requirements.

Acceptance Criteria

Containment structure cooling subsystems have been demonstrated capable of establishing and maintaining containment temperature(s) as required in Updated FSAR Subsection 9.4.5.

Acceptance Test Abstracts

22. CONTAINMENT PURGE SYSTEM

Objective

To demonstrate proper operation of the Primary Containment Purge System.

Plant Conditions/Prerequisites

Prior to initial core loading.

Test Method

Containment purge subsystems equipment and controls operability will be verified.

Containment "online" and "pre-entry/refueling" purge will be operated with heating equipment (as necessary) and in conjunction with containment recirculation filter system to demonstrate ability to control temperature levels as required by design.

Filters will be installed and in-place verification tests will be conducted to satisfy Regulatory Guide 1.140.

Acceptance Criteria

Containment purge systems have been demonstrated capable of required heating and atmosphere cleanup as required in Updated FSAR Subsection 9.4.5.

Acceptance Test Abstracts

24. ELECTRICAL PENETRATION AREA AIR CONDITIONING SYSTEM

Objective

To demonstrate proper operation of the electrical penetration area air conditioning system.

Plant Conditions/Prerequisites

Prior to and during hot functional testing.

Test Method

Tests will be performed to verify the proper operation of system components and controls for the electrical penetration area air conditioning system. During hot functional testing, data will be recorded to verify satisfactory area temperatures.

Acceptance Criteria

The electrical penetration area air conditioning system operates in accordance with Updated FSAR Subsection 9.4.7.

Acceptance Test Abstracts

25. TURBINE BUILDING VENTILATION

Objective

To demonstrate proper operation of the Turbine Building ventilation system.

Plant Conditions/Prerequisites

Prior to initial core loading.

Test Method

Turbine building ventilation system functional tests will be performed to verify equipment operation and air flows.

Acceptance Criteria

The Turbine Building ventilation system operates in accordance with United Engineers & Constructors Inc., System Design Description for Turbine Building Heating and Ventilating System (SD-45).

Acceptance Test Abstracts

26. ROD CONTROL SYSTEM

Objective

To verify the proper operation of the rod control system logic and power supplies.

Plant Conditions/Prerequisites

Prior to initial core loading.

Test Method

To the extent practical, rod control system tests will be performed to verify proper operation of system logic and associated alarm and interlock functions. The operation of the rod control motor-generator sets will be demonstrated.

Acceptance Criteria

The rod control system logic, interlocks indication and power supplies have demonstrated the design requirements specified in Updated FSAR Section 7.7.

Acceptance Test Abstracts

27. INDIVIDUAL ROD POSITION INDICATION SYSTEM

Objective

To verify proper operation of the digital rod position indication (DRPI) system logic.

Plant Conditions/Prerequisites

Prior to initial core loading.

Test Method

To the extent practical, tests will be performed to verify proper operation of the DRPI logic.

Acceptance Criteria

The digital rod position indication system logic and indication has been demonstrated to perform the design function required by Updated FSAR Section 7.7.

Acceptance Test Abstracts

28. COMPUTER

Objective

To demonstrate the operation of the plant computer and the associated software.

Plant Conditions/Prerequisites

Prior to initial criticality.

Test Method

Tests will be performed to demonstrate computer interface with plant parameters and computer response to changing variables.

Annunciators for reactor control and engineered safety features will be demonstrated operable.

Acceptance Criteria

Computer interface with plant parameters and the computer response to changing parameters are in accordance with computer Input/Output List, DWG-M-510004.

Acceptance Test Abstracts

29. PRIMARY PLANT INSTRUMENTATION

Objective

To verify the initial calibration of the primary plant instrumentation.

Plant Conditions/Prerequisites

Prior to hot functional testing for specified instruments, otherwise prior to initial core loading.

Test Method

The calibration and alignment of various primary plant instrumentation (temperature, pressure, level, flow) will be performed to verify the operation of each instrument and associated setpoints. Plant calibration procedures will be used to the maximum extent practical.

Acceptance Criteria

The primary plant instrumentation has been calibrated to within the setpoint accuracies required by Technical Specifications Tables 2.2-1 and 3.3-4.

Acceptance Test Abstracts

30. RADIATION MONITORING SYSTEM

Objective

To verify proper operation of the process, area, and airborne radiation monitors.

Plant Conditions/Prerequisites

Prior to initial core loading. Fuel storage building monitors will be tested prior to new fuel storage.

Test Method

The radiation monitors will be calibrated and functionally tested to demonstrate proper operation of the channels and any associated interlock and alarm functions.

Acceptance Criteria

The Radiation Monitoring System and associated indicator and interlocks have been demonstrated to perform the design requirements specified in Updated FSAR Section 11.5.

Acceptance Test Abstracts

31. SEISMIC MONITORING SYSTEM

Objective

To verify the proper operation of the seismic monitoring instrumentation.

Plant Conditions/Prerequisites

Prior to initial core loading.

Test Method

The seismic monitoring instrumentation will be calibrated and functionally tested.

Acceptance Criteria

The operability of the Seismic Monitoring System and its associated indication will be demonstrated as meeting the requirements of Technical Specification Table 3.3-7.

Acceptance Test Abstracts

33. COMMUNICATIONS SYSTEM

Objective

To verify proper operation of the plant page and sound powered phone systems. To verify operability of other plant communications that are utilized in the facility emergency plan.

Plant Conditions/Prerequisites

Prior to initial core loading.

Test Method

Communications will be verified between stations, and the outputs of speakers and amplifiers will be adjusted as required.

Acceptance Criteria

The communications system operates in accordance with Updated FSAR Subsections 9.5.2.2a.2 and 9.5.2.2a.3.

Acceptance Test Abstracts

34. EMERGENCY LIGHTING

Objective

To demonstrate the operation of the emergency lighting system.

Plant Conditions/Prerequisites

Prior to initial core loading.

Test Method

Tests will be performed to demonstrate the operation of the emergency lighting systems during partial and total loss of AC power.

Acceptance Criteria

The emergency lighting system operates in accordance with Updated FSAR Subsection 9.5.3.2c.

Acceptance Test Abstracts

35. POLAR CRANE

Objective

To demonstrate the proper operation of the containment polar crane.

Plant Conditions/Prerequisites

Prior to initial core loading. Dynamic and static load tests of the polar crane and associated lifting and rigging equipment has been performed. Static testing at 125 percent of rated load and full operational testing at 100 percent of rated load has been performed.

Test Method

Functional tests will be performed to demonstrate proper operation of the crane and its controls. The operation of interlocks and safety devices will be verified.

Acceptance Criteria

The polar crane operates in accordance with Updated FSAR Subsection 9.1.4.3a.6.

TABLE 14.2-5 STARTUP TEST ABSTRACTS

Index

Index

Startup Test Abstracts

1. STARTUP PROGRAM ADMINISTRATION

Objective

To provide general guidance for the administration of the initial startup test program and a recommended sequence for the conduct of startup testing.

Plant Conditions/Prerequisites

A list of general precautions for the overall test program are presented. General plant conditions are specified in the test sequence with specific requirements delineated in individual tests.

Test Method

General program information and guidelines, including personnel duties and responsibilities are outlined in the Startup Test Program Description. ST-1 presents precautions and guidelines for actual test performance, test sequencing, and power escalations. Specific instructions are given in individual tests. A recommended sequence of testing is included, along with test program holdpoints.

Acceptance Criteria

A recommended sequence of startup testing has been developed. Administrative guidelines for startup testing have been provided.

Startup Test Abstracts

2. PRIMARY SOURCE INSTALLATION

Objective

To provide detailed instructions for the handling and installation of the primary sources into the required fuel assemblies.

Plant Conditions/Prerequisites

Prior to initial core loading.

Test Method

Instructions include a sequence of steps for unloading the shipping cask and installation of the sources into the respective fuel assemblies.

Acceptance Criteria

The primary sources are loaded in the required fuel assemblies.

Startup Test Abstracts

3. CORE LOADING PREREQUISITES

Objective

To provide a detailed list of plant conditions, systems, and equipment necessary for a safe and controlled core loading.

Plant Conditions/Prerequisites

Plant conditions are established as required by test instructions.

Test Method

A detailed list will summarize plant system and equipment status required prior to the start of core loading. In addition, sampling of reactor coolant and associated auxiliary systems will be performed to verify uniform boron concentration and the alignment, calibration, and response of the temporary core loading instrumentation will be verified. Final functional testing of the Reactor Protection System will be verified.

Acceptance Criteria

All requirements specified in the procedure have been completed.

Startup Test Abstracts

4. INITIAL CORE LOADING

Objective

To provide detailed instructions for the conduct of initial core loading in a safe, controlled manner.

Plant Conditions/Prerequisites

Required preoperational testing is complete and plant systems are operational as required by the core loading prerequisites.

Test Method

A detailed loading sequence giving specific fuel assembly identification numbers and core locations will be provided with appropriate data taking requirements. Specific administrative control and core monitoring, procedures to be applied during initial fuel loading will be provided.

Acceptance Criteria

Detailed core loading instructions, including a sequence, have been developed and executed.

Startup Test Abstracts

5. CONTROL ROD DRIVE MECHANISM OPERATIONAL TEST

Objective

To demonstrate the proper operation of the full-length control rod drive mechanisms (CRDM) and provide verification of proper slave cycler timing.

Plant Conditions/Prerequisites

Prior to initial criticality, during cold shutdown or hot standby conditions as required by the test instructions.

Test Method

During cold shutdown, the ability of the slave cycler devices to supply the proper operating signals to the CRDM stepping magnet coils will be confirmed. The proper operation of each CRDM during both cold and hot plant conditions will be verified by recording CRDM magnet coil currents and audio signals.

Acceptance Criteria

CRDM operation conforms to the requirements of proper mechanism operation as described in Chapter 4 of the Westinghouse Magnetic Control Rod Drive Mechanism Technical Manual.

Startup Test Abstracts

6. ROD CONTROL SYSTEM

Objective

To demonstrate that the full-length Rod Control System performs the required control and indication functions just prior to initial criticality.

Plant Conditions/Prerequisites

Prior to initial criticality at no load operating temperature and pressure.

Test Method

Testing of control rod withdrawal and insertion speeds and sequences, control functions, status lights, and indication will be performed to verify proper operation.

Acceptance Criteria

The Rod Control System performs the required control and indication functions in accordance with Chapter 3 of the Westinghouse Rod Control System Technical Manual.

Startup Test Abstracts

7. ROD DROP TIME MEASUREMENT

Objective

To determine the drop time of each full length control rod under various plant conditions.

Plant Conditions/Prerequisites

Prior to initial criticality, during hot standby conditions with full flow in the Reactor Coolant System.

Test Method

During each of the applicable plant conditions, the drop time for each rod control cluster assembly will be determined. Those control rods whose drop times fall outside the two-sigma limit determined from the data for all control rods will be retested at least three times to ensure proper performance.

Acceptance Criteria

The rod drop times meet the requirements given in Technical Specifications, Section 3.1.3.4.

Startup Test Abstracts

8. ROD POSITION INDICATION

Objective

To verify that the Rod Position Indication System performs the required indication functions for each individual rod and to demonstrate that each control rod operates satisfactorily over its entire range of travel.

Plant Condition/Prerequisites

Prior to initial criticality during hot standby conditions.

Test Method

Each control bank will be fully withdrawn in 24 step increments. Each shutdown bank will be fully withdrawn, stopping at 18, 210, and 228 steps. Individual rod position indication and group step indication data is recorded 43 at each bank holdpoint.

Acceptance Criteria

The Rod Position System meets the requirements of Technical Specifications 3.1.3.2 and functions as described in the Westinghouse Digital Rod Position Indication Technical Manual.

Startup Test Abstracts

9. PRESSURIZER SPRAY AND HEATER CAPABILITY TEST

Objective

To establish the continuous spray flow rate and to verify pressurizer spray and heater effectiveness.

Plant Conditions/Prerequisites

Prior to initial criticality during hot standby conditions.

Test Method

The spray bypass valves are adjusted for the minimum continuous spray flow. Both spray valves are opened to initiate a pressure transient which is recorded and compared to the expected pressure response. All heaters are energized to initiate a pressure transient which is recorded and compared to expected pressure response.

Acceptance Criteria

The continuous spray flow has been set, and the spray and heater effectiveness is in accordance with the Westinghouse performance curves as attached to ST-9.

Startup Test Abstracts

10. RESISTANCE TEMPERATURE DETECTOR BYPASS LOOP FLOW VERIFICATION

Objective

To calculate the hot and cold leg bypass line flow rates necessary to provide adequate transport times, to determine the actual flow rates, and to verify the low flow alarm setpoints.

Plant Conditions/Prerequisites

Prior to initial criticality during hot standby conditions.

Test Method

The required bypass loop flows will be calculated from measurements taken on the installed RTD piping and compared to the measured bypass loop flow rates. The hot leg bypass loop isolation valves will be throttled to verify the low flow alarm setpoints.

Acceptance Criteria

The measured flow rates meet the calculated values, with regard to transport times, as defined by the test procedure. The low flow alarms actuate at the setpoint shown in the Westinghouse Precautions, Limitations, and Setpoints Manual.

Startup Test Abstracts

11. REACTOR COOLANT SYSTEM FLOW MEASUREMENT

Objective

To measure actual reactor coolant system flow.

Plant Conditions/Prerequisites

Prior to initial criticality during hot standby conditions.

Test Method

Measurements will be made of elbow tap differential pressure for each loop. This data will be used to obtain a measurement of actual reactor coolant system flow.

Acceptance Criteria

The calculated reactor coolant system flow rate is greater than the thermal design flow shown in Table 5.1-1 of the Updated FSAR.

Startup Test Abstracts

12. REACTOR COOLANT SYSTEM FLOW COASTDOWN

Objective

To measure the rate at which reactor coolant flow changes following various reactor coolant pump trips and to determine delay times associated with the loss of flow accident.

Plant Conditions/Prerequisites

Prior to initial criticality during hot standby conditions.

Test Method

The reactor coolant pumps will be simultaneously tripped from various operating configurations. Data will be recorded for coolant loop differential pressure, coolant pump breaker position, low flow trip relay output, reactor trip breaker position, as required by the test procedure.

Acceptance Criteria

The reactor coolant system flow coastdown rate and measured time delays are conservative with respect to those used in Section 15.3 of the Updated FSAR.

Startup Test Abstracts

13. OPERATIONAL ALIGNMENT OF NUCLEAR INSTRUMENTATION

Objective

To determine voltage settings, trip settings, operational settings, alarm settings, and overlap for the source, intermediate, and power range instrumentation.

Plant Conditions/Prerequisites

Portions of this test will be performed prior to core loading, prior to initial criticality, at hot zero power conditions, and during each of the major power plateaus (30%, 50%, 75%, 100%) as required by the test instructions.

Test Method

The nuclear instrumentation will be calibrated and functionally tested using permanently installed control and adjustment mechanisms. The operational settings for the various ranges will be adjusted for their proper function during the applicable portions of the startup program.

Acceptance Criteria

The voltage settings, operational settings, alarm settings, and trip settings have been determined and are within the range shown in Chapter 5 of the Westinghouse Nuclear Instrumentation System Technical Manual and

Technical Specification 3.3.1.

Startup Test Abstracts

14. OPERATIONAL ALIGNMENT OF PROCESS TEMPERATURE INSTRUMENTATION

Objective

To align the ΔT and T_{avg} process instrumentation.

Plant conditions/Prerequisites

As required by the test instructions, portions of this alignment will be performed prior to initial criticality at 30%, 50%, 75% and 100% power plateaus.

Test Method

During plant heatup, installation correction factors will be determined for the RCS RTDS and incore thermocouples.

The ΔT and T_{avg} instrumentation will be aligned at isothermal conditions prior to criticality and at approximately 75% power. An extrapolation of the 75% power data will be made for the 100% power values of ΔT and T_{avg}. At or near full power an alignment check will be performed and any necessary readjustments will be made.

Acceptance Criteria

The ΔT and T_{avg} process instrumentation have been aligned.

Startup Test Abstracts

15. REACTOR PLANT SYSTEMS SETPOINT VERIFICATION

Objective

To verify that initial setpoint adjustments have been made prior to plant startup and to maintain a record of setpoints which required readjustment during initial startup testing.

Plant Conditions/Prerequisites

Prior to initial criticality and following full power testing.

Test Method

Verify and record initial setpoint values and any changes performed during initial startup testing.

Acceptance Criteria

The initial setpoints are verified to be in agreement with the settings and tolerances specified by the Westinghouse Precautions, Limitations and Setpoints Manuals, and any adjustments made during initial startup testing are recorded.

Startup Test Abstracts

16. INITIAL CRITICALITY

Objective

To achieve initial criticality in a controlled manner.

Plant Conditions/Prerequisites

The plant is in hot standby conditions and all required portions of the startup testing program have been completed.

Test Method

All control rods will be fully withdrawn except for the controlling bank which will be withdrawn to a preselected position. A controlled dilution will be performed until criticality is achieved. At periodic points during the rod withdrawal and dilution, data will be taken and inverse count rate ratio plots made to enable extrapolation of the expected critical point. Following criticality, the power level for physics testing will be determined and the operation of the reactivity computer will be verified.

Acceptance Criteria

The reactor is critical. The average absolute deviation between indicated reactivity on the reactivity computer and the theoretical values is within the NSSS vendor's guidelines.

Startup Test Abstracts

17. BORON ENDPOINT MEASUREMENTS

Objective

To determine the critical reactor coolant system boron concentration appropriate for a specific control rod endpoint configuration.

Plant Conditions/Prerequisites

The plant is critical at hot zero power conditions and at the control rod configuration specified by the startup sequence.

Test Method

The boron endpoints will be determined by measuring the boron concentration of the Reactor Coolant System at or near the desired control rod configuration. If required the rods are quickly moved to the desired configuration with no boron adjustment. The change in reactivity is measured and converted to an equivalent amount of boron to yield the endpoint at that rod configuration. The data obtained will be utilized to determine the boron worth.

Acceptance Criteria

The calculated boron worth agrees with the value contained in the Westinghouse Nuclear Design Report for Cycle 1.

Startup Test Abstracts

18. ISOTHERMAL TEMPERATURE COEFFICIENT MEASUREMENT

Objective

To determine the isothermal temperature coefficient.

Plant Conditions/Prerequisites

The plant is critical at hot zero power conditions and at the control rod configuration specified by the startup sequence.

Test Method

The isothermal temperature coefficient will be determined by alternately heating up and cooling down the Reactor Coolant System at constant rates while data on reactivity and reactor coolant temperatures are obtained.

Acceptance Criteria

The measured values of the isothermal temperature coefficient meet the requirements of the Westinghouse Nuclear Design Report for Cycle 1.

Startup Test Abstracts

19. FLUX DISTRIBUTION MEASUREMENTS AT LOW POWER

Objective

To measure the reactor core flux distribution at low power.

Plant Conditions/Prerequisites

The plant is at a low power level (less than 5%) at the control rod configuration specified by the startup sequence.

Test Method

The flux distribution will be obtained by analysis of data acquired by means of the Incore Movable Detector System.

Acceptance Criteria

Flux map results are in agreement with the predicted distributions contained in the Westinghouse Nuclear Design Report for Cycle 1.

Startup Test Abstracts

20. CONTROL ROD WORTH MEASUREMENT

Objective

To determine the differential and integral reactivity worth of an individual control rod bank and to ensure proper shutdown margin.

Plant Conditions/Prerequisites

The plant is critical at zero power at the control rod configuration specified by the startup sequence.

Test Method

Control rod worths will be obtained by maintaining a constant boron addition or dilution and compensating for the reactivity change by rod movement. These changes in reactivity are recorded by a reactivity computer and analyzed to obtain the control rod worths. Analysis of the collected data will be performed to confirm adequate shutdown margin. Additional control rod worth measurements may be conducted using the rod swap technique.

Acceptance Criteria

The measured control rod worths are conservative with respect to the values assumed in Chapter 15 of the Updated FSAR.

Startup Test Abstracts

21. PSEUDO ROD EJECTION TEST

Objective

To verify the conservatism of the ejected rod analysis at zero power.

Plant Conditions/Prerequisites

The test will be performed at zero power.

Test Method

The selected RCCA will be fully withdrawn while compensating for the reactivity change by boron additions as necessary. A flux map will be taken to measure the resulting flux distribution.

Acceptance Criteria

Analysis of the data obtained yields rod worths and hot channel factors which are conservative with respect to the values assumed in Subsection 15.4.8 of the Updated FSAR.

Startup Test Abstracts

22. NATURAL CIRCULATION TEST

Objective

To verify the ability of the Reactor Coolant System to remove heat by means of natural circulation.

Plant Conditions/Prerequisites

The plant is subcritical with sufficient decay heat to demonstrate natural circulation.

Test Method

With the plant at hot standby conditions, the reactor coolant pumps will be tripped. This test will determine the length of time necessary to stabilize natural circulation and will demonstrate the reactor coolant flow distribution by obtaining incore thermocouple maps. Data will be collected during the test to verify simulator modeling.

Acceptance Criteria

Natural circulation is established and maintained as indicated by stable temperature indication.

Startup Test Abstracts

23. DYNAMIC AUTOMATIC STEAM DUMP CONTROL

Objective

To verify proper operation of the T_{avg} Steam Dump Control System, to demonstrate the dynamic characteristics of the controller, and to obtain the final settings for steam pressure control of the condenser dump valves.

Plant Conditions/Prerequisites

The plant is critical at no load temperature and pressure.

Test Method

Reactor power will be increased to approximately 5% by rod withdrawal with either a simulated plant trip or load rejection to demonstrate proper operation and setpoint adequacy of the T_{avg} controllers. With the Steam Dump System in the pressure control mode, power will be increased to demonstrate proper operation of the steam header pressure controller. Adjustment of controller gains and/or setpoints will be made as necessary.

Acceptance Criteria

The Steam Dump Control System operates as described in Updated FSAR Subsection 7.7.1.8, and is capable of maintaining the reactor coolant system temperature at the no load temperature.

Startup Test Abstracts

24. AUTOMATIC REACTOR CONTROL

Objective

To verify the capability of the Reactor Control System to maintain the reactor coolant average temperature within acceptable limits.

Plant Conditions/Prerequisites

The plant is stable at the 30% power plateau.

Test Method

T_{avg} will be varied from the T_{ref} setpoint and the control system will be placed in automatic to verify its ability to return plant temperature to the reference value.

Acceptance Criteria

The reactor control system functions as described in Updated FSAR Subsection 7.7.1.1.

Startup Test Abstracts

25. AUTOMATIC STEAM GENERATOR LEVEL CONTROL

Objective

To verify the stability of the Automatic Steam Generator Level Control System following simulated transients at low power and to verify the operation of the Main Feed Pump Control System.

Plant Conditions/Prerequisites

The plant is stable at the 30% power plateau.

Test Method

Steam generator level transients will be simulated to verify proper level control response. The operability of the Main Feed Pump Control System will be verified by manipulation of the controllers and by simulating selected input signals.

Acceptance Criteria

The Steam Generator Level and Main Feedwater Pump Control Systems function as described in Updated FSAR Subsection 7.7.1.7.

Startup Test Abstracts

26. THERMAL POWER MEASUREMENT AND STATEPOINT DATA COLLECTION

Objective

To obtain various primary and secondary plant temperatures, pressures, and flows and to perform a calorimetric determination of reactor power and verify that the Main Steam and Feedwater Systems perform as described in the Updated FSAR.

Plant Conditions/Prerequisites

This test will be performed at each of the major power plateaus (30%, 50%, 75%, 100%) as required by the startup test sequence.

Test Method

With the plant stable at the required power level, data will be collected to allow calculation of thermal power. Additional statepoint data will be collected to provide baseline plant operating temperatures and pressures. Some of the data collected will be utilized by other tests to align various plant instruments.

Acceptance Criteria

The data specified in the procedure has been collected and calorimetric performed. Main Steam and Feedwater Systems operate as described in Updated FSAR Subsection 10.4.7.

Startup Test Abstracts

27. STARTUP ADJUSTMENTS OF REACTOR CONTROL SYSTEM

Objective

To determine the T_{avg} program resulting in the optimum plant efficiency without exceeding plant pressure and temperature limitations.

Plant Conditions/Prerequisites

Portions of this test will be performed at hot zero power and various major power plateaus (50%, 75%, 100%) as required by the startup test sequence.

Test Method

Analysis of system temperature and pressure data obtained by this or other tests at the required plant conditions will be used to provide a basis for the adjustment of the Reactor Control System.

Acceptance Criteria

The Reactor Control System has been adjusted to provide optimum plant performance without exceeding the requirements of Technical Specification 3.2.5.

Startup Test Abstracts

28. CALIBRATION OF STEAM AND FEEDWATER FLOW INSTRUMENTATION

Objective

To calibrate the steam and feedwater flow instruments.

Plant Conditions/Prerequisites

Portions of this test will be performed at hot zero power conditions and at selected major power plateaus (30%, 50%, 75% 90% and 100%) as required by the startup test sequence.

Test Method

Permanent plant feedwater flow transmitters will be calibrated using station procedures which are based on laboratory test data. The feedwater flow transmitters will then be used to determine measured flow at all selected power plateaus. To ensure accurate readings, calibration checks will be conducted prior to and following each plateau.

Data will be obtained during the performance of ST-26, Thermal Power Measurement and Statepoint Data Collection, under very stable plant conditions. Data for feedwater and steam flow will be the raw transmitter voltages (1-5VDC). Voltages will be converted to lb_m/hr using the latest individual transmitter scaling, where 5VDC equals 5 X 10^6 lb_m/hr.

Steam flow readings will be compared to the calibrated feedwater flow readings. Adjustments will be made to the steam flow transmitters based on this comparison to obtain a best fit of the data (±2% steam/feedwater flow mismatch at full span).

Acceptance Criteria

The steam and feedwater flow instrumentation has been calibrated.

Startup Test Abstracts

29. CORE PERFORMANCE EVALUATION

Objective

To provide instructions for obtaining incore movable detector flux and thermocouple maps at power and to verify proper core performance margins.

Plant Conditions/Prerequisites

This test will be performed at selected power plateaus (30%, 50%, 75%, 90%, 100%) as required by the startup test sequence.

Test Method

Flux distribution data will be obtained utilizing the movable detector system. Incore thermocouple data will be obtained using the analog readout instrument or process computer. This data will be analyzed to indicate core performance.

Acceptance Criteria

The flux map results, including DNBR, radial and axial powder peaking factors, and quadrant power tilt, meet the requirements of Technical Specifications 3.2.2, 3.2.4, and 3.2.5.

Startup Test Abstracts

30. POWER COEFFICIENT MEASUREMENT

Objective

To verify the design prediction of the power coefficient.

Plant Conditions/Prerequisites

This test will be performed at selected power plateaus (30%, 50%, 75%, 100%) as specified by the startup test sequence.

Test Method

Generator load will be varied and data will be collected for ΔT, T_{avg}, and reactor power. Analysis of this data will be correlated to the power coefficient. This inferred actual power coefficient will be compared to the predicted power coefficient.

Acceptance Criteria

The average measured power coefficient verification factor shall be within ±.5°F/% of the predicted power coefficient verification factor.

Startup Test Abstracts

33. SHUTDOWN FROM OUTSIDE THE CONTROL ROOM

Objective

To demonstrate the capability to shutdown and maintain the reactor in a hot standby condition from outside the control room.

Plant Conditions/Prerequisites

The plant is at a stable power level of equal to or greater than 10% power.

Test Method

The plant will be tripped from a location external to the control room and maintained in the stable hot standby condition for at least 30 minutes. Control will then be transferred back to the control room.

Acceptance Criteria

The plant has been tripped from a location external to the control room. The plant has been maintained in a stable hot standby condition for at least 30 minutes.

Startup Test Abstracts

34. LOAD SWING TEST

Objective

To verify proper plant response, including automatic control system performance, to 10% step load changes.

Plant Conditions/Prerequisites

This test will be initiated from steady state conditions at selected power plateaus (30%, 50%, 75%, 100%) as required by the startup test sequence.

Test Method

Turbine generator output will be changed as rapidly as possible to achieve an approximate 10% load decrease or increase, as required. Various plant parameters will be recorded or observed during the transient.

Acceptance Criteria

The following criteria will be used to determine successful test completion:

1. Reactor or turbine must not trip.
2. Safety injection is not initiated.
3. Neither steam generator relief valves nor safety valves lift.
4. Neither pressurizer relief valves nor safety valves lift.
5. No manual intervention should be necessary to bring plant conditions to steady state.
6. Plant parameters should not incur sustained or diverging oscillations.
7. Nuclear power overshoot (undershoot) must be less than 3% for load increase (decrease).

Note: Failure to meet these criteria does not constitute a need for stopping the test program, but correction of any deficiencies should be accomplished as required consistent with the plant schedule.

Startup Test Abstracts

35. LARGE LOAD REDUCTION TEST

Objective

To verify proper plant response, including automatic control system performance, to a 50% load reduction.

Plant Conditions/Prerequisites

This test will be initiated from steady state conditions at the 75% and 100% power plateaus.

Test Method

Turbine generator output will be reduced as rapidly as possible to achieve an approximate 50% load reduction. Various plant parameters will be observed or recorded during the transient.

Acceptance Criteria

The following criteria will be used to determine successful test completion:

1. Reactor or turbine must not trip.
2. Safety injection is not initiated.
3. Steam generator safety valves should not lift.
4. Pressurizer safety valves should not lift.
5. No manual intervention should be necessary to bring plant conditions to steady state.

Note: Failure to meet these criteria does not constitute a need for stopping the test program, but correction of any deficiencies should be accomplished as required consistent with the plant schedule.

Startup Test Abstracts

36. AXIAL FLUX DIFFERENCE INSTRUMENTATION CALIBRATION

Objective

To determine the relationship between the excore detector currents and incore axial flux difference and to derive the calibration factors for the F(ΔI) component of the ΔT reactor trip setpoints and the ΔI instrumentation.

Plant Conditions/Prerequisites

This test will be performed at the 75% power plateau.

Test Method

The Incore Movable Detector System will be used at various axial offsets to obtain flux distribution data from which axial flux differences can be obtained. This data in conjunction with excore detector current data, taken during the mapping, will generate the incore-excore relationships from which the ΔI instrumentation can be calibrated.

Acceptance Criteria

The relationship between incore axial offset and ΔI has been determined and the ΔI instrumentation has been calibrated.

Startup Test Abstracts

37. STEAM GENERATOR MOISTURE CARRYOVER MEASUREMENT

Objective

To determine the moisture carryover performance of the steam generators.

Plant Conditions/Prerequisites

This test will be performed at 100% power, as required by the startup test sequence.

Test Method

A tracer will be injected into the steam generator and an analysis of selected water and steam samples will be performed. These results will be used to calculate the steam generator moisture carryover.

Acceptance Criteria

The steam generator moisture carryover has been calculated.

Startup Test Abstracts

38. UNIT TRIP FROM 100% POWER

Objective

To verify the ability of the plant to sustain a trip from 100% power and return to stable conditions following the transient, and to determine the overall response time of the reactor coolant hot leg resistance temperature detectors (RTD).

Plant Conditions/Prerequisites

This test will be initiated from steady-state conditions at the 100% power plateau.

Test Method

A plant trip will be initiated by tripping the generator main breaker. Plant response will be monitored and plant parameters will be recorded as required. This data will be evaluated to determine if changes in control system settings are required to improve system response.

Acceptance Criteria

The following criteria will be used to determine acceptance:

1. Pressurizer safety valves shall not lift.
2. Steam generator safety valves shall not lift.
3. Safety injection is not initiated.
4. The overall hot leg RTD response time is conservative with respect to the value used in Updated FSAR Chapter 15.

Startup Test Abstracts

39. LOSS OF OFFSITE POWER TEST

Objective

To demonstrate starting of emergency diesels and proper sequencing of loads following a main generator trip without an available source of offsite power.

Plant Conditions/Prerequisites

The plant is at a stable power level of equal to or greater than 10% power.

Test Method

Generator output breakers will be tripped resulting in a reactor trip with no offsite power available. The starting of the emergency diesel generators and overall plant response will be monitored.

The loss of offsite power will be maintained long enough for plant systems to stabilize (at least 30 minutes or longer).

Acceptance Criteria

The diesel generators reach rated voltage and frequency within the limits specified in Technical Specifications 4.8.1.1.2a. The emergency power sequencers function as described in Updated FSAR Section 8.3.

Startup Test Abstracts

40. NSSS ACCEPTANCE TEST

Objective

To demonstrate reliability of the NSSS at rated power and to measure the NSSS output at its warranted rating.

Plant Conditions/Prerequisites

The plant will be at rated full power.

Test Method

The NSSS will be maintained at its rated thermal output (+0% -5% for a specified period of time to demonstrate reliability. Steady-state conditions will be established as close as possible to warranty conditions and appropriate data recorded to allow determination of plant performance.

Acceptance Criteria

The reliability of the NSSS has been demonstrated by operating at or near full power, as mutually agreed by the owner and NSSS vendor, for a specified period of time, and the NSSS is capable of developing the warranted output as calculated during the performance measurement.

Startup Test Abstracts

41. RADIATION SURVEY

Objective

To determine dose rate levels at preselected locations throughout the plant, and to identify high radiation areas. To verify operability of selected area radiation monitors.

Plant Conditions/Prerequisites

The plant is at steady state conditions at selected power levels (HZP, 50%, 100%) as specified by the startup test sequence.

Test Method

Radiation surveys will be made during steady-state plant conditions to determine gamma and neutron dose levels at preselected points throughout the plant. The response of area radiation monitors will be compared with the survey readings.

Acceptance Criteria

Neutron and gamma radiation dose levels have been measured at various preselected locations. The response of selected area radiation monitors agrees with values obtained during ST-41. High radiation areas have been identified.

Startup Test Abstracts

42. WATER CHEMISTRY CONTROL

Objective

To demonstrate that chemical and radiochemical control and analysis systems function as described in the Updated FSAR and verify that water chemistry requirements can be maintained at various plant conditions.

Plant Conditions/Prerequisites

This test will be performed prior to criticality and at major power plateaus (HZP, 30%, 50%, 75%, 100%) as specified by the startup test sequence.

Test Method

Samples of reactor coolant will be analyzed to verify that primary chemistry requirements can be maintained. During power operation, samples of secondary plant water will also be obtained to verify that chemistry specifications are met. These results will be compared with those from selected analyzers to demonstrate proper operation.

Acceptance Criteria

Control and alarm systems function as described in Updated FSAR Subsections 9.3.2 and 9.3.4, and water chemistry is maintained within limits established by "Westinghouse Guidelines for Secondary Water Chemistry" and Technical Specifications 3.4.7 and 6.7.4c, as well as Updated FSAR Table 5.2-5. Analyzer responses agree with analysis results.

Startup Test Abstracts

43. PROCESS COMPUTER

Objective

To verify the process computer is receiving correct inputs from process variables and performing related calculations correctly.

Plant Conditions/Prerequisites

This test will be performed at major power plateaus (30%, 50%, 75%, 100%) as specified by the startup test sequence.

Test Method

Computer outputs for various plant parameters will be compared with the values indicated by plant process instrumentation.

Acceptance Criteria

The process computer inputs and process instrumentation agree and the related calculations are being performed correctly.

Startup Test Abstracts

44. LOOSE PARTS MONITOR

Objective

To obtain baseline data for the Loose Parts Monitoring System (LPMS) and to establish the alert levels for power operation.

Plant Conditions/Prerequisites

This test will be performed prior to initial criticality at cold and hot plant conditions and at selected power plateaus (50%, 100%) as specified by the startup test sequence.

Test Method

Accelerometer data will be obtained at various plant conditions to establish a set of baseline data for the plant. Analysis of this data will be used to verify the proper setting of the alert limits.

Acceptance Criteria

Baseline data has been obtained and the alert limits have been established.

Startup Test Abstracts

45. PROCESS AND EFFLUENT RADIATION MONITORING SYSTEM

Objective

To demonstrate the proper operation of the Process and Effluent Radiation Monitoring Systems.

Plant Conditions/Prerequisites

This test will be performed at selected power plateaus (HZPL, 50%, 100%) as specified by the startup sequence.

Test Method

The response of various process and effluent monitors including selected airborne radioactivity monitors will be compared to the analysis of actual samples obtained from the specific monitoring points.

Acceptance Criteria

The Process and Effluent Monitoring Systems operate in accordance with the criteria given in Updated FSAR Section 11.5.

Startup Test Abstracts

46. VENTILATION SYSTEM OPERABILITY TEST

Objective

To demonstrate the ability of various ventilation and air conditioning systems to maintain proper environmental conditions in various equipment spaces under operating conditions.

Plant Conditions/Prerequisites

This test will be performed at the 50% and 100% power plateaus.

Test Method

Ambient temperatures will be monitored in selected plant location including areas containing engineered safety feature equipment to ensure proper environmental conditions are maintained.

Acceptance Criteria

The ventilation systems are capable of maintaining equipment space environmental conditions as described in Updated FSAR Section 9.4.

Startup Test Abstracts

48. TURBINE GENERATOR STARTUP TEST

Objective

To provide instructions for the initial startup and synchronization of the turbine generator and to obtain operational data for the turbine generator during the initial startup and at various loads.

Plant Conditions/Prerequisites

Portions of this test will be performed at selected power plateaus (10%, 30%, 50%, 75%, 100%) as specified by the startup sequence.

Test Method

Detailed instructions will be provided for the initial startup and synchronization of the turbine generator. Data will be recorded for the various turbine parameters during the startup and through the power ascension.

Acceptance Criteria

The turbine generator is synchronized to the grid. Operational data has been collected as specified by ST-48.

Startup Test Abstracts

50. MOVABLE INCORE DETECTOR SYSTEM

Objective

To verify proper installation and operation of the Movable Incore Detector System.

Plant Conditions/Prerequisites

Prior to initial criticality and during low power physics testing.

Test Method

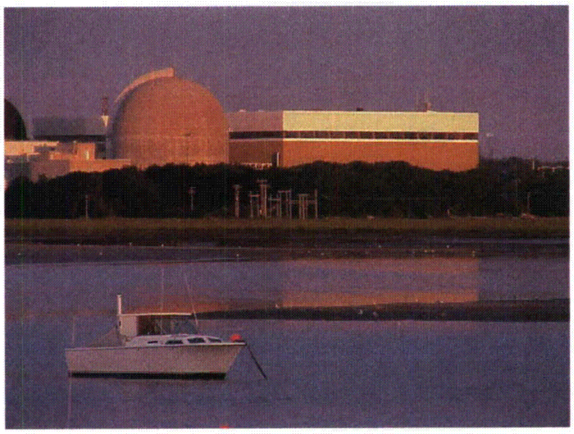
Testing will be performed on the Movable Incore Detector System to verify system performance in all modes of operation. System indexing will be checked using a dummy cable. The system will be operationally checked to ensure free detector passage in all thimbles. The final limit switch settings will be made during initial core flux mapping.

Acceptance Criteria

The Movable Incore Detector System has been demonstrated operational and meets Technical Specification 3.3.3.2.

SEABROOK STATION
UPDATED FINAL SAFETY ANALYSIS REPORT

CHAPTER 15
ACCIDENT ANALYSES



15.0 GENERAL

15.0.1 Classification of Plant Conditions

Since 1970 the American Nuclear Society (ANS) classification of plant conditions has been used which divides plant conditions into four categories in accordance with anticipated frequency of occurrence and potential radiological consequences to the public. The four categories are as follows:

- Condition I: Normal Operation and Operational Transients
- Condition II: Faults of Moderate Frequency
- Condition III: Infrequent Faults
- Condition IV: Limiting Faults.

The basic principle applied in relating design requirements to each of the conditions is that the most probable occurrences should yield the least radiological risk to the public and those extreme situations having the potential for the greatest risk to the public shall be those least likely to occur. Where applicable, reactor trip system and engineered safeguards functioning is assumed to the extent allowed by considerations, such as the single failure criterion, in fulfilling this principle.

15.0.1.1 Condition I - Normal Operation and Operational Transients

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

A typical list of Condition I events are identified below:

a. Steady state and shutdown operations:

1. Power operation (> 5 to 100 percent of rated thermal power)
2. Startup ($K_{eff} \geq$ 0-99, $\leq$ 5 percent of rated thermal power)
3. Hot standby (subcritical, residual heat removal system isolated)

4. Hot shutdown (subcritical, Residual Heat Removal System in operation)
5. Cold shutdown (subcritical, Residual Heat Removal System in operation)
6. Refueling.

b. Operation with permissible deviations:

Various deviations which may occur during continued operation as permitted by the plant Technical Specifications must be considered in conjunction with other operational modes. These include:

1. Operation with components or systems out of service
2. Leakage from fuel with clad defects
3. Radioactivity in the reactor coolant
   (a) Fission products
   (b) Corrosion products
   (c) Tritium
4. Operation with steam generator leaks up to the maximum operational leakage allowed by the Technical Specifications.
5. Testing as allowed by the Technical Specifications.

c. Operational transients:

1. Plant heatup and cooldown (up to 100°F/hour for the Reactor Coolant System; 200°F/hour for the pressurizer
2. Step load changes (up to plus 10 percent and minus 15 percent)
3. Ramp load changes (up to 5 percent/minute)
4. Load rejection up to and including design load rejection transient.

15.0.1.2 Condition II - Faults of Moderate Frequency

These faults, at worst, result in a reactor trip with the plant being capable of returning to operation. By definition, these faults (or events) do not propagate to cause a more serious fault, i.e., Condition III or IV events. In addition, Condition II events are not expected to result in fuel rod failures or Reactor Coolant System or secondary system overpressurization.

For the purposes of this report, the following faults are included in this category:

a. Feedwater System malfunction causing a decrease in feedwater temperature (Subsection 15.1.1) or an increase in feedwater flow (Subsection 15.1.2)

b. Excessive increase in secondary steam flow (Subsection 15.1.3)

c. Accidental depressurization of the Main Steam System (Subsection 15.1.4)

d. Loss of external load (Subsection 15.2.2)

e. Turbine trip (Subsection 15.2.3)

f. Inadvertent closure of main steam isolation valves (Subsection 15.2.4)

g. Loss of condenser vacuum and other events resulting in turbine trip (Subsection 15.2.5)

h. Loss of nonemergency AC power to the station auxiliaries (Subsection 15.2.6)

i. Loss of normal feedwater flow (Subsection 15.2.7)

j. Partial loss of forced reactor coolant flow (Subsection 15.3.1)

k. Uncontrolled rod cluster control assembly bank withdrawal from a subcritical or low power startup condition (Subsection 15.4.1)

l. Uncontrolled rod cluster control assembly bank withdrawal at power (Subsection 15.4.2)

m. Control rod misalignment - Dropped full length assembly, dropped full length assembly bank, or statically misaligned full length assembly (Subsection 15.4.3)

n. Startup of an inactive reactor coolant pump at an incorrect temperature (Subsection 15.4.4)

o. Chemical and Volume Control System (CVCS) malfunction that results in a decrease in the boron concentration in the reactor coolant (Subsection 15.4.6)

p. Inadvertent operation of the Emergency Core Cooling System during power operation (Subsection 15.5.1)

q. CVCS malfunction causing an increase in reactor coolant inventory (Subsection 15.5.2)

r. Inadvertent opening of a pressurizer safety or relief valve (Subsection 15.6.1)

s. Failure of small lines outside Containment (Subsection 15.6.2).

15.0.1.3 Condition III - Infrequent Faults

By definition Condition III occurrences are faults which may occur very infrequently during the life of the plant. They will be accommodated with the failure of only a small fraction of the fuel rods although sufficient fuel damage might occur to preclude resumption of plant operation for a considerable outage time. The release of radioactivity will not be sufficient to interrupt or restrict public use of those areas beyond the exclusion radius. A Condition III fault will not, by itself, generate a Condition IV fault or result in a consequential loss of function of the Reactor Coolant System or containment barriers. For the purposes of this report the following faults are included in this category:

a. Minor steam system piping failure (Subsection 15.1.5)

b. Complete loss of forced reactor coolant flow (Subsection 15.3.2)

c. Control rod misalignment - single Rod Cluster Control Assembly withdrawal at full power (Subsection 15.4.3)

d. Inadvertent loading and operation of a fuel assembly in an improper position (Subsection 15.4.7)

e. Loss of reactor coolant, from small ruptured pipes or from cracks in large pipes, which actuate the Emergency Core Cooling System (Subsection 15.6.5)

f. Waste Gas System failure (Subsection 15.7.1)

g. Radioactive Liquid Waste System leak or failure (atmospheric release) (Subsection 15.7.2)

h. Liquid containing tank failure (Subsection 15.7.3)

i. Spent fuel cask drop accidents (Subsection 15.7.5). [Historical]

15.0.1.4 Condition IV - Limiting Faults

Condition IV occurrences are faults which are not expected to take place, but are postulated because their consequences would include the potential for the release of significant amounts of radioactive material. They are the most drastic which must be designed against and represent limiting design cases. Condition IV faults are not to cause a fission product release to the environment resulting in an undue risk to public health and safety in excess of guideline values of 10 CFR 100. A single Condition IV fault is not to cause a consequential loss of required functions of systems needed to cope with the fault including those of the Emergency Core Cooling System and the Containment. For the purposes of this report the following faults have been classified in this category:

a. Steam system piping failure (Subsection 15.1.5)

b. Feedwater system pipe break (Subsection 15.2.8)

c. Reactor coolant pump shaft seizure (locked rotor) (Subsections 15.3.3 and 15.3.4)

d. Reactor coolant pump shaft break (Subsection 15.3.5)

e. Spectrum of Rod Cluster Control Assembly ejection accidents (Subsection 15.4.8).

f. Loss-of-coolant accidents resulting from the spectrum of postulated piping breaks within the reactor coolant pressure boundary (Subsection 15.6.5)

g. Steam generator tube rupture (Subsection 15.6.3)

h. Fuel handling accidents (Subsection 15.7.4).

15.0.2 Optimization of Control Systems

A control system setpoint study has been performed to simulate performance of the reactor control and protection systems. In this study, emphasis is placed on the development of a control system which will automatically maintain prescribed conditions in the plant even under a conservative set of reactivity parameters with respect to both system stability and transient performance.

Nominal protection system setpoints on which the accident analysis is based are also used in the control system setpoint study. Instrumentation errors are calculated consistent with the method used in the accident analysis. These errors are applied in an adverse direction with respect to maintaining system stability and transient performance. The accident analysis and setpoint study combine to show that the plant can be operated and meet both safety and operability requirements.

For each mode of plant operation, a group of optimum controller setpoints is determined. In areas where the resultant setpoints are different, compromises based on the optimum overall performance are made and verified. A consistent set of control system parameters is derived satisfying plant operational requirements throughout the core life and for various levels of power operation.

The study comprises an analysis of the following control systems: rod control, steam dump, steam generator level, pressurizer pressure and pressurizer level.

15.0.3 Plant Characteristics and Initial Conditions Assumed in the Accident Analyses

15.0.3.1 Design Plant Conditions

Table 15.0-1 lists the principal power rating values which are assumed in analyses performed in this report.

Where initial power operating conditions are assumed in accident analyses, the "guaranteed nuclear steam supply system thermal power output" plus allowance for errors in steady state power determination is assumed. The thermal power values used for each transient analyzed are given in Table 15.0-3.

The values of other pertinent plant parameters utilized in the accident analyses are given in Table 15.0-2.

15.0.3.2 Initial Conditions

Table 15.0-2 and Table 15.0-3 provides a list of conditions representing nominal plant parameters. These parameters also represent a set of initial conditions for the accidents and transients. Uncertainties in these parameters are accounted for either through RTDP or in the initial conditions selected for the transient cases. The following uncertainties are considered:

a.	Core power [+]	0 percent allowance for calorimetric error
b.	Average RCS temperature	± 3.0°F random with a –3.0°F bias[*] allowance for controller deadband and measurement error and steam generator fouling penalty
c.	Pressurizer pressure	± 50 psi allowance for steady-state fluctuations and measurement error

[+] Analysis performed at a reactor thermal power of 3659 MWt +0% calorimetric uncertainty. This value reflects the licensed power of 3648 MWt +0.3% uncertainty starting in Cycle 12.

[*] A negative bias means that the indication is lower than actual.

15.0.3.3 Power Distribution

The power distribution in the core, and in particular, the radial peaking factor ($F_{\Delta H}$) and the total peaking factor (F_q), are of major importance in determining the transient margin. Initial power distributions for the transients are selected from a range of possible conditions within the allowable axial flux difference LCO band. Such a band, corresponding to Wide-band operation for Seabrook Station, is illustrated in Figure 15.0-32. Power distributions used to generate the axial flux difference LCO band consider both steady-state operation and xenon transients.

The radial peaking factor ($F_{\Delta H}$), the total peaking factor (F_q), and the axial flux difference LCO band are controlled through the COLR. Transient power peaking involving rod motion or rod misalignment is explicitly treated on an event-by-event basis.

15.0.3.4 Component Response Times and Capacities

A tabulation of the component response-time and design capacities, as assumed for the various accidents, is presented in Table 15.0-7 and Table 15.0-8.

15.0.3.5 Non-LOCA Accidents

This section summarizes the non-LOCA analyses and evaluations performed to support the SPU program at Seabrook Unit 1.

15.0.3.5.1 Fuel Features

The fuel features which were evaluated are:

a. $ZIRLO_{TM}$ Intermediate Flow Mixing (IFMs) grids;

b. $ZIRLO_{TM}$ mid grids;

c. $ZIRLO_{TM}$ fuel clad;

d. $ZIRLO_{TM}$ instrument and thimble tubes;

e. Removable top nozzles;

f. Protective bottom grids;

g. Debris filter bottom nozzles

15.0.3.5.2 Other Major Assumptions

a. An NSSS power level of 3678 MWt

b. A reactor thermal power of 3659 MWt

c. A reactor coolant system Thermal Design Flow (TDF) of 93,600 gpm/loop

d. A reactor coolant system Minimum Measured Flow (MMF) of 95,950 gpm/loop

e. An average vessel average coolant temperature of between 571.0°F and 589.1°F

f. An average reactor coolant pressure of 2250 psia

g. An average steam generator tube plugging (SGTP) of 10%

For most accidents which are DNB-limited, nominal values of the initial conditions are assumed. The uncertainty allowances on power, temperature, pressure, and RCS flow are included on a statistical basis and are included in the limit DNBR value by using the Revised Thermal Design Procedure (RTDP)[19].

For accidents analyses which are not DNB limited, or for which RTDP is not employed, the initial conditions are obtained by applying the maximum steady-state errors to rated values. The following steady-state errors are considered in the analyses:

a. For reactor power, a 0%

b. For average RCS temperature, a $\pm$ 3.0°F random and –3.0°F bias*

c. For pressurizer pressure, a $\pm$ 50 psi

* A negative bias means that the indication is lower than actual.

Accidents employing RTDP assume a Minimum Measured Flow (MMF), while others assume the Thermal Design Flow (TDF). In addition to being the flow used in the DNB analysis for RTDP methodology, the MMF is bounded by the Tech Specs minimum flow measurement requirement. The MMF includes allowance for plant flow measurement uncertainty.

15.0.3.5.3 Overtemperature-ΔT and Overpower-ΔT

The overtemperature-ΔT and overpower-ΔT setpoints were recalculated for the power uprate program based on the most conservative core limits. The core limits used to calculate the OTΔT/OPΔT setpoints are provided in the COLR. All of the UFSAR events which rely on OTΔT and OPΔT for protection were analyzed to reflect the setpoint changes, as provided in the COLR. It has been confirmed that these OTΔT and OPΔT setpoints protect the core safety limits as shown in Figure 15.0-1.

15.0.3.5.4 RCCA Reactivity Characteristics

The negative reactivity insertion following a reactor trip is a function of the acceleration of the RCCAs and the variation in rod worth as a function of rod position. With respect to the accident analyses, the critical parameter is the time from beginning of RCCA insertion to dashpot entry, or approximately 85% of the RCCA travel. For the accident analyses, the insertion time from fully withdrawn to dashpot entry remains at the Tech Spec limit of 2.4 seconds from the beginning of stationary gripper coil voltage decay.

The normalized RCCA position (fraction insertion) versus the normalized time from release is presented in Figure 15.0-4. The reactivity worth versus rod insertion (fraction) assumed in the safety analyses is shown in Figure 15.0-5.

For analyses requiring the use of a dimensional diffusion theory code, the negative reactivity insertion resulting from the reactor trip is calculated directly by the reactor kinetic code and is not separable from other reactivity feedback effects. In this case, the RCCA position versus time of Figure 15.0-4 is used.

15.0.4 Reactivity Coefficients Assumed in the Accident Analyses

The transient response of the RCS is dependent on reactivity feedback effects, in particular the moderator density coefficient and the Doppler Power Coefficient (DPC). Depending upon event specific characteristics, conservatism dictates use of either large or small reactivity coefficient values. Justification for the use of the reactivity coefficient values is treated on an event-specific basis.

Maximum and minimum integrated DPCs assumed in the safety analyses are provided in Figure 15.0-2. The formulas for calculating the DPCs used are $[(.034Q^2) - 19.4Q] \times 10^{-5}$ for the maximum and $[(.0175Q^2) - 9.55Q] \times 10^{-5}$ for the minimum, where Q is the power level. Note that Steamline Break Core Response uses a different DPC based on a stuck RCCA.

In the analysis of certain events, conservatism requires the use of large reactivity coefficient values, whereas in the analysis of other events, conservatism requires the use of small reactivity coefficient values. The values used for each accident are given in Table 15.0-3. Conservative combinations of parameters are used for each event selected on a case-by-case basis.

15.0.5 Rod Cluster Control Assembly Insertion Characteristics

The negative reactivity insertion following a reactor trip is a function of the acceleration of the Rod Cluster Control Assemblies and the variation in rod worth as a function of rod position. Another critical parameter is the time of insertion up to the dashpot entry, or approximately 85 percent of the rod cluster travel. For accident analyses, the insertion time to dashpot entry is conservatively taken as 2.4 seconds. The Rod Cluster Control Assembly position versus time assumed in accident analyses is shown in Figure 15.0-4.

Figure 15.0-5 illustrates the fraction of total negative reactivity insertion versus normalized rod position for a core where the axial power distribution is skewed to the bottom. This curve is used to compute the negative reactivity insertion versus time following a reactor trip, for the majority of cases presented in Chapter 15.

There is inherent conservatism in the use of Figure 15.0-5, particularly for DNB related events which are typically limiting for top skewed power distributions. For DNB related events a curve based on a slightly bottom skewed shape was used.

The normalized Rod Cluster Control Assembly negative reactivity insertion versus time is shown in Figure 15.0-6. The curve shown in this figure was obtained from Figure 15.0-4 and Figure 15.0-5. Transient analyses performed with less conservative yet still bounding scram curves are specifically identified in subsequent sections. A total negative reactivity insertion following a trip of 4 percent ΔK is assumed in the transient analyses except where specifically noted otherwise. This assumption is conservative with respect to the calculated trip reactivity worth available. For Figure 15.0-4 and Figure 15.0-5, the rod cluster control assembly drop time is normalized to 2.4 seconds.

15.0.6 Trip Points and Time Delays to Trip Assumed in Accident Analyses

A reactor trip signal acts to open two trip breakers connected in series feeding power to the control rod drive mechanisms. Opening either trip breaker initiates a turbine trip. The loss of power to the mechanism coils causes the mechanisms to release the Rod Cluster control Assemblies, which then fall by gravity into the core. There are various 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 total delay to trip is defined as the time delay from the time that trip conditions are reached at the sensor to the time the rods are free and begin to fall. Limiting trip setpoints assumed in accident analyses and the total time delay assumed for each trip function are given in Table 15.0-4. The Overtemperature ΔT trip functions are illustrated in Figure 15.0-1.

The difference between the limiting trip point assumed for the analysis and the nominal trip point represents an allowance for instrumentation channel error and setpoint error. Nominal trip setpoints are specified in the plant Technical Specifications and Core Operating Limits Report.

In the analysis of the Chapter 15 events, control system action is considered only if that action results in more severe accident results. No credit is taken for control system operation if that operation mitigates the results of an accident.

15.0.7 Instrumentation Drift and Calorimetric Errors - Power Range Neutron Flux

Instrumentation drift and calorimetric errors are considered when establishing the power range high neutron flux setpoint.

The calorimetric error is the error assumed in the determination of core thermal power as obtained from secondary plant measurements. The total ion chamber current (sum of the top and bottom sections) is normalized to this measured power on a periodic basis.

15.0.8 Plant Systems and Components Available for Mitigation of Accident Effects

The Nuclear Steam Supply System (NSSS) is designed to afford proper protection against the possible effects of natural phenomena, postulated environmental conditions and dynamic effects of the postulated accidents. In addition, the design incorporates features which minimize the probability and effects of fires and explosions. Chapter 17 discusses the quality assurance program which has been implemented to assure that the NSSS will satisfactorily perform its assigned safety functions. The incorporation of these features in the NSSS, coupled with the reliability of the design, ensures that the normally operating systems and components listed in Table 15.0-5 will be available for mitigation of the events discussed in Chapter 15. In determining which systems are necessary to mitigate the effects of these postulated events, the classification system of ANSI N18.2-1973 is utilized. The design of "systems important to safety" (including protection systems) is consistent with IEEE Standard 379-1972 and Regulatory Guide 1.53 in the application of the single failure criterion.

In the analysis of the Chapter 15 events, control system action is considered only if that action results in more severe accident results. No credit is taken for control system operation if that operation mitigates the results of an accident. For some accidents, the analysis is performed both with and without control system operation to determine the worst case.

A functional analysis of the plant systems, in response to the various accidents, has been conducted. Results of the analysis, in the form of protection sequence diagrams, are presented in Figure 15.0-7, Figure 15.0-8, Figure 15.0-9, Figure 15.0-10, Figure 15.0-11, Figure 15.0-12, Figure 15.0-13, Figure 15.0-14, Figure 15.0-15, Figure 15.0-16, Figure 15.0-18, Figure 15.0-20, Figure 15.0-21, Figure 15.0-22, Figure 15.0-23, Figure 15.0-24, Figure 15.0-25, Figure 15.0-26, Figure 15.0-27, Figure 15.0-28, Figure 15.0-29, Figure 15.0-30, and Figure 15.0-31.

15.0.8.1 Effects of Operator Actions

For most of the events analyzed in Chapter 15, the plant will be in a safe and stable hot standby condition following the automatic actuation of reactor trip. This condition will in fact be similar to plant conditions following any normal, orderly shutdown of the reactor. At this point, the actions taken by the operator would be no different than normal operating procedures. The exact actions taken, and the time these actions would occur, will depend on what systems are available (e.g., Steam Dump System, Main Feedwater System, etc.) and the plans for further plant operation. As a minimum, to maintain the hot stabilized condition, decay heat must be removed via the steam generators. The Main Feedwater System and the Steam Dump or Atmospheric Relief System could be used for this purpose. Alternatively, the Emergency Feedwater System and the steam generator safety valves may be used, both of which are safety grade systems. Although the Emergency Feed System may be started manually, it will be automatically actuated if needed by one of the signals shown on Figure 7.2-1, Sheet 15, such as low-low steam generator water level. If hot standby conditions are maintained for an extended period of time, operator action may be required to transfer the emergency feedwater source. The time at which such action is required will be sufficiently long after initiation of the event to permit operator action. Also, if the hot standby condition is maintained for an extended period of time (greater than approximately 18 hours), operator action may be required to add boric acid via the CVCS to compensate for the xenon decay and maintain shutdown margin. Again, the actions taken by the operator would be no different than during normal plant shutdown.

For several events involving breaks in the Reactor Coolant System or secondary system piping, additional requirements for operator action can be identified. (Additional information about the impact of equipment failures or erroneous operator actions may be found in WCAP-9691, "NUREG-0578 2.1.9.C, Transient and Accident Analysis," Reference 14.

15.0.9 Fission Product Inventories

15.0.9.1 Activities in the Core

The Alternate Source Term (AST) for activities in the core is provided in Appendix 15C.

15.0.9.2 Activities in the Fuel Pellet Cladding Gap

The Alternate Source Term (AST) for activities in the Fuel Pellet Cladding Gap is provided in Table 15C-4.

15.0.9.3 Activities in the Secondary Side Coolant

The Alternate Source Term (AST) for activities in the Secondary Side Coolant is provided in Table 15C-3.

15.0.10 Residual Decay Heat

15.0.10.1 Total Residual Heat

Residual heat in a subcritical core is calculated for the loss-of-coolant accident per the requirements of Appendix K of 10 CFR 50.46 (Reference 4), as described in References 5 and 6. These requirements include assuming infinite irradiation time before the core goes subcritical to determine fission product decay energy. For all other accidents, the same models are used except that fission product decay energy is based on core average exposure at the end of the equilibrium cycle.

15.0.10.2 Distribution of Decay Heat Following Loss-of-Coolant Accident

During a loss-of-coolant accident the core is rapidly shut down by void formation or Rod Cluster Control Assembly insertion, or both, and a large fraction of the heat generation to be considered comes from fission product decay gamma rays. This heat is not distributed in the same manner as steady-state fission power. Local peaking effects which are important for the neutron dependent part of the heat generation do not apply to the gamma ray contribution. The steady-state factor of 97.4 percent which represents the fraction of heat generated within the clad and pellet drops to 95 percent for the hot rod in a loss-of-coolant accident.

For example, consider the transient resulting from the postulated double-ended break of the largest reactor coolant system pipe; 1/2 second after the rupture about 30 percent of the heat generated in the fuel rods is from gamma ray absorption. The gamma power shape is less peaked than the steady-state fission power shape, reducing the energy deposited in the hot rod at the expense of adjacent colder rods. A conservative estimate of this effect is a reduction of 10 percent of the gamma ray contribution or 3 percent of the total. Since the water density is considerably reduced at this time, an average of 98 percent of the available heat is deposited in the fuel rods, the remaining 2 percent being absorbed by water, thimbles, sleeves and grids. The net effect is a factor of 0.95 rather than 0.974, to be applied to the heat production in the hot rod.

15.0.11 Computer Codes Utilized

Summaries of some of the principal computer codes used in transient analyses are given below. Other codes, in particular very specialized codes in which the modeling has been developed to simulate one given accident, such as those used in the analysis of the reactor coolant system pipe rupture (Section 15.6), are summarized in their respective accident analyses sections. The codes used in the analyses of each transient have been listed in Table 15.0-3.

15.0.11.1 LOFTRAN

Transient response studies of a Pressurized Water Reactor (PWR) to specified perturbations in process parameters use the LOFTRAN[8] program. The LOFTRAN program models all four reactor coolant loops. This code simulates a multi-loop system by a model containing the reactor vessel, hot and cold leg piping, steam generators (tube and shell sides), the pressurizer and the pressurizer heaters, spray, relief valves, and safety valves. LOFTRAN also includes a point neutron kinetics model and reactivity effects of the moderator, fuel, boron, and rods. The secondary side of the steam generator uses a homogeneous, saturated mixture for the thermal transients and a water level correlation for indication and control.

The code simulates the Reactor Protection System (RPS) which includes reactor trips on high neutron flux, OTΔT, OPΔT, high and low pressurizer pressure, low flow, and high pressurizer level. Control systems are also simulated including rod control, steam dump, feedwater control, and pressurizer pressure control. The safety injection system (SIS), including the accumulators, is also modeled. LOFTRAN also has the capability of calculating transient values of DNBR based on the input from the core limits.

15.0.11.2 RETRAN-02

The RETRAN-02 program is used for studies of transient response of a PWR system to specified perturbations in process parameters. RETRAN-02 simulates a multi-loop system by a model containing reactor vessel, hot and cold leg piping, steam generator (tube and shell sides) and the pressurizer. The pressurizer heaters, spray, relief and safety valves are also considered in the program. Point model neutron kinetics and reactivity effects of the moderator, fuel, boron and rods are included. The secondary side of the steam generator utilizes a homogeneous, saturated mixture for the thermal transients and a water level correlation for indication and control. The Reactor Protection System is simulated to include reactor trips on high neutron flux, Overtemperature ΔT. Overpower ΔT, high and low pressure, low flow, and high pressurizer level. Control systems are also simulated including rod control, steam dump, feedwater control, and pressurizer pressure and level control. The Emergency Feedwater and Emergency Core Cooling System (except accumulators) are also modeled.

RETRAN-02 is a versatile program that is suited to both accident evaluation and control studies, as well as parameter sizing.

RETRAN-02 is further discussed in Reference 10.

15.0.11.3 SIMULATE-3 and CASMO-3

SIMULATE-3 is a two group, advanced nodal code, capable of determining detailed pin by pin power distributions for steady state and xenon transient conditions. All cross section data for SIMULATE-3 is given by CASMO-3 infinite lattice calculation. CASMO-3 uses neutron transport methods in forty neutron groups and collapses the results into two neutron group cross sections and discontinuity factors. Both codes have been extensively benchmarked and proven accurate in current safety analysis calculations performed by Yankee Atomic Electric Company and other utilities. Generic approval of both codes for this type of work was granted in YAEC-1363-A for CASMO-3 and YAEC-1659-A for SIMULATE-3.

Power distributions and local peaking factors are obtained from SIMULATE-3 calculations. Core conditions such as: control rod position, power level, and other parameters, are explicitly modeled within SIMULATE-3. The code uses the plant operating history, cross sections from CASMO-3, core conditions and control rod position to start the neutronic calculations. An industry standard advanced nodal technique is used to determine the incore flux and power distribution for each of nearly 20,000 nodes. Each node is defined as a quarter of an assembly in the radial direction and six inches in the axial direction. SIMULATE-3 has pin power reconstruction capabilities that will determine the power of each pin within each node.

SIMULATE-3 is further described in Reference 11.

15.0.11.4 VIPRE

The VIPRE computer program performs thermal-hydraulic calculations. This code calculates coolant density, mass velocity, enthalpy, void fractions, static pressure and DNBR distributions along flow channels within a reactor core. The VIPRE code is described in Reference 18, and Section 4.4.

15.0.11.5 FACTRAN

FACTRAN[16] calculates the transient temperature distribution in a cross-section of a metal clad UO_2 fuel rod and the transient heat flux at the surface of the clad, using as input the nuclear power and the time-dependent coolant parameters of pressure, flow, temperature and density. The code uses a fuel model that simultaneously contains the following features:

a. A sufficiently large number of radial space increments to handle fast transients such as a rod ejection accident;

b. Material properties which are functions of temperature and a sophisticated fuel-to-clad gap heat transfer calculation; and

c. The necessary calculations to handle post-DNB transients: film boiling heat transfer correlations, Zircaloy-water reaction, and partial melting of the fuel.

15.0.11.6 TWINKLE

The TWINKLE[17] program is a multi-dimensional spatial neutron kinetics code. The code uses an implicit finite-difference method to solve the two-group transient neutron diffusion equations in one, two, and three dimensions. The code uses six delayed neutron groups and contains a detailed multi-region fuel-clad-coolant heat transfer model for calculating pointwise Doppler and moderator feedback effects. The code handles up to 2,000 spatial points and performs steady-state initialization. Aside from basic cross-section data and thermal-hydraulic parameters, the code accepts as input basic driving functions such as inlet temperature, pressure, flow, boron concentration, control rod motion, and others. The code provides various output, e.g., channelwise power, axial offset, enthalpy, volumetric surge, pointwise power and fuel temperatures. It also predicts the kinetic behavior of a reactor for transients that cause a major perturbation in the spatial neutron flux distribution.

15.0.11.7 ANC

ANC[20] is an advanced nodal code capable of two-dimensional and three-dimensional neutronics calculations. ANC is the reference model for certain safety analysis calculations, power distributions, peaking factors, critical boron concentrations, control rod worths, reactivity coefficients, etc. In addition, three-dimensional ANC validates one-dimensional and two-dimensional results and provides information about radial (x-y) peaking factors as a function of axial position. It can calculate discrete pin powers from nodal information as well.

15.0.12 Radiological Consequences

Radiological consequences have been calculated for each hypothetical accident which can potentially result in radioactivity releases in excess of those expected to be experienced during normal plant operating conditions. In general, two hour TEDE doses are presented at the 914 meter site exclusion area boundary and duration of accident doses for the outer boundary of the low-population zone (2012 meters). Parameters and assumptions used to evaluate the radiological consequences are presented in the following discussions of each hypothetical accident, and are summarized in Appendix 15C.

The physical and mathematical models used in calculating radioactivity source terms are discussed in Section 11.1. Core fission products (halogens and noble gases) used to calculate accident doses are given in Appendix 15C.

The radioactive fission product source terms are determined for the fuel, fuel rod gap and reactor coolant for full power operation at 3654 MWt core thermal power as discussed in Appendix 15B.

The effect of V5H and RFA (w/IFMs) fuel upgrade implementation on each of the Seabrook Non-LOCA UFSAR transients were evaluated or analyzed. These transient evaluations and analyses demonstrate that all applicable safety analysis acceptance criteria continue to be met for the intended V5H and RFA (w/IFMs) fuel upgrade implementation at Seabrook Unit 1.

The hypothetical accident analyses show that the radiological consequences result in no offsite consequences, are bounded by radiological consequences calculated for other related accidents, or are below the guideline values of 10 CFR 100. Therefore, it is concluded that the Seabrook plant, Units 1 and 2, have been adequately designed to mitigate the potential radiological consequences of postulated accidents, and that they do not represent an undue hazard to public health and safety.

15.0.13 References

1. Deleted
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," RSIC-DLC-38, Radiation Shielding Information Center, Oak Ridge National Library, September 1975.
4. "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Cooled Nuclear Power Reactors," 10 CFR 50.46 and Appendix K of 10 CFR 50. Federal Register, Volume 39, Number 3, January 4, 1974.
5. Not used
6. Not used
7. Not used
8. Burnett, T. W. T., et al., "LOFTRAN Code Description," WCAP-7907-P-A, April 1984.
9. Not used
10. EPRI NP-1850, Volume 1, Rev. 4, "RETRAN-02 - A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems, Volume 1: Theory and Numerics (Rev. 4)", Electric Power Research Institute, November 1988.
11. YAEC-1659-A, "SIMULATE-3 Validation and Verification", A. S. Digiovine, September 1988.
12. Not used
13. Not used
14. Hithcler, M J., et al., "NUREG-0578 2.1.9.c Transient and Accident Analysis," WCAP-9691, March 1980.
15. Not used

16. WCAP-7908-A, "FACTRAN - A FORTRAN-IV Code for Thermal Transients in a UO_2 Fuel Rod," H. G. Hargrove, December 1989.

17. WCAP-7979-A, "TWINKLE - A Multi-Dimensional Neutron Kinetics Computer Code," D. H. Risher, and R. F. Barry, January 1975.

18. WCAP-14565, "VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis," Y. X. Sung, et al., April 1997.

19. WCAP-11397-P-A, "Revised Thermal Design Procedure," A. J. Friedland and S. Ray, April 1984.

20. WCAP-10965-P-A, "ANC: A Westinghouse Advanced Nodal Computer Code," Y.S. Liu, et al., September 1986.

15.1 INCREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM

A number of events have been postulated which could result in an increase in heat removal from the Reactor Coolant System by the secondary system. Detailed analyses are presented for several such events which have been identified as limiting cases.

Discussions of the following reactor coolant system cooldown events are presented in this section:

a. Feedwater system malfunction causing a reduction in feedwater temperature

b. Feedwater system malfunction causing an increase in feedwater flow

c. Excessive increase in secondary steam flow

d. Inadvertent opening of a steam generator relief or safety valve

e. Steam system piping failure.

The above are considered to be ANS Condition II events, with the exception of a major steam system pipe break, which is considered to be an ANS Condition IV event. Subsection 15.0.1 contains a discussion of ANS classification and applicable acceptance criteria.

15.1.1 Feedwater System Malfunctions Causing a Reduction in Feedwater Temperature

15.1.1.1 Identification of Causes and Accident Description

Reductions in feedwater temperature will cause an increase in core power by decreasing reactor coolant temperature. Such transients are attenuated by the thermal capacity of the secondary plant and of the Reactor Coolant System (RCS). The overpower - overtemperature protection (neutron overpower, Overtemperature and Overpower ΔT trips) prevents any power increase which could lead to a Departure from Nucleate Boiling Ratio (DNBR) less than the safety analysis limit value.

A reduction in feedwater temperature may be caused by the accidental opening of a feedwater bypass valve which diverts flow around a portion of the feedwater heaters. In the event of an accidental opening of a bypass valve, there is a sudden reduction in feedwater inlet temperature to the steam generators. At power, this increased subcooling will create a greater load demand on the RCS.

With the plant at no-load conditions, the addition of cold feedwater may cause a decrease in RCS temperature and thus a reactivity insertion due to the effects of the negative moderator temperature coefficient of reactivity. However, the rate of energy change is reduced as load and feedwater flow decrease, so the no-load transient is less severe than the full power case. The net effect on the RCS due to a reduction in feedwater temperature is similar to the effect of increasing secondary steam flow, i.e., the reactor will reach a new equilibrium condition at a power level corresponding to the new steam generator ΔT.

A decrease in normal feedwater temperature is classified as an ANS Condition II event, a fault of moderate frequency. See Subsection 15.0.1 for a discussion of Condition II events.

The protection available to mitigate the consequences of a decrease in feedwater temperature is the same as that for an excessive steam flow increase, as discussed in Subsection 15.0.8 and listed in Table 15.0-5.

15.1.1.2 Analysis of Effects and Consequences

a. Method of Analysis

This transient is analyzed by computing conditions at the feedwater pump inlet following opening of the heater bypass valve. These feedwater conditions are then used to recalculate a heat balance through the high pressure heaters. This heat balance gives the new feedwater conditions at the steam generator inlet.

The following assumptions are made:

1. Plant initial power level corresponding to guaranteed NSSS thermal output.
2. Low pressure heater bypass valve opens, resulting in condensate flow splitting between the bypass line and the low pressure heaters; the flow through each path is proportional to the pressure drops.
3. Heater drain pumps trip; this increases the effect of the cold bypass flow.

Plant characteristics and initial conditions are further discussed in Subsection 15.0.3.

b. Results

Opening of a low pressure heater bypass valve and trip of the heater drain pumps cause a reduction in feedwater temperature which increases the thermal load on the primary system. The calculated reduction in feedwater temperature is less than 35°F, resulting in an increase in heat load on the primary system of less than 10 percent of full power. The increased thermal load, due to opening of the low pressure heater bypass valve, thus would result in a transient very similar (but of reduced magnitude) to that presented in Subsection 15.1.5 for a steam system piping failure initiated at full power conditions. Therefore, the transient results of this analysis are not presented.

15.1.1.3 Radiological Consequences

No radioactivity releases are anticipated as a direct result of this malfunction. Consequently, no radiological consequences are predicted.

15.1.1.4 Conclusions

The decrease in feedwater temperature transient is less severe than the increase in feedwater flow event (Subsection 15.1.2), and the steam system piping failure initiated at full power conditions (Subsection 15.1.5). Based on results presented in Subsections 15.1.2 and 15.1.5, the applicable acceptance criteria for the decrease in feedwater temperature event have been met.

15.1.2 Feedwater System Malfunctions Causing an Increase in Feedwater Flow

15.1.2.1 Identification of Causes and Accident Description

Additions of excessive feedwater will cause an increase in core power by decreasing reactor coolant temperature. Such transients are attenuated by the thermal capacity of the secondary plant and of the RCS. The overpower-overtemperature protection (neutron overpower, Overtemperature and Overpower ΔT trips) prevent any power increase which could lead to a DNBR less than the safety analysis limit value.

An example of excessive feedwater flow would be a full opening of a feedwater control valve due to a feedwater control system malfunction or an operator error. At power this excess flow causes a greater load demand on the RCS due to increased subcooling in the steam generator. With the plant at no-load conditions, the addition of an excess of feedwater may cause a decrease in RCS temperature and thus a reactivity insertion due to the effects of the negative moderator coefficient of reactivity.

Continuous addition of excessive feedwater is prevented by the steam generator high-high level trip, which activates the feedwater isolation. Pre-trip alarm of high steam generator level is available in the control room.

An increase in normal feedwater flow is classified as an ANS Condition II event, a fault of moderate frequency. See Subsection 15.0.1 for a discussion of ANS Condition II events.

Plant systems and equipment which are available to mitigate the effects of the accident are discussed in Subsection 15.0.8 and listed in Table 15.0-5.

15.1.2.2 Analysis of Effects and Consequences

a. Method of Analysis

The excessive heat removal due to a feedwater system malfunction transient is analyzed by using the detailed digital computer RETRAN[1] code. This code simulates the neutron kinetics of the reactor coolant system, pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generator, and steam generator safety valves. The code computes pertinent plant variables including temperatures, pressures, and power level.

The system is analyzed to demonstrate acceptable consequences in the event of an excessive feedwater addition, due to a control system malfunction or operator error which allows a feedwater control valve to open fully. Three cases are analyzed as follows:

1) Accidental opening of one feedwater control valve with the reactor in automatic control at full power.

2) Accidental opening of one feedwater control valve with the reactor in manual control at full power.

3) Accidental opening of one feedwater control valve with the reactor at zero load, with the reactor just critical.

This accident is analyzed with the revised thermal design procedure as described in WCAP-11397[5]. The reactivity insertion rate following a feedwater system malfunction is calculated with the following assumptions:

a. For the feedwater control valve accident at full power, one feedwater control valve is assumed to malfunction resulting in a step increase to 187 percent of nominal feedwater flow to one steam generator.

b. For the feedwater control valve accident at zero load condition, a feedwater control valve malfunction occurs which results in an increase in flow to one steam generator from zero to 200 percent of the nominal full load value for one steam generator.

c. For the zero load condition, feedwater temperature is at a conservatively low value of 100°F.

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

e. The feedwater flow resulting from a fully-open control valve is terminated by a steam generator high-high level trip signal which closes all feedwater control and isolation valves, trips the main feedwater pumps, and trips the turbine.

Normal reactor control systems and engineered safety 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.

b. Results

The calculated sequence of events for this accident is shown in Table 15.1-1.

The full power cases with maximum reactivity feedback coefficients give the largest reactivity feedback and result in the highest peak power. The manual and automatic rod control cases give similar results (although the manual control case has a slightly higher peak power and a slightly lower minimum DNBR value). The rod control system is not required to function for an excessive feedwater flow event.

When the steam generator water level in the faulted loop reaches the high-high level setpoint, all feedwater control valves and feedwater isolation valves are automatically closed and the main feedwater pumps are tripped. In addition, a turbine trip is initiated.

Transient results for the full power and zero power cases are provided in Figure 15.1-1. The DNBR does not fall below the limit value. Following reactor trip (full power cases), the plant approaches a stabilized condition; standard plant shutdown procedures may then be followed to further cool down the plant.

Since the power level rises during the excessive feedwater flow incident, the fuel temperatures 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 value does not exceed 118 percent of its nominal value (i.e., the assumed high neutron flux trip setpoint). The peak fuel temperature will thus remain well below the fuel melting temperature.

The transient results have shown that the DNBR does not go below the limit value at any time during the excessive feedwater flow incident; thus, the ability of the primary coolant to remove heat from the fuel rod is not reduced.

15.1.2.3 Radiological Consequences

No fuel failure and radioactivity releases are anticipated as a direct result of this malfunction. Consequently, no radiological consequences are predicted.

15.1.2.4 Conclusions.

The results of the analysis show that the DNB ratio encountered for an excessive feedwater addition at power is above the limit value; hence, no fuel or clad damage is predicted.

15.1.3 Excessive Increase in Secondary Steam Flow

15.1.3.1 Identification of Causes and Accident Description

An excessive increase in secondary system steam flow (excessive load increase incident) is defined as a rapid increase in steam flow that causes a power mismatch between the reactor core power and the steam generator load demand. The Reactor Control System is designed to accommodate a 10 percent step load increase or a 5 percent per minute ramp load increase in the range of 15 percent to 100 percent of full power. Any loading rate in excess of these values may cause a reactor trip actuated by the Reactor Protection System.

Steam flow increases greater than 10 percent are analyzed in Subsections 15.1.4 and 15.1.5.

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.

During power operation, steam dump to the condenser is controlled by reactor coolant condition signals, i.e., high reactor coolant temperature indicates a need for steam dump. A single controller malfunction does not cause steam dump; an interlock is provided which blocks the opening of the valves unless a large turbine load decrease or a turbine trip has occurred.

Protection against an excessive load increase accident is provided by the following Reactor Protection System (RPS) signals:

- Overpower ΔT
- Overtemperature ΔT
- Power range high neutron flux
- Low pressurizer pressure

An excessive load increase incident is considered to be an ANS Condition II event, a fault of moderate frequency. See Subsection 15.0.1 for a discussion of Condition II events.

15.1.3.2 Analysis of Effects and Consequences

a. Method of Analysis

Historically, four cases are analyzed to demonstrate the plant behavior following a 10 percent step load increase from rated load. These cases are as follows:

1. Reactor control in manual with minimum reactivity feedback;
2. Reactor control in manual with maximum reactivity feedback;
3. Reactor control in automatic with minimum reactivity feedback; and
4. Reactor control in automatic with maximum reactivity feedback.

A conservative limit on the turbine valve opening was assumed, and all cases were analyzed without credit being taken for pressurizer heaters.

This accident is analyzed with the revised thermal design procedure as described in WCAP-11397[5]. Initial reactor power, RCS pressure and temperature are assumed to be at their nominal values.

Normal reactor control systems and engineered safety systems were not required to function for this event. The reactor protection system was assumed to be operable; however, reactor trip was not encountered for most cases due to the error allowances assumed in the setpoints. No single active failure would prevent the reactor protection system from performing its intended function.

The cases which assume automatic rod control were analyzed to ensure that the worst case with respect to minimum DNBR is presented. The automatic rod control function is not required for core protection.

Given the non-limiting nature of this event with respect to the DNBR safety analysis criterion, an explicit analysis was not performed as part of the Power Uprate Program. Instead, a detailed evaluation of this event was performed. The evaluation model consists of the generation of statepoints based on generic conservative data. The statepoints are in the form of changes to the initial conditions and then applied to the actual operating conditions of the plant. The statepoints are then compared to the core thermal limits to ensure that the DNBR limit is not violated. Four cases are evaluated for both manual rod control and automatic rod control. These cases are:

- Reactor in manual rod control with BOL (minimum moderator) reactivity feedback.
- Reactor in manual rod control with EOL (maximum moderator) reactivity feedback.
- Reactor in automatoc rod control with BOL (minimum moderator) reactivity feedback.
- Reactor in automatic rod control with EOL (minimum moderator) reactivity feedback.

15.1.3.3 Results and Conclusions

An evaluation of this event was performed to support the Power Uprate Program. The evaluation determined that the DNB design basis for a 10% step load increase continues to be met.

15.1.4 Inadvertent Opening of a Steam Generator Relief or Safety Valve

15.1.4.1 Identification of Causes and Accident Description

The most severe core conditions resulting from an accidental depressurization of the Main Steam System are associated with an inadvertent opening of a single steam dump, relief, or safety valve. The analyses performed assuming a rupture of a main steam line are given in Subsection 15.1.5.

The steam release as a consequence of this accident results in an initial increase in steam flow which decreases during the accident as the steam pressure falls. The energy removal from the RCS causes a reduction in coolant temperature and pressure. In the presence of a negative moderator temperature coefficient, the cooldown results in an insertion of positive reactivity and a reduction of core shutdown margin.

The analysis is performed to demonstrate that the following criterion is satisfied:

> Assuming a stuck Rod Cluster Control Assembly, with offsite power available and assuming a single failure in the Engineered Safety Features System, there will be no consequential damage to the core or Reactor Coolant System after reactor trip for a steam release equivalent to the spurious opening, with failure to close of the largest of any single steam dump, relief, or safety valve.

Accidental depressurization of the secondary system is classified as an ANS Condition II event. See Subsection 15.0.1 for a discussion of Condition II events.

The following systems provide the necessary protection against an accidental depressurization of the Main Steam System.

a. Safety injection system actuation from any of the following:

    1. Two out of four 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 one loop.

b. The overpower reactor trips (neutron flux and ΔT) and the reactor trip occurring in conjunction with receipt of the safety injection signal.

c. Redundant isolation of the main feedwater lines.

    Sustained high feedwater flow would cause additional cooldown. Therefore, in addition to the normal control action which will close the main feedwater valves following reactor trip, a safety injection signal will rapidly close all feedwater control valves and backup feedwater isolation valves and trip the main feedwater pumps.

d. Trip of the fast-acting steam line stop valves (designed to close in less than 5 seconds) on:

    1. High-2 containment pressure

    2. Safety injection system actuation derived from two out of three low steam line pressure signals in any one loop (above Permissive P-11)

    3. Two out of three high negative steam line pressure rate in any one loop (below Permissive P-11).

Plant systems and equipment which are available to mitigate the effects of the accident are also discussed in Subsection 15.0.8 and listed in Table 15.0-5.

15.1.4.2 Analysis of Effects and Consequences

a. Method of Analysis

The consequences of an inadvertent opening of a steam generator relief or safety valve are bounded by the zero power steam line rupture discussed in Section 15.1.5. The opening of a steam generator relief or safety valve causes a slower steam generator blowdown and RCS cooldown than the steam line rupture event. This would result in a lower power level if a return to power were to occur as predicted for the zero power steam line rupture. The minimum DNBR for the zero power steam line rupture, which remains above the safety analysis limit, would be lower than that for the opening of a steam generator relief or safety valve.

b. Results

Since the minimum DNBR for the zero power steam line rupture (Subsection 15.1.5) remains above the safety analysis limit, there would be no fuel failure predicted for an inadvertent opening of a steam generator relief or safety valve.

15.1.4.3 Radiological Consequences

The radiological consequences resulting from this malfunction are considerably less than those calculated for a main steam line rupture. The analyses performed assuming a rupture of a main steam line are given in Subsection 15.1.5.3.

15.1.4.4 Conclusions

The analysis shows that the criteria stated earlier in this section are satisfied. The DNBR in maintained above the safety analysis limit value.

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 would result in an initial increase in steam flow which decreases during the accident as the steam pressure falls. The energy removal from the 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 requires evaluation 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 delivered by the Safety Injection System.

The limiting steam line break presented in this section corresponds to a double-ended rupture of the main steam line at the steam generator nozzle at zero power with offsite power available.

The analysis of a main steam line rupture is performed to demonstrate that the following criteria are satisfied:

a. Assuming a stuck RCCA with or without offsite power, and assuming a single failure in the Engineered Safety Features, the core remains in place and intact. Radiation doses do not exceed the guidelines of 10 CFR 100.

b. Although DNB and possible clad perforation following a steam pipe rupture are not necessarily unacceptable, the following analysis, in fact, shows that no DNB occurs for any rupture assuming the most reactive assembly stuck in its fully withdrawn position.

A major steam line rupture is classified as an ANS Condition IV event. A minor steam line rupture is classified as an ANS Condition III 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.

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 return to power. A detailed analysis of this transient with the most limiting break size, a double-ended rupture, is presented here.

The following functions provide the protection for a steam line rupture:

a. Safety injection 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 one loop.

b. The overpower reactor trips (neutron flux and ΔT) and the reactor trip occurring in conjunction with receipt of the safety injection 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, a safety injection signal will rapidly close all feedwater isolation valves and backup feedwater control valves and trip the main feedwater pumps.

d. Trip of the fast-acting Main Steam Isolation Valves (MSIVs) which are designed to close in less than 5 seconds after receipt of a signal on:

   1. High-2 containment pressure

2. Safety injection system actuation derived from two out of three low steam line pressure signals in any one loop (above Permissive P-11)

3. Two out of three high negative steam pressure rate in any one loop (below Permissive P-11).

For breaks downstream of the isolation valves, closure of all valves would 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 isolation valves fails to close.

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 square feet, 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. Also, the main steam isolation valve seat area limits the reverse blowdown from the intact steam generators.

15.1.5.2 Analysis of Effects and Consequences

a. 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 transients resulting from the cooldown following the steam line break. The RETRAN[1] code has been used.

b. The thermal and hydraulic behavior of the core following a steam line break. A detailed thermal and hydraulic digital computer code, VIPRE[4], has been used to determine if DNB occurs for the core conditions computed in item a above.

Studies have been performed to determine the sensitivity of steam line break results to various assumptions (Reference 6). Based upon this study, the following conditions were assumed to exist at the time of a main steam line break accident:

1. 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 banks during core burnup is restricted in such a way that the addition of positive reactivity in a steam line break accident will not lead to a more adverse condition than the case analyzed.

2. A negative moderator coefficient corresponding to the end-of-life rodded core with the most reactive RCCA in the fully withdrawn position. The effect of power generation in the core on overall reactivity is shown in Figure 15.1-6.

The core properties associated with the sector nearest the affected steam generator and those associated with the remaining sector were conservatively combined to obtain average core properties for reactivity feedback calculation. 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.

To verify the conservatism of this method, the reactivity, as well as the power distribution, was checked for the limiting conditions for 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 non-uniform 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 statepoints. These results verify conservatism; i.e., underprediction of negative reactivity feedback from power generation.

3. Minimum safety injection flow capability corresponding to the most restrictive single failure in the safety injection system. The Emergency Core Cooling System (ECCS) consists of three systems: a) the passive accumulators, b) the low head safety injection (residual heat removal) system, and c) the high head safety injection (charging) system. Only the safety injection system and the passive accumulators are modeled for the steam line break accident analysis.

   The modeling of the safety injection system in RETRAN is described in Reference 1. 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 low concentration borated water, which must be swept from the lines downstream prior to the delivery of high concentration boric acid to the reactor coolant loops.

When offsite power is assumed, the sequence of events in the safety injection system is the following: After the generation of the safety injection signal (appropriate delays for instrumentation, logic, and signal transport included), the appropriate valves begin to operate and the high head safety injection pump starts. In 27 seconds, the valves are assumed to be in their final position and the pump is assumed to be at full speed. The volume containing the low concentration borated water is swept into core before the 2,400 ppm borated water from the refueling water storage tank reaches the core. This delay, described above, is inherently included in the modeling.

4. Design value of the steam generator heat transfer coefficient, with no allowance for fouling factor, to maximize the cooldown.

5. Since the steam generators are provided with integral flow restrictors with a 1.4 square foot throat area, any rupture with a break area greater than 1.4 ft^2, regardless of location, would have the same effect on the NSSS as the 1.4 ft^2 break.

6. Power peaking factors corresponding to one stuck RCCA and non-uniform 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 thus 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 the cases above assume initial hot shutdown conditions at time zero since this represents the most pessimistic initial condition. Should the reactor be 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 point. Following a trip at power, the reactor coolant system 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.

In addition, since the initial steam generator water inventory is greatest at no-load, the magnitude and duration of RCS cooldown are more severe than steam line breaks occurring at power.

7. In computing the steam flow during a steam line break, the Moody curve[15] for fL/D = 0 is used.

8. Emergency Feedwater flow is limited by passive flow restrictors to protect the pumps against a runout condition during main steam line rupture.

The following cases have been considered in determining the core power and RCS transients:

a. Complete severance of a pipe, with the plant initially at no-load conditions, full reactor coolant flow with offsite power available.

b. Case (a) with loss of offsite power simultaneous with the steam line break and initiation of the SIS. Loss of offsite power results in reactor coolant pump coastdown.

The limiting steam line break with return to power corresponds to the case with offsite power available. The offsite power available case results in the greatest challenge to DNB.

b. Results

The calculated sequence of events is shown on Table 15.1-1. 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.

1. Core Power and Reactor Coolant System Transient

Figure 15.1-6, sheets 1 through 6, 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 a).

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 safety injection 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 isolation valves in the steam lines, by high containment pressure signals, or low steam line pressure. Even with the failure of one isolation valve, release is limited to no more than 10 seconds for the other steam generators while the one generator blows down. The steam line isolation valves are designed to be fully closed in less than 5 seconds from receipt of a closure signal.

As shown in sheets 1 through 6 of Figure 15.1-6, the core attains criticality with the RCCAs inserted (with the design shutdown assuming one stuck RCCA) before boron solution at 2,400 ppm enters the RCS. A peak core power less 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 flow rates in the RCS and in the safety injection system. The variation of mass flow rate in the RCS due to water density changes is included in the calculation as is the variation of flow rate in the safety injection system due to changes in the RCS pressure. The safety injection system flow calculation includes the line losses in the system as well as the pump head curve.

Once the pressure in the RCS falls below the pressure in the accumulators, boron solution at 2,300 ppm also enters the RCS from the accumulators.

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.

2. Margin to Critical Heat Flux

   A DNB analysis was performed. It was found that all cases had a minimum DNBR greater than the limit value.

15.1.5.3 Radiological Consequences Using Alternate Source Term Methodology

a. Background

This event consists of a double-ended break of one main steam line outside of containment. The radiological consequences of such an accident bound those of a MSLB inside containment. The affected steam generator (SG) rapidly depressurizes and releases the initial contents of the SG to the environment. Plant cool down is achieved via the remaining unaffected SGs.

b. Compliance with RG 1.183 Regulatory Positions

The MSLB dose consequence analysis is consistent with the guidance provided in RG 1.183, Appendix E "Assumptions for Evaluating the Radiological Consequences of a PWR Main Steam Line Break Accident," as discussed below:

1. Regulatory Position 1 – No fuel damage is postulated to occur for the Seabrook MSLB event.

2. Regulatory Position 2 – No fuel damage is postulated to occur for the Seabrook MSLB event. Two cases of iodine spiking are evaluated.

3. Regulatory Position 2.1 – One iodine spiking case assumes a reactor transient prior to the postulated MSLB that raises the primary coolant iodine concentration to the maximum allowed by Tech Specs, which is a value of 60.0 μCi/gm DE I-131 for the analyzed conditions. This is the pre-accident spike case.

4. Regulatory Position 2.2 – One case assumes the transient associated with the MSLB causes an iodine spike. The spiking model assumes the primary coolant activity is initially at the Tech Spec value of 1.0 μCi/gm DE I-131. Iodine is assumed to be released from the fuel into the RCS at a rate of 500 times the iodine equilibrium release rate for a period of 8 hours. This is the accident-induced spike case.

5. Regulatory Position 3 – The activity released from the fuel is assumed to be released instantaneously and homogeneously through the primary coolant.

6. Regulatory Position 4 – Iodine releases from the steam generators to the environment are assumed to be 97% elemental and 3% organic.

7. Regulatory Position 5.1 – The primary-to-secondary accident induced leakage rate is apportioned between the SGs as specified by the Technical Specification Steam Generator Program (1.0 gpm total, 500 gpd to any one SG). The tube leakage is conservatively apportioned as 500 gpd to the faulted SG and 940 gpd total to the other three SGs in order to maximize dose consequences.

8. Regulatory Position 5.2 – The density used in converting volumetric leak rates to mass leak rates is consistent with the basis of surveillance tests used to show compliance with the SG leak rate TS. For the intact Steam Generators, the primary to secondary leak rate is based on a density of 1.0 gm/cc (cold liquid).

9. Regulatory Position 5.3 – The primary-to-secondary leak rate is assumed to continue until the temperature of the leakage is less than 212°F at 48 hours. The release of radioactivity from the unaffected SGs is assumed to continue until the steam release is terminated due to RHR initiation at 8 hours.

10. Regulatory Position 5.4 – All noble gas radionuclides released from the primary system are assumed to be released to the environment without reduction or mitigation.

11. Regulatory Position 5.5.1 – In the faulted SG, all of the primary-to-secondary leakage is assumed to flash to vapor and be released to the environment with no mitigation. For the unaffected steam generators used for plant cooldown, tube bundle uncovery is not postulated; therefore, the primary-to-secondary leakage is assumed to mix with the secondary water without flashing.

12. Regulatory Position 5.5.2 – Tube bundle uncovery is not postulated for the unaffected SGs; therefore, this section does not apply. In the faulted SG, all of the fluid is assumed to flash and be released without mitigation.

13. Regulatory Position 5.5.3 – All leakage that does not immediately flash is assumed to mix with the bulk water.

14. Regulatory Position 5.5.4 – The radioactivity within the bulk water is assumed to become vapor at a rate that is a function of the steaming rate and the partition coefficient. A partition coefficient of 100 is assumed for the iodine. The retention of particulate radionuclides in the unaffected SGs is limited by the moisture carryover from the SGs. The same partition coefficient of 100, as used for iodine, is assumed for other particulate radionuclides. This assumption is consistent with the SG carryover rate of less than 1%. No reduction in the release is assumed from the faulted SG.

15. Regulatory Position 5.6 – Steam generator tube bundle uncovery is not postulated for the intact SGs for Seabrook.

c. <u>Other Assumptions</u>

1. RG 1.183 does not address secondary coolant activity. This analysis assumes that the equilibrium specific activity on the secondary side of the steam generators is equal to the Tech Spec limit of 0.1 μCi/gm Dose Equivalent I-131.

2. The steam mass release rates for the intact SGs are provided in Table 15.1-3.

3. This evaluation assumes that the RCS mass remains constant throughout the MSLB event (no change in the RCS mass is assumed as a result of the MSLB or from the safety injection system).

4. The SG secondary side mass in the unaffected SGs is assumed to remain constant throughout the event.

5. Releases from the faulted main steam line (and associated SG) are postulated to occur from the main steam line associated with the most limiting atmospheric dispersion factors. Releases from the unaffected SGs are postulated to occur from the MSSV or ASDV with the most limiting atmospheric dispersion factors.

d. <u>Methodology</u>

Input assumptions used in the dose consequence analysis of the MSLB are provided in Table 15.1-2. The postulated accident assumes a double-ended break of one main steam line outside containment. The radiological consequences of such an accident bound those of a MSLB inside of containment. Upon a MSLB, the affected SG rapidly depressurizes and releases the initial contents of the SG to the environment. Plant cooldown is achieved via the remaining unaffected SGs.

The analysis assumes that the entire fluid inventory from the affected SG is immediately released to the environment. The secondary coolant iodine concentration is assumed to be the maximum value of 0.1 μCi/gm DE I-131 permitted by Tech Specs. Primary coolant is also released into the affected steam generator by leakage across the SG tubes based on the Technical Specification Steam Generator Program primary to secondary accident induced leakage limits. Activity is released to the environment from the affected steam generator, as a result of the postulated primary-to-secondary leakage and the postulated activity levels of the primary and secondary coolants, until the affected steam generator is completely isolated at 48 hours (primary system temperature less than 212°F). Additional activity, based on the Tech Spec primary-to-secondary leakage limits (SG tube leakage), is released via the unaffected SGs via steaming from the unaffected SGs MSSVs/ASDVs for 8 hours (time of RHR initiation). These release assumptions are consistent with the requirements of RG 1.183.

Fuel damage is not postulated for the MSLB event. Consistent with Regulatory Guide 1.183, Appendix E, Regulatory Position 2, if no or minimal fuel damage is postulated for the limiting event, the activity released is assumed as the maximum allowed by Technical Specifications for two cases of iodine spiking: (1) maximum pre-accident iodine spike; and (2) maximum accident-induced or concurrent, iodine spike.

For the case of a pre-accident iodine spike, a reactor transient is assumed to have occurred prior to the postulated MSLB event. The primary coolant iodine concentration is increased to the maximum value of 60 μCi/gm DE I-131 permitted by Technical Specification 3.4.8. The iodine activities for the pre-accident spike case are presented in Table 15.1-4.

For the case of the accident-induced spike, the postulated MSLB event induces an iodine spike. The RCS activity is initially assumed to be 1.0 μCi/gm DE I-131 as allowed by Tech Specs. Iodine is released from the fuel into the RCS at a rate of 500 times the iodine equilibrium release rate for a period of 8 hours. The iodine activities for the accident-induced (concurrent) iodine spike case are presented in Table 15.1-6.

For this event, the Control Room ventilation system cycles through two modes of operation:

- Initially, the ventilation system is assumed to be operating in normal mode. The air intake to the Control Room during this mode is 1000 cfm of unfiltered fresh air.

- After the start of the event, the Control Room normal air intake is isolated on a CR intake radiation monitor signal. A 30-second delay is conservatively applied to account for the time to reach the signal, the diesel generator start time, load sequencing and damper actuation and positioning time. After isolation of the Control Room normal air intake, the air flow distribution consists of 600 cfm of filtered makeup flow through the worst of the two emergency intakes, 150 cfm of unfiltered inleakage and 390 cfm of filtered recirculation flow.
- 20 cfm of unfiltered inleakage was assumed to enter the Control Room via the CR fire exit and 130 cfm was assumed to enter via the Diesel Building.
- The Control Room ventilation filter efficiencies that are applied to the filtered makeup and recirculation flows are 99% for particulates/aerosols, and 95% for elemental and organic iodine.

e. Radiological Consequences

The atmospheric dispersion factors (*X/Qs*) used for this event for the Control Room does are based on the postulated release locations and the operational mode of the control room ventilation system. These *X/Qs* are summarized in Table 2R-2 and Table 2R-3.

Releases from the intact SGs are assumed to occur from the MSSV/ASDV that produces the most limiting *X/Qs*. Releases from the faulted SG are assumed to occur from the location on a steam line that produces the most limiting *X/Qs*.

For the EAB and LPZ dose analysis, the *X/Q* factors for the appropriate time intervals are used. These *X/Q* factors are provided in Appendix 2Q.

The radiological consequences of the MSLB Accident are analyzed using the RADTRAD-NAI code and the inputs/assumptions previously discussed. Cases for MSLB pre-accident and concurrent iodine spikes are analyzed. As shown in Table 15.1-7, the results of both cases for EAB dose, LPZ dose, and Control Room dose are within the appropriate regulatory acceptance criteria.

15.1.6 References

1. WCAP-14882-P-A, "RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analyses," D.S. Huegel, et al., April 1999
2. WCAP-7908-A, "FACTRAN - A FORTRAN-IV Code for Thermal Transients in a UO_2 Fuel Rod," H. G. Hargrove, December 1989

3. WCAP-7979-A, "TWINKLE - A Multi-Dimensional Neutron Kinetics Computer Code," D. H. Risher, and R. F. Barry, January 1975

4. WCAP-14565, "Vipre-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis," Y. X. Sung, et al., April 1997

5. WCAP-11397-P-A, "Revised Thermal Design Procedure," A. J. Friedland and S. Ray, April 1984

6. WCAP-9226-P-A. "Reactor Core Response to Excessive Secondary Steam Releases," S. D. Hollingsworth, et al., January 1998

7. WCAP-9230, "Report on the Consequences of a Postulated Main Feedline Rupture," G. E. Lang and J. P. Cunningham, January 1978

8. WCAP-11394-P-A, "Methodology for the Analysis of the Dropped Rod Event," R. L. Haessler, et al., January 1990

9. WCAP-8424, Revision 1, An Evaluation of Loss of Flow Accidents Caused by Power System Frequency Transients in Westinghouse PWRs," M. S. Baldwin, et al., June 1975

10. WCAP-7588, Revision 1-A, "An Evaluation of the Rod Ejection Accident in Westinghouse Pressurized Water Reactors Using Spatial Kinetics Methods," D. H. Risher, January 1975

11. IN-1370, "Annual Report - SPERT Project, October 1968 - September 1969 Edition (Idaho Nuclear Corporation)," T. G. Taxelius, June 1970

12. ANL-7225, "Studies in TREAT of Zircaloy-2-Clad, UO_2-Core Simulated Fuel Elements," R. C. Liimantaininen and F. J. Testa, p. 177, November 1966

13. Letter from W. J. Johnson (Westinghouse) to R. C. Jones (USNRC), "Use of 2700°F PCT Acceptance Limit in Non-LOCA Accidents," NS-NRC-893466, October 1989

14. ANSI/ANS-5.1-1979, "American National Standard for Decay Heat Power in Light Water Reactors," August 1979

15. Journal of Heat Transfer, "Maximum Flow Rate of a Single Component, Two-Phase Mixture," F.J. Moody, February 1965

15.2 DECREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM

A number of transients and accidents have been postulated in this section which could result in a reduction of the capacity of the secondary system to remove heat generated in the Reactor Coolant System (RCS). Detailed analyses are presented for the most limiting of these events.

Discussions of the following RCS coolant heatup events are presented in this section:

a. Steam pressure regulator malfunction

b. Loss of external load

c. Turbine trip

d. Inadvertent closure of main steam isolation valves

e. Loss of condenser vacuum and other events resulting in turbine trip

f. Loss of nonemergency AC power to the station auxiliaries

g. Loss of normal feedwater flow

h. Feedwater system pipe break.

The above items are considered to be American Nuclear Society (ANS) Condition II events, with the exception of a feedwater system pipe break, which is considered to be an ANS Condition IV event. Subsection 15.0.1 contains a discussion of ANS classifications and applicable acceptance criteria.

15.2.1 Steam Pressure Regulator Malfunction Or Failure that Results In Decreasing Steam Flow

There are no steam pressure regulators in the Seabrook plant whose failure or malfunction could cause a steam flow transient.

15.2.2 Loss of External Load

15.2.2.1 Identification of Causes and Accident Description

A major plant load loss can result from the loss of external electrical load due to some electrical system disturbance. Offsite alternating current power remains available to operate plant components such as the reactor coolant pumps; as a result, the onsite emergency diesel generators are not required to function for this event. Following the loss of generator load, an immediate fast closure of the turbine control valves will occur. This will cause a sudden reduction in steam flow, resulting in an increase in pressure and temperature in the steam generator shell. As a result, the heat transfer rate in the steam generator is reduced, causing the reactor coolant temperature to rise, which in turn causes coolant expansion, pressurizer insurge, and RCS pressure rise.

For a loss of external electrical load without subsequent turbine trip, no direct reactor trip signal would be generated. The plant would be expected to trip from the Reactor Protection System if a safety limit were approached. A continued steam load of approximately 5 percent would exist after total loss of external electrical load because of the steam demand of plant auxiliaries.

In the event that a safety limit is approached, protection would be provided by the high pressurizer pressure, Overtemperature ΔT and steam generator low-low water level trips. Voltage and frequency relays associated with the reactor coolant pump provide no additional safety function for this event. Following a complete loss of load, the maximum turbine overspeed would be approximately 8 to 9 percent, resulting in an overfrequency of less than 6 hertz (hz). This resulting overfrequency is not expected to damage the sensors (Nonnuclear Steam Supply System) in any way. However, it is noted that frequent testing of this equipment is required by the Technical Specifications. Any degradation in their performance could be ascertained at that time. Any increased frequency to the reactor coolant pump motors will result in slightly increased flow rate and subsequent additional margin to safety limits. Safeguards loads are supplied from offsite power or alternatively, from emergency diesels. Reactor protection system equipment is supplied from the 118 volt AC instrument Power Supply System, which in turn is supplied from the inverters; the inverters are supplied from a direct current bus energized from batteries or by a rectified AC voltage from safeguards buses.

In the event the steam dump valves fail to open following a large loss of load, the steam generator safety valves may lift and the reactor may be tripped by the high pressurizer pressure signal, the steam generator low-low water level signal, or the Overtemperature ΔT signal. The steam generator shell side pressure and reactor coolant temperatures will increase rapidly. The pressurizer safety valves and steam generator safety valves are, however, sized to protect the RCS and steam generator against overpressure for all load losses without assuming the operation of the Steam Dump System, pressurizer spray, pressurizer power-operated relief valves, automatic rod cluster control or direct reactor trip on turbine trip.

The steam generator safety valve capacity is capable of removing the steam flow at 100 percent of the analyzed core power from the steam generator without exceeding 110 percent of the steam system design pressure. The pressurizer safety valve capacity is sized based on a complete loss of heat sink with the plant initially operating at the maximum calculated turbine load along with operation of the steam generator safety valves. The pressurizer safety valves are then able to relieve sufficient steam to maintain the RCS pressure within 110 percent of the RCS design pressure.

A more complete discussion of overpressure protection can be found in Reference 1.

A loss of external load is classified as an ANS Condition II event, a fault of moderate frequency. See Subsection 15.0.1 for a discussion of Condition II events.

A loss of external load event results in a nuclear steam supply system transient that is less severe than a turbine trip event (see Subsection 15.2.3). Therefore, a detailed transient analysis is not presented for the loss of external load.

The primary side transient is caused by a decrease in heat transfer capability from primary to secondary due to a rapid termination of steam flow to the turbine, accompanied by an automatic reduction of feedwater flow (should feed flow not be reduced, a larger heat sink would be available and the transient would be less severe). Termination of steam flow to the turbine following a loss of external load occurs due to automatic fast closure of the turbine control valves in approximately 0.3 seconds. Following a turbine trip event, termination of steam flow occurs via turbine stop valve closure, which occurs in approximately 0.1 seconds. Therefore, the transient in primary pressure, temperature, and water volume will be less severe for the loss of external load than for the turbine trip due to a slightly slower loss of heat transfer capability.

The protection available to mitigate the consequences of a loss of external load is the same as that for a turbine trip, as listed in Table 15.0-5.

15.2.2.2 Analysis of Effects and Consequences

a. Method of Analysis

Refer to Subsection 15.2.3.2 for the method used to analyze the limiting transient (turbine trip) in this grouping of events. The results of the turbine trip event analysis are more severe than those expected for the loss of external load, as discussed in Subsection 15.2.2.1.

Normal Reactor Control Systems and Engineered Safety Systems are not required to function. The Emergency Feedwater System may, however, be automatically actuated following a loss of main feedwater, which will further mitigate the effects of the transient.

The Reactor Protection System may be required to function following a complete loss of external load to terminate core heat input and prevent Departure from Nucleate Boiling (DNB). Depending on the magnitude of the load loss, pressurizer safety valves and/or steam generator safety valves may be required to open to maintain system pressure below allowable limits. No single active failure will revent operation of any system required to function.

15.2.2.3 Radiological Consequences

The radiological consequences resulting from this malfunction are considerably less than those calculated for a main steam line rupture. The analyses performed assuming a rupture of a main steam line are given in Subsection 15.1.5.3.

15.2.2.4 Conclusions

Based on results obtained for the turbine trip event (Subsection 15.2.3) and considerations described in Subsection 15.2.2.1, the applicable acceptance criteria for a loss of external load event are met.

15.2.3 Turbine Trip

15.2.3.1 Identification of Causes and Accident Description

For a turbine trip event, the reactor would be tripped directly (unless below the P-9 setpoint) from a signal derived from the turbine emergency trip fluid pressure and turbine stop valves. The turbine stop valves close rapidly (typically 0.1 seconds) on loss of trip fluid pressure actuated by one of a number of possible turbine trip signals. Turbine trip initiation signals include:

a. Electrical faults associated with the generator or transformers

b. Low condenser vacuum

c. Loss of lubricating oil

d. Turbine thrust bearing failure

e. Turbine overspeed

f. Main steam reheat high level

g. Manual trip.

Upon initiation of stop valve closure, steam flow to the turbine stops abruptly. Sensors on the stop valves detect the turbine trip and initiate steam dump and, if above the P-9 setpoint, a reactor trip. The loss of steam flow results in a rapid rise in secondary system temperature and pressure. The turbine trip event is analyzed because it results in the most rapid reduction in steam flow.

The automatic Steam Dump System would normally accommodate the excess steam generation when the unit is operating below the P-9 setpoint. Reactor coolant temperatures and pressure do not significantly increase if the Steam Dump System and Pressurizer Pressure Control System are functioning properly.

If the turbine condenser were not available, the excess steam generation would be dumped to the atmosphere and main feedwater flow would be lost. For this situation, feedwater flow would be maintained by the Auxiliary Feedwater System to ensure adequate residual and decay heat removal capability. Should the Steam Dump System fail to operate, the steam generator safety valves may lift to provide pressure control. See Subsection 15.2.2.1 for a further discussion of the transient.

A turbine trip is classified as an ANS Condition II event, a fault of moderate frequency. See Subsection 15.0.1 for a discussion of Condition II events.

A turbine trip event is more limiting than loss of external load, loss of condenser vacuum, and other turbine trip events. As such, this event has been analyzed in detail. Results and discussion of the analysis are presented in Subsection 15.2.3.2.

The plant systems and equipment available to mitigate the consequences of a turbine trip are discussed in Subsection 15.0.8 and listed in Table 15.0-5.

15.2.3.2 Analysis of Effects and Consequences

a. Method of Analysis

In this analysis, two cases are analyzed. In one case, the behavior of the unit is evaluated for a complete loss of steam load from 100 percent or rated thermal power without direct reactor trip, primarily to show the adequacy of the pressure relieving devices to limit the maximum RCS pressure to 110 percent of its design value. The second case analyzes the accident with respect to determining the minimum DNBR. This second case typically also represents the limiting transient with respect to peak steam generator pressure because it usually results in a longer time to reactor trip. In both cases the turbine trip is assumed to trip without actuating any of the sensors for reactor trip on the turbine stop valves. This assumption delays reactor trip until conditions in the RCS result in a trip due to other signals. Thus, the analysis assumes a worst case transient.

In addition, no credit is taken for steam dump. Main feedwater flow is terminated at the time of turbine trip, with no credit taken for auxiliary feedwater to mitigate the consequences of the transient.

The turbine trip transients are analyzed by employing the detailed digital computer program RETRAN[6]. The program simulates the neutron kinetics, RCS, pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generator, and steam generator safety valves. The program computes pertinent plant variables including temperatures, pressures, and power level.

The following turbine trip cases are analyzed:

A. Minimum reactivity feedback, with RCS pressure control

B. Minimum reactivity feedback, with no RCS pressure control

Case A is performed to calculate a conservative minimum DNBR, and is analyzed using the revised thermal design procedure as described in WCAP-11397[3]. Case B is analyzed to calculate a conservative maximum RCS pressure.

Major assumptions are summarized below:

1. Initial Operating Conditions - For case A, the initial core power, reactor coolant temperature, and pressurizer pressure are assumed to be at their nominal full power values. Uncertainties in initial conditions are included in the limit Departure from Nucleate Boiling Ratio (DNBR) as described in Reference 3.

For Case B, an uncertainty of 50 psi is applied in the most limiting direction to the initial reactor coolant pressure. No uncertainty is applied to the core power as it is already accounted for in the conservatively high nominal core power level assumed. No uncertainty is applied to the nominal full-power value for reactor coolant temperature as this yields more conservative results.

2. Moderator and Doppler Coefficients of Reactivity - The turbine trip is analyzed with minimum reactivity feedback, which assumes a 0 pcm/°F moderator temperature coefficient and a least negative Doppler Power coefficient.

3. Reactor Control - From the standpoint of both the maximum pressures attained and DNBR, it is conservative to assume that the reactor is in manual control. If the reactor were in automatic control, the control rod banks would move prior to trip and reduce the severity of the transient.

4. Steam Release - No credit is taken for operation of the steam dump system or steam generator power-operated relief valves. The steam generator pressure rises to the safety valve setpoint where steam release through safety valves limits secondary steam pressure at the setpoint value.

5. Pressurizer Spray and Power-Operated Relief Valves - Two cases are analyzed:

   a. For evaluating the minimum DNBR, full credit is taken for the effect of pressurizer spray and power-operated relief valves in reducing or limiting the coolant pressure. Pressurizer safety valves are also available. This case results in a delayed reactor trip on overtemperature ΔT which results in a more limiting steam generator pressure transient.

   b. For evaluating maximum RCS pressure, no credit is taken for the effect of pressurizer spray and power-operated relief valves in reducing or limiting the coolant pressure. Pressurizer safety valves are operable.

6. Feedwater Flow - Main feedwater flow to the steam generators is assumed to be lost at the time of turbine trip. No credit is taken for auxiliary feedwater flow since a stabilized plant condition will be reached before auxiliary feedwater initiation is normally assumed to occur. However, the auxiliary feedwater pumps would be expected to start once a steam generator low-low level condition is reached. The auxiliary feedwater flow would remove core decay heat following plant stabilization.

7. Steam Flow - Steam flow is assumed to be lost at the time of turbine trip.

8. Reactor trip is actuated by the first reactor protection system trip setpoint reached with no credit taken for the direct reactor trip on the turbine trip. Trip signals are expected due to high pressurizer pressure, overtemperature ΔT, and low-low steam generator water level.

   Except as discussed above, normal reactor control system and engineered safety systems are not required to function. A case is presented in which pressurizer spray and power-operated relief valves are assumed, but the more limiting case where these functions are not assumed is also presented.

   The reactor protection system may be required to function following a turbine trip. Pressurizer safety valves and/or steam generator safety valves may be required to open to maintain system pressures below allowable limits. No single active failure will prevent operation of any system required to function.

b. Results

The transient responses for a turbine trip from full power operation are shown for two cases: 1) for minimum DNBR and 2) for maximum RCS pressure. The calculated sequence of events for the accident is shown in Table 15.2-1.

Figure 15.2-1, sh. 1, Figure 15.2-1, sh. 2, and Figure 15.2-1, sh. 3 show the transient responses for the turbine trip with minimum reactivity feedback assuming full credit for the pressurizer spray and pressurizer power-operated relief valves. No credit is taken for the steam dump. The reactor is tripped by the overtemperature ΔT trip signal. The minimum DNBR remains well above the limit value. The steam generator safety valves limit the secondary side pressure below 110 percent of the design value.

The turbine trip accident was also studied assuming the plant to be initially operating at 100 percent of rated thermal power with no credit taken for the pressurizer spray, pressurizer power-operated relief valves, or steam dump. The reactor is tripped on the high pressurizer pressure signal. Figure 15.2-1, sh. 4 and Figure 15.2-1, sh. 5 show the transients with minimum reactivity feedback without pressure control. In this case, the pressurizer safety valves are actuated, and maintain system pressure below 110 percent of the design value.

15.2.3.3 Radiological Consequences

The radiological consequences resulting from this malfunction are considerably less than those calculated for a main steam line rupture. The analyses performed assuming a rupture of a main steam line are given in Subsection 15.1.5.3.

15.2.3.4 Conclusions

Results of the analyses show that the plant design is such that a turbine trip without a direct or immediate reactor trip presents no hazard to the integrity of the RCS or the Main Steam System. Pressure relieving devices incorporated in the two systems are adequate to limit the maximum pressures to within the design limits.

The integrity of the core is maintained by operation of the Reactor Protection System, i.e., the DNBR will be maintained above the safety analysis limit value. The above analysis demonstrates the ability of the Nuclear Steam Supply System to safely withstand a full load rejection.

15.2.4 Inadvertent Closure of Main Steam Isolation Valves

Inadvertent closure of the main steam isolation valves would result in a turbine trip. Turbine trips are discussed in Subsection 15.2.3.

15.2.5 Loss of Condenser Vacuum and Other Events Resulting in Turbine Trip

Malfunction of the condenser vacuum pumps, improper valve positioning or excessive air leakage may result in loss of condenser vacuum.

The loss of condenser vacuum is one of the events that will cause a turbine trip. Other turbine trip initiating events are described in Section 10.2 and Subsection 15.2.3. In case of loss of condenser vacuum, the Condenser Steam Dump System cannot be used and the excess steam generated is discharged to the atmosphere through the relief and/or safety valves. On loss of condenser vacuum, an alarm will activate at 5.0"HgA, and at 7.5"HgA, turbine trip will occur.

A turbine trip due to loss of condenser vacuum does not entail more adverse effects than the general turbine trip accident analyzed in detail in Subsection 15.2.3, because in that analysis no credit is taken for condenser steam dump. Therefore, the analysis results and conclusions of Subsection 15.2.3 apply to the loss of condenser vacuum.

15.2.6 Loss of Nonemergency AC Power to The Plant Auxiliaries (Loss of Offsite Power)

15.2.6.1 Identification of Causes and Accident Description

A complete loss of nonemergency AC power may result in the loss of all power to the station auxiliaries, i.e., the reactor coolant pumps, condensate pumps, etc. The loss of power may be caused by a complete loss of the offsite grid accompanied by a turbine generator trip at the plant, or by a loss of the onsite AC distribution system.

For this event the decrease in heat removal by the secondary system is accompanied by a flow coastdown which further reduces the capacity of the primary coolant to remove heat from the core. The reactor will trip: (1) due to turbine trip; (2) upon reaching one of the trip setpoints in the primary and secondary systems as a result of the flow coastdown and decrease in secondary heat removal; or (3) due to loss of power to the control rod drive mechanisms as a result of the loss of power to the plant.

Following a loss of AC power with turbine and reactor trips, the sequence described below will occur:

a. Plant vital instruments are supplied from emergency DC power sources.

b. As the steam system pressure rises following the trip, the steam generator power-operated relief valves may be automatically opened to the atmosphere. The condenser is assumed not to be available for steam dump. If the power-operated relief valves are not available, the steam generator self-actuated safety valves may lift to dissipate the sensible heat of the fuel and coolant plus the residual decay heat produced in the reactor.

c. As the no-load temperature is approached, the steam generator power-operated relief valves (or the safety valves, if the power-operated relief valves are not available) are used to dissipate the residual decay heat and to maintain the plant at the hot shutdown condition.

d. The emergency diesel generators, started on loss of voltage on the plant emergency buses, begin to supply plant vital loads.

The Emergency Feedwater System is started automatically as described below.

Both the motor-driven emergency feedwater pump and the turbine-driven emergency feedwater pump are started on any of the following:

a. Low-low level in any steam generator

b. Any safety injection signal (SIS)

c. Manual actuation.

Refer to Section 6.8 for a discussion of the Emergency Feedwater System.

The motor-driven emergency feedwater pump is supplied power from the ESF buses. The turbine-driven emergency feedwater pump is driven by steam from the secondary system and exhausts to the atmosphere. Both types of pumps will start and supply rated flow within 75 seconds of the initiating signal. The emergency pumps take suction from the condensate storage tank for delivery to the steam generators.

Upon the loss of power to the reactor coolant pumps, coolant flow necessary for core cooling and the removal of residual heat is maintained by natural circulation in the reactor coolant loops.

A loss of nonemergency AC power to the station auxiliaries is classified as an ANS Condition II event, a fault of moderate frequency. See Subsection 15.0.1 for a discussion of Condition II events.

A loss of AC power event is a more limiting event with respect to DNB than the turbine trip initiated decrease in secondary heat removal without loss of AC power, which was analyzed in Subsection 15.2.3. A loss of AC power to the station auxiliaries as postulated above could also result in a loss of normal feedwater if the condensate pumps lose their power supply.

When a loss of nonemergency AC power is the initiating event, the first few seconds of the transient will closely resemble the simulation of the complete loss of reactor coolant flow event (Section 15.3.2), where DNB and core damage due to rapidly increasing core temperature is prevented by promptly tripping the reactor. For the loss of nonemergency AC power scenario, the DNBR results would be less limiting since the reactor is already tripped when RCP coastdown begins. Thus, the DNBR is not evaluated for this event since it would be bounded by the loss of reactor coolant flow analysis.

In addition, the maximum RCS and main steam system (MSS) pressures for this event are bounded by the loss of external electrical load analysis, which demonstrates that the peak pressures remain below 110 percent of the respective design limit values. For the loss of nonemergency AC power event, turbine trip occurs after reactor trip, whereas for loss of external electrical load analysis the turbine trip is the initiating fault. Therefore, the primary/secondary power mismatch and resultant RCS and main steam system heatup and pressurization transients are always more severe for loss of external electrical load than for loss of nonemergency AC power.

Following the reactor coolant pump coastdown caused by the loss of AC power, the natural circulation capability of the RCS will remove residual and decay heat from the core, aided by emergency feedwater in the secondary system. An analysis is presented below to show that the natural circulation flow in the RCS following a loss of AC power event is sufficient to remove residual heat from the core and prevent the pressurizer from becoming water solid.

The loss of nonemergency AC power and the resulting loss of feedwater occurs at the start of the transient. However, the reactor trip and loss of RCS flow, which would normally occur, is not assumed to happen at this time. This causes the primary side coolant to heat up and the steam generator inventory to decrease. The reactor is finally tripped on a low-low steam generator level signal, and at this time, the loss of primary flow due to the loss of AC is assumed to occur.

The above assumptions are more conservative than an actual loss of nonemergency AC because the reactor power is maintained following the loss of AC and loss of feedwater. This minimizes the steam generator heat transfer capability and increases the amount of RCS stored energy at the time of reactor trip and loss of primary coolant flow.

The plant systems and equipment available to mitigate the consequences of a loss of AC power event are discussed in Subsection 15.0.8 and listed in Table 15.0-5.

15.2.6.2 Analysis of Effects and Consequences

a. Method of Analysis

A detailed analysis using the RETRAN computer code (Reference 6) is performed to determine the plant transient following a loss of nonemergency AC power to the plant auxiliaries. The code simulates the core neutron kinetics, reactor coolant system including natural circulation, pressurizer, pressurizer power operated relief valves and safety valves, pressurizer heaters and spray, steam generators, main steam safety valves, and the emergency feedwater system, and computes pertinent variables, including pressurizer pressure, pressurizer water level, steam generator mass, and reactor coolant average temperature.

The assumptions used in the analysis are as follows:

1. The plant is initially operating at an NSSS power level of 3678 MWt.
2. Core residual heat is based on the 1979 version of ANS 5.1[(4)]. 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 reactor trip is assumed.
3. Reactor trip occurs on steam generator low-low level. No credit is taken for immediate release of the control rod drive mechanisms caused by a loss of offsite power.
4. Emergency feedwater at a temperature of 100°F is delivered by one emergency feedwater pump. A total flow of 650 gpm is assumed to be delivered equally to all four steam generators 77 seconds after the steam generator low-low level setpoint is reached.
5. Secondary system steam relief is achieved through the steam generator safety valves.
6. The initial pressurizer pressure is assumed to be 50 psi higher and lower than the nominal value to determine the limiting case.
7. Cases are analyzed assuming initial hot full power reactor vessel average coolant temperatures at the upper and lower ends of the operating range with uncertainty applied in both the positive and negative direction. The vessel average temperature assumed at the upper end of the range is 589.1°F with an uncertainty of +6/-5 °F. The average temperature assumed at the lower end of the range is 571.0°F with an uncertainty of +6/-5 °F. Results for the limiting case are presented.
8. A moderator temperature coefficient of 0 pcm/°F, the least negative Doppler temperature coefficient, and the most negative Doppler-only power were assumed for conservatism.

9. Cases are analyzed assuming initial feedwater temperatures of 452.4°F and 390°F.

10. Analysis with both minimum (0%) and maximum (10%) steam generator tube plugging was performed to conservatively bound potential operating conditions.

11. The pressurizer relief valves, sprays, and heaters are assumed to function.

b. Results

The transient responses of the RCS and the secondary side following a loss of nonemergency AC power are shown in Figure 15.2-5 sheets 1 through 4.

The first few seconds after the loss of power to the reactor coolant pumps will closely resemble a simulation of the complete loss of flow incident (see Subsection 15.3.2), i.e., core damage due to rapidly increasing core temperatures is prevented by promptly tripping the reactor. After the reactor trip, stored and residual decay heat must be removed to prevent damage to either the RCS or the core.

Natural circulation flow is available and is sufficient to provide adequate core decay heat removal following reactor trip and reactor coolant pump coastdown.

As noted previously, the DNBR result for this event is bounded by the complete loss of flow event, and the maximum RCS and MSS pressures for this event are bounded by the loss of external electrical load analysis. The sole acceptance criterion for this analysis is that the pressurizer does not become water solid. Figure 15.2-5, sheet 2 demonstrates that this criterion is met. The calculated sequence of events for this accident is listed in Table 15.2-1.

15.2.6.3 Radiological Consequences

The radiological consequences resulting from this malfunction are considerably less than those calculated for a main steam line rupture. The analyses performed assuming a rupture of a main steam line are given in Subsection 15.1.5.3.

15.2.6.4 Conclusions

Analysis of the natural circulation capability of the RCS has demonstrated that sufficient heat removal capability exists following reactor coolant pump coastdown to prevent fuel or clad damage. The radiological consequences of this event would be less severe than the steam line break event analyzed in Subsection 15.1.5.3.

15.2.7 Loss of Normal Feedwater Flow

15.2.7.1 Identification of Causes and Accident Description

A loss of normal feedwater (from pump failures, valve malfunctions, or loss of offsite AC power) results in a reduction in capability of the secondary system to remove the heat generated in the reactor core. If an alternative supply of feedwater were not supplied to the plant, core residual heat following reactor trip would heat the primary system water to the point where water relief from the pressurizer would occur, resulting in a substantial loss of water from the RCS. Since the plant is tripped well before the steam generator heat transfer capability is reduced, the primary system variables never approach a DNB condition.

The following events occur upon loss of normal feedwater (assuming main feedwater pump failures or valve malfunctions):

a. As the steam system pressure rises following the trip, the steam generator power-operated relief valves are automatically opened to the atmosphere. Steam dump to the condenser is assumed not to be available. If the steam flow rate through the power-operated relief valves is not adequate, the steam generator self-actuated safety valves may lift to dissipate the sensible heat of the fuel and coolant plus the residual decay heat produced in the reactor.

b. As the no-load temperature is approached, the steam generator power-operated relief valves (or the safety valves, if the power-operated relief valves are not available) are used to dissipate the residual decay heat and to maintain the plant at the hot shutdown condition.

A loss of normal feedwater is classified as an ANS Condition II event, a fault of moderate frequency. See Subsection 15.0.1 for a discussion of Condition II events.

Reactor trip on low-low water level in any steam generator provides protection for a loss of normal feedwater.

The Emergency Feedwater System is started automatically as discussed in Subsection 15.2.6.1. The motor-driven emergency feedwater pump is supplied power from the ESF buses. The turbine-driven emergency feedwater pump is driven by steam from the secondary system and exhausts to the atmosphere. The pumps take suction directly from the condensate storage tank for delivery to the steam generators.

The DNBR consequences of this event are bounded by the Loss of load/turbine trip (LOL/TT) event. Both of these events represent a reduction in the heat removal capability of the secondary system. For the loss of normal feedwater event, the RCS temperature increases gradually as the steam generators boil down to the low-low level trip setpoint, at which time reactor trip occurs, followed by turbine trip. For the LOL/TT event, the turbine trip is the initiating event, and the loss of heat sink is much more severe. Therefore, the initial RCS heatup will be much more severe for the LOL/TT event than for the loss of normal feedwater event, and the LOL/TT event will always be more severe with respect to the minimum DNBR criterion.

With respect to system overpressure concerns, the loss of normal feedwater event is also bounded by the LOL/TT event analysis (minimum reactivity feedback, without pressure control). For the loss of normal feedwater event, turbine trip occurs after reactor trip, whereas for LOL/TT the turbine trip is the initiating fault. Therefore, the primary/secondary power mismatch and resultant RCS and main steam system heatup and pressurization transients are always more severe for LOL/TT than for loss of normal feedwater.

An analysis of the system transient is presented below to show that following a loss of normal feedwater, the Emergency Feedwater System is capable of removing the stored and residual heat, thus preventing either overpressurization of the RCS or loss of water from the reactor core, and returning the plant to a safe condition. This is demonstrated by showing that the pressurizer does not become water solid.

15.2.7.2 Analysis of Effects and Consequences

a. Method of Analysis

A detailed analysis using the RETRAN computer code (Reference 6) is performed to determine the plant transient following a loss of normal feedwater. The code simulates the core neutron kinetics, reactor coolant system, pressurizer, pressurizer power operated relief valves and safety valves, pressurizer heaters and spray, steam generators, main steam safety valves, and the emergency feedwater system, and computes pertinent variables, including pressurizer pressure, pressurizer water level, steam generator mass, and reactor coolant average temperature.

Assumptions made in the analysis are:

1. The plant is initially operating at an NSSS power level of 3678 MWt.
2. Core residual heat is based on the 1979 version of ANS 5.1[(4)]. ANSI/ANS-5.1-1979 is a conservative representation of decay energy release rates. Long-term operation at the initial power level preceding the trip is assumed.
3. Reactor trip occurs on steam generator low-low level.

4. Emergency feedwater at a temperature of 100°F is delivered by one emergency feed pump. A total flow of 650 gpm is assumed to be delivered equally to all four steam generators 77 seconds after the steam generator low-low level setpoint is reached.
5. Secondary system steam relief is achieved through the steam generator safety valves.
6. The initial pressurizer pressure is assumed to be 50 psi higher and lower than the nominal value to determine the limiting case.
7. Cases are analyzed assuming initial hot full power reactor vessel average coolant temperatures at the upper and lower ends of the operating range with uncertainty applied in both the positive and negative direction. The vessel average temperature assumed at the upper end of the range is 589.1°F with an uncertainty of +6/-5 °F. The average temperature assumed at the lower end of the range is 571.0°F with an uncertainty of +6/-5 °F. Results for the limiting case are presented.
8. A moderator temperature coefficient of 0 pcm/°F, the least negative Doppler temperature coefficient, and the most negative Doppler-only power were assumed for conservatism.
9. Cases are analyzed assuming initial feedwater temperatures of 452.4°F and 390°F.
10. Analysis with both minimum (0%) and maximum (10%) steam generator tube plugging was performed to conservatively bound potential operating conditions.
11. The pressurizer relief valves, sprays, and heaters are assumed to function.

The loss of normal feedwater analysis is performed to demonstrate the adequacy of the reactor protection and engineered safeguards systems (i.e., the emergency feedwater system) in removing long-term decay heat and preventing excessive heatup of the RCS with possible resultant RCS overpressurization or loss of RCS water.

The assumptions used in the analysis are similar to the loss of AC power incident (Subsection 15.2.6) except that the reactor coolant pumps are assumed to continue to operate.

Plant characteristics and initial conditions are further discussed in Subsection 15.0.3.

Plant systems and equipment which are available to mitigate the effects of a loss of normal feedwater accident are discussed in Subsection 15.0.8 and listed in Table 15.0-5. Normal reactor control systems are not required to function. The Emergency Feedwater System is required to deliver a minimum emergency feedwater flow rate. No single active failure will prevent operation of any system required to function.

b. Results

Figure 15.2-6, sh.1-4 shows the significant plant parameters following a loss of normal feedwater.

Following the reactor and turbine trip from full load, the water level in the steam generators will fall due to the reduction of steam generator void fraction and because steam flow through the safety valves continues to dissipate the stored and generated heat. Within 75 seconds following the initiation of the low-low level trip, the emergency feedwater pumps are automatically started, reducing the rate of water level decrease. The capacity of the emergency feedwater pumps is such that the water level in the steam generators being fed does not recede below the lowest level at which sufficient heat transfer area is available to dissipate core residual heat without water relief from the pressurizer safety valves. Figure 15.2-6, sheet 2 shows that at no time is there water relief from the pressurizer.

The calculated sequence of events for this accident is listed in Table 15.2-1.

As shown in Figure 15.2-6, sheets 1 through 4, the plant approaches a stabilized condition following reactor trip and emergency feedwater initiation at hot standby with the emergency feedwater removing decay heat. The plant may be maintained at hot standby or further cooled through manual control of the emergency feed flow. The operating procedures would also call for operator action to control RCS boron concentration and pressurizer level using the CVCS and to maintain steam generator level through control of the Emergency Feedwater System. Any action required of the operator to maintain the plant in a stabilized condition will be in a time frame in excess of ten minutes following reactor trip.

15.2.7.3 Radiological Consequences

The radiological consequences resulting from this malfunction are considerably less than those calculated for a main steam line rupture. The analyses performed assuming a rupture of a main steam line are given in Subsection 15.1.5.3.

15.2.7.4 Conclusions

Results of the analysis show that a loss of normal feedwater does not adversely affect the core, the RCS, or the steam system since the emergency feedwater capacity is such that sufficient core heat removal is maintained, the RCS does not overpressurize, and reactor coolant water is not relieved from the pressurizer relief or safety valves.

15.2.8 Feedwater System Pipe Break

15.2.8.1 Identification of Causes and Accident Description

A major feedwater line rupture is defined as a break in a feedwater line large enough to prevent the addition of sufficient feedwater to the steam generators to maintain shell-side fluid inventory in the steam generators. If the break is postulated in a feedwater line between the check valve and the steam generator, fluid from the steam generator may also be discharged through the break. A break in this location could preclude the subsequent addition of emergency feedwater to the affected steam generator. Also, all Emergency Feedwater (EFW) flow is assumed to be lost through the break prior to isolation of EFW flow to the faulted steam generator. A break upstream of the feedwater line check valve would affect the NSSS only as a loss of feedwater, which is covered by the analyses in Sections 15.2.6 and 15.2.7.

Depending upon the size of the break and the plant operating conditions at the time of the break, the break could cause either an RCS cooldown (by excessive discharge through the break) or an RCS heatup. Potential RCS cooldown resulting from a secondary pipe rupture is evaluated in Subsection 15.1.5. Therefore, only the RCS heatup effects are evaluated for a feedwater line rupture.

This event is analyzed in order to evaluate the capacity of the emergency feedwater system to remove core decay heat, and to ensure that the core remains in a coolable geometry. In order to demonstrate this, a more limiting criterion is imposed such that the maximum hot leg temperature remains below the saturation temperature until the EFW heat removal capability exceeds the RCS heat generation, which demonstrates that the core remains covered with water.

A major feedwater line rupture is classified as an ANS Condition IV event. See Subsection 15.0.1 for a discussion of Condition IV events.

A feedwater line rupture reduces the ability to remove heat generated by the core from the RCS for the following reasons:

a. Feedwater flow to the steam generators is reduced. Since feedwater is subcooled, its loss may cause reactor coolant temperatures to increase prior to reactor trip.

b. Fluid in the steam generator may be discharged through the break, and would then not be available for decay heat removal after trip.

c. The break may be large enough to prevent the addition of any main feedwater after trip.

An emergency feedwater system is provided to assure that adequate feedwater will be available such that:

a. No substantial overpressurization of the RCS shall occur; and

b. Sufficient liquid in the RCS shall be maintained in order to provide adequate decay heat removal.

A major feedwater line rupture is classified as an ANS Condition IV event.

The severity of the feedwater line rupture transient depends on a number of system parameters including break size, initial reactor power, and credit taken for the functioning of various control and safety systems. Sensitivity studies presented in WCAP-9230[5] illustrate that the most limiting feedwater line rupture is a double-ended rupture of the largest feedwater line. Analyses were performed at full power with and without loss of offsite power. The pressurizer power-operated relief valves were modeled, as their modeling results in more limiting conditions.

The following provides the protection for a main feedwater line rupture:

a. A reactor trip on any of the following conditions:

1. High pressurizer pressure
2. Overtemperature ΔT
3. Low-low steam generator water level in any steam generator
4. Safety injection signals from any of the following:
   - (a) Two out of three low steam line pressure in any one loop,
   - (b) Two out of three high containment pressure (hi-1), or
   - (c) Low pressurizer pressure.

   Refer to Chapter 7 for a discussion of the actuation system.

b. An Emergency Feedwater System to provide an assured source of feedwater to the steam generators for decay heat removal. Refer to Section 6.8 for a description of the Emergency Feedwater System.

15.2.8.2 Analysis of Effects and Consequences

a. Method of Analysis

A detailed analysis using the RETRAN[6] code is performed in order to determine the plant transient following a feedwater line rupture. The code describes the plant thermal kinetics, RCS including natural circulation, pressurizer, steam generators, and feedwater system, and computes pertinent variables including the pressurizer pressure, pressurizer water level, and reactor coolant average temperature.

The cases analyzed assume a double-ended rupture of the largest feedwater pipe at full power. Major assumptions made in the analyses are as follows:

1. The plant is initially operating at an NSSS power of 3678 MWt, which includes calorimetric uncertainties.

2. Initial reactor coolant average temperature is 6.0 degrees F above the nominal value, and the initial pressurizer pressure is 50 psi below its nominal value.

3. Normal reactor control systems are not assumed to function unless their function results in more severe consequences. Therefore, the pressurizer PORVs are assumed to operate normally in order to minimize the RCS pressure.

4. Initial pressurizer level is at the nominal programmed value plus 5% uncertainty, initial steam generator water level is at the nominal value plus 8% in the faulted steam generator, and at the nominal value minus 14.5 percent in the intact steam generators.

5. The worst case assumes minimum reactivity feedback – zero moderator density coefficients, least negative Doppler temperature coefficients, least negative Doppler-only power coefficients and maximum delayed neutron beta-effective values.

6. Main feedwater to all steam generators is assumed to stop at the time the break occurs (all main feedwater spills out through the break).

7. The worst possible break area, a double-ended break downstream of the EFW connection, is assumed. This maximizes the blowdown discharge rate following the time of trip, which maximizes the resultant heatup of the reactor coolant.

8. Choked flow is assumed at the break.

9. The analysis assumes a conservatively low value of 0% NRS for the steam generator low-low level setpoint, which actuated the EFW system.

10. EFW pump performance is based on loss of one train (single failure) and minimum flow versus steam generator back pressure injected to the three intact steam generators by the operational pump. Cold EFW is assumed to not reach the steam generators until the three feedwater branch lines have been swept clear of hot feedwater.

11. Turbine trip is assumed to occur 0.5 seconds after break initiation and no credit is taken for the atmospheric steam dump valves.

12. Safety Injection Actuation is credited on low pressurizer pressure.

13. Minimum high head ECCS pump performance and maximum ECCS temperature (98°F) are assumed. The flow rates assumed conservatively account for flow from only one centrifugal charging pump with 10% head degradation.

14. No credit is taken for heat energy deposited in RCS metal during the RCS heatup.
15. No credit is taken for charging or letdown.
16. Steam generator heat transfer area is assumed to decrease as the shell side liquid inventory decreases.
17. Conservative core residual heat generation is assumed based upon long-term operation at the initial power level preceding the reactor trip is assumed.
18. One of the redundant EFW flow control valves leading to the faulted steam generator is assumed to close on a high flow rate signal with a bounding stroke time of 23 seconds to terminate EFW flow through the break. This stroke time is conservatively modeled as an additional delay over and above the EFW signal delay (2 seconds) and the delay for EFW pump start (77 seconds).
19. No credit is taken for the following potential protection logic signals to mitigate the consequences of the accident:
    (a) High pressurizer pressure
    (b) Overtemperature ΔT
    (c) High pressurizer level.

Receipt of a low-low steam generator water level signal in at least one steam generator starts both the motor-driven emergency feedwater pump and turbine-driven emergency feedwater pump, which in turn initiates emergency feedwater flow to the steam generators. Similarly, receipt of a low steam line pressure signal in at least one steam line initiates a steam line isolation signal which closes all main steam line isolation valves. This signal also gives a safety injection signal which initiates flow of cold borated water into the RCS. The amount of safety injection flow is a function of RCS pressure.

Plant characteristics and initial conditions are further discussed in Subsection 15.0.3.

The Reactor Protection System is required to function following a feedwater line rupture as analyzed here. No single active failure will prevent operation of this system.

The Engineered Safety Systems assumed to function are the Emergency Feedwater System and the Safety Injection System.

Two Emergency Feedwater System configurations were considered. In the first configuration, both emergency feedwater pumps were assumed to operate; however, the emergency feedwater flow control valve to one intact steam generator was assumed to fail closed (single failure). As a result, only two intact steam generators receive emergency feedwater following the break. The flow restrictor and control valves on the faulted loop limit the flow spilling out the break to 750 gpm prior to control valve closure. The flow through the open control valves to the remaining two intact loops is at least 235 gpm each, ensuring the minimum required flow of 470 gpm. The second configuration considered operation of only one of the two emergency feedwater pumps (single failure), providing flow to all three intact steam generators. Flow from the operating emergency feedwater pump spills out of the break in the faulted loop prior to automatic closure of one of the redundant flow control valves. With the control valve closed the intact steam generators in combination will receive the minimum required flow of 470 gpm. The analysis presented was performed using the second configuration. This configuration is slightly more conservative because it maximizes the time elapsed prior to cold emergency feedwater reaching the intact steam generators.

A detailed description and analysis of the Safety Injection System is provided in Section 6.3. The Emergency Feedwater System is described in Section 6.8.

b. Results

Calculated plant parameters following a major feedwater line rupture are shown in Figure 15.2-7. The calculated sequence of events is listed in Table 15.2-1.

The RCS heatup prior to reactor trip is due to loss of subcooling as a result of MFW spillage through the break and the increased secondary temperature and pressure following the turbine trip. Reactor power increases prior to the trip due to the RCS heatup. The primary and secondary systems were calculated to remain below 110 percent of their respective design pressures.

Following the reactor trip, steam flow out the break cools the RCS and eventually causes the pressurizer to empty. However, the core remains covered with water. Low main steam line pressure causes closure of the MSIV's, ends the cooldown period, and starts safety injection. Addition of safety injection flow aids in cooling down the primary and ensures that sufficient fluid exists to keep the core covered with water.

The MSIV closure and resulting increase in steam generator pressure and temperature cause the second RCS heatup. As a result, the rising primary system pressure exceeds the shutoff head of the ECCS pumps and then increases to the pressurizer power operated relief valve setpoint. The heatup ends when the intact steam generators reach their main steam safety valve (MSSV) setpoint and the combination of steam relief through the MSSV's and EFW injection match core decay heat plus RCP heat, the adequacy of the EFW system is demonstrated and the event is terminated.

The maximum hot leg temperature remains below the saturation temperature throughout the transient. Therefore, no fuel damage will occur.

15.2.8.3 Radiological Consequences

No fuel failures are predicted for this event. The radiological consequences resulting from this malfunction are considerably less than those calculated for a main steam line rupture. The analyses performed assuming a rupture of a main steam line are given in Subsection 15.1.5.

15.2.8.4 Conclusions

Results of the analyses show that for the postulated feedwater line rupture, the Emergency Feedwater System capacity is adequate to remove decay heat, to prevent overpressurizing the RCS, and to prevent uncovering the reactor core. The maximum hot leg temperature remains below the saturation temperature. Therefore, no fuel damage will occur.

15.2.9 References

1. Cooper, L., Miselis, V. and Starek, R.M., "Overpressure Protection for Westinghouse Pressurized Water Reactors," WCAP-7769, Revision 1, June, 1972 (also letter NS-CE-622, dated April 16, 1975, C Eicheldinger (Westinghouse) to D. B. Vassallo (NRC), additional information on WCAP-7769, Revision 1)
2. WCAP-7907-P-A, "LOFTRAN Code Description," T. W. T. Burnett, et al., April 1984
3. WCAP-11397-P-A, "Revised Thermal Design Procedure," A. J. Friedland and S. Ray, April 1984
4. ANSI/ANS-5.1-1979, "American National Standard for Decay Heat Power in Light Water Reactors," August 1979
5. WCAP-9230, "Report on the Consequences of a Postulated Main Feedline Rupture," G. E. Lang and J. P. Cunningham, January 1978
6. WCAP-14882-P-A, "RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analyses," D.S. Huegel, et al., April 1999

15.3 DECREASE IN REACTOR COOLANT SYSTEM FLOW RATE

A number of faults are postulated which could result in a decrease in Reactor Coolant System (RCS) flow rate. These events are discussed in this section. Detailed analyses are presented for the most limiting of these events.

Discussions of the following flow decrease events are presented in Section 15.3:

a. Partial Loss of Forced Reactor Coolant Flow

b. Complete Loss of Forced Reactor Coolant Flow

c. Reactor Coolant Pump Shaft Seizure (Locked Rotor)

d. Reactor Coolant Pump Shaft Break.

Item a above is considered to be an ANS Condition II event, item b an ANS Condition III event, and items c and d ANS Condition IV events. Subsection 15.0.1 contains a discussion of ANS classifications.

15.3.1 Partial Loss of Forced Reactor Coolant Flow

15.3.1.1 Identification of Causes and Accident Description

A partial loss-of-coolant flow accident can result from a mechanical or electrical failure in a reactor coolant pump, or from a fault in the power supply to the pump or pumps supplied by a reactor coolant pump bus. If the reactor is at power at the time of the accident, the immediate effect of loss-of-coolant flow is a rapid increase in the coolant temperature. This increase could result in DNB with subsequent fuel damage if the reactor is not tripped promptly.

The plant design is such that the four reactor coolant pumps are supplied through two buses, two pumps per bus, connected to the generator. When a generator trip occurs, the generator breaker is tripped open, the buses are automatically transferred to an offsite power source, and the pumps will continue to supply coolant flow to the core. Following any turbine trip, there is immediate generator trip and automatic transfer of the buses to offsite power.

This event is classified as an ANS Condition II incident (an incident of moderate frequency) as defined in Subsection 15.0.1.

The necessary protection against a partial loss-of-coolant flow accident is provided by the low primary coolant flow reactor trip signal which is actuated in any reactor coolant loop by two out of three low flow signals, in any reactor coolant loop. Above Permissive 8, low flow in any loop will actuate a reactor trip. Between approximately 10 percent power (Permissive 7) and the power level corresponding to Permissive 8, low flow in any two loops will actuate a reactor trip. Above Permissive 7, either power supply low voltage on both buses or opening of one reactor coolant pump breaker on each bus will actuate the corresponding undervoltage relays, resulting in a reactor trip. Additionally, underfrequency on the two buses will actuate a reactor trip above P-7. These trips serve as a backup to the low flow trip.

15.3.1.2 Analysis of Effects and Consequences

a. Method of Analysis

Partial loss of flow involving loss of two pumps with four loops in operation has been analyzed.

This transient is analyzed by two digital computer codes. The RETRAN[1] code is used to calculate the loop and core flow during the transient, the time of reactor trip based on the calculated flows, the nuclear power transient, and the primary system pressure and temperature transients. The VIPRE[4] code is then used to calculate the heat flux and the Departure from Nucleate Boiling Ratio (DNBR) transients based on the nuclear power and RCS flow calculated by RETRAN. The DNBR transients presented represent the minimum of the typical or thimble cell.

This accident is analyzed with the revised thermal design procedure described in WCAP-11397[5].

1. Initial Conditions

   Initial reactor power, pressure, and RCS temperature are assumed to be at their nominal values. Uncertainties in initial conditions are included in the limit DNBR as described in reference 5.

2. Reactivity Coefficients

   A conservatively large absolute value of the Doppler-only power coefficient is used. The most positive moderator temperature coefficient allowed by the Technical Specifications at full power conditions, 0.0 pcm/°F, is assumed. This results in the maximum core power during the initial part of the transient when the minimum DNBR is reached.

3. Flow Coastdown

   The flow coastdown analysis is based on a momentum balance around each reactor coolant loop and across the reactor core. This momentum balance is combined with the continuity equation, a pump momentum balance and conservative pump characteristics.

Plant systems and equipment, which are necessary to mitigate the effects of the accident, are discussed in Subsection 15.0.8 and listed in Table 15.0-5. No single active failure in any of these systems or equipment will adversely affect the consequences of the accident.

b. Results

Figure 15.3-1 and Figure 15.3-2 shows the transient response for the loss of two reactor coolant pumps with four loops in operation. Figure 15.3-2 shows the DNBR to be always greater than the limit value. Since the DNBR limit is not violated, the ability of the primary coolant to remove heat from the fuel rod is not reduced. Thus, the average fuel and clad temperatures do not increase significantly above their respective initial values.

The calculated sequence of events is shown on Table 15.3-1. The affected reactor coolant pumps will continue to coast down, and the core flow will reach a new equilibrium value corresponding to the two pumps still in operation. With the reactor tripped, a stable plant condition will eventually be attained. Normal plant shutdown may then proceed.

15.3.1.3 Radiological Consequences

The radiological consequences of this malfunction are bounded by the results presented in Subsection 15.3.2 (Complete Loss of Forced Reactor Coolant Flow).

15.3.1.4 Conclusions

The analysis shows that the DNBR will not decrease below the safety analysis limit value at any time during the transient. Thus, no fuel or clad damage is predicted, and all applicable acceptance criteria are met.

15.3.2 Complete Loss of Forced Reactor Coolant Flow

15.3.2.1 Identification of Causes and Accident Description

A complete loss of forced reactor coolant flow may result from a simultaneous loss of electrical supplies to all reactor coolant pumps. If the reactor is at power at the time of the accident, the immediate effect of loss-of-coolant flow is a rapid increase in the coolant temperature. This increase could result in DNB with subsequent fuel damage if the reactor were not tripped promptly.

Normal power for the reactor coolant pumps is supplied through buses from a transformer connected to the generator. When a generator trip occurs, the generator breaker is tripped open, the buses are automatically transferred to an offsite power source, and the pumps will continue to supply coolant flow to the core. Following any turbine trip there is immediate generator trip and automatic transfer of the buses to offsite power.

This event is classified as an ANS Condition III incident (an infrequent incident) as defined in Subsection 15.0.1.

The following provide the necessary protection against a complete loss of flow accident:

a. Reactor coolant pump power supply undervoltage or underfrequency

b. Low reactor coolant loop flow.

The reactor trip on reactor coolant pump undervoltage is provided to protect against conditions which can cause a loss of voltage to all reactor coolant pumps, i.e., loss of offsite power. Channel response time includes consideration of the bus voltage decay time due to generated Electro-Motive Force (EMF) from motors connected to the bus as the motors coast down. This function is blocked below approximately 10 percent power (Permissive 7).

The reactor trip on reactor coolant pump underfrequency is provided to trip the reactor for an underfrequency condition, resulting from frequency disturbances on the power grid. Reference 9 provides analyses of grid frequency disturbances and the resulting nuclear steam supply system protection requirements, which are generally applicable.

The reactor trip on low primary coolant loop flow is provided to protect against loss of flow conditions which affect only one reactor coolant loop. This function is generated by two out of three low flow signals per reactor coolant loop. Above Permissive 8, low flow in any loop will actuate a reactor trip. Between approximately 10 percent power (Permissive 7) and the power level corresponding to Permissive 8, low flow in any two loops will actuate a reactor trip. If the maximum grid frequency decay rate is less than approximately 2.5 Hz/second the low flow trip function will protect the core from underfrequency events. This effect is fully described in Reference 9.

15.3.2.2 Analysis of Effects and Consequences

a. Method of Analysis

The complete loss of flow transient has been analyzed for a loss of four pumps with four loops in operation.

This transient is analyzed by two digital computer codes. The RETRAN (Reference 1) Code is used to calculate the loop and core flow during the transient, the time of reactor trip based on the calculated flows, the nuclear power transient, and the primary system pressure and temperature transients. The VIPRE Code (see Section 4.4) is used to calculate the heat flux and DNBR transients based on the RETRAN calculated nuclear power and RCS flow. The DNBR transients presented represent the minimum of the typical or thimble cell.

The method of analysis and the assumptions made regarding initial operating conditions and reactivity coefficients are identical to those discussed in Subsection 15.3.1.2, except that following the loss of power supply to all pumps at power, a reactor trip is actuated by either reactor coolant pump power supply undervoltage or underfrequency.

b. Results

Figure 15.3-3, Figure 15.3-4, and Figure 15.3-5 show the transient response for the loss of power to all reactor coolant pumps. The reactor is assumed to be tripped on an undervoltage signal. Figure 15.3-5 shows the DNBR to be always greater than the limit value. Since the DNBR limit is not violated, the ability of the primary coolant to remove heat from the fuel rod is not greatly reduced. Thus, the average fuel and clad temperatures do not increase significantly above their respective initial values.

The calculated sequence of events for the case analyzed is shown on Table 15.3-1. The reactor coolant pumps will continue to coast down, and natural circulation flow will eventually be established, as demonstrated in Subsection 15.2.6.

With the reactor tripped, a stable plant condition will be attained. Normal plant shutdown may then proceed.

15.3.2.3 Radiological Consequences

No radiological consequences have been calculated for this postulated accident since no fuel or clad damage is predicted.

15.3.2.4 Conclusions

The analysis performed has demonstrated that for the complete loss of forced reactor coolant flow, the DNBR does not decrease below the safety analysis limit at any time during the transient. Thus, no fuel or clad damage is predicted, and all applicable acceptance criteria are met.

15.3.3 Reactor Coolant Pump Shaft Seizure (Locked Rotor

15.3.3.1 Identification of Causes and Accident Description

The accident postulated is an instantaneous seizure of a reactor coolant pump rotor. Flow through the affected reactor coolant loop is rapidly reduced, leading to an initiation of a reactor trip on a low flow signal.

Following initiation of the reactor trip, heat stored in the fuel rods continues to be transferred to the coolant causing the coolant to expand. At the same time, heat transfer to the shell side of the steam generators is reduced, first because the reduced flow results in a decreased tube side film coefficient and then because the reactor coolant in the tubes cools down while the shell side temperature increases (turbine steam flow is reduced to zero upon plant trip). The rapid expansion of the coolant in the reactor core, combined with reduced heat transfer in the steam generators causes an insurge into the pressurizer and pressure increase throughout the Reactor Coolant System. The insurge into the pressurizer compresses the steam volume, actuates the automatic spray system, opens the power-operated relief valves, and opens the pressurizer safety valves, in that sequence. The two power-operated relief valves are designed for reliable operation and would be expected to function properly during the accident. However, for conservatism, their pressure reducing effect as well as the pressure reducing effect of the spray is not included in the analysis.

This event is classified as an ANS Condition IV incident (a limiting fault) as defined in Subsection 15.0.1.

15.3.3.2 Analysis of Effects and Consequences

a. Method of Analysis

Two digital computer codes are used to analyze this transient. The RETRAN Code (Reference 1) is used to calculate the resulting loop and core flow transients following the pump seizure, the time of reactor trip based on the loop flow transients, the nuclear power following reactor trip, and to determine the peak pressure. The thermal behavior of the fuel located at the core hot spot are investigated using the VIPRE Code (Reference 4), using core flow and nuclear power calculated by RETRAN. The VIPRE code includes the use of a film boiling heat transfer coefficient.

At the beginning of the postulated locked rotor accident (i.e., at the time the shaft in one of the reactor coolant pumps is assumed to seize) the plant is assumed to be in operation under the most adverse steady-state operating conditions (i.e., maximum steady-state power level, maximum steady-state pressure, and maximum steady-state coolant average temperature) including uncertainties.

1. Evaluation of the Pressure Transient

   After pump seizure, the neutron flux is rapidly reduced by control rod insertion. Rod motion begins one second after the flow in the affected loop reaches 87 percent of nominal flow. No credit is taken for the pressure reducing effect of the pressurizer relief valves, pressurizer spray, steam dump or controlled feedwater flow after plant trip. Although these systems are expected to function and would result in a lower peak RCS pressure, an additional degree of conservatism is provided by ignoring their effect.

2. Evaluation of DNB in the Core During the Accident

   For this accident, DNB is assumed to occur in the core, and therefore, an evaluation of the consequences with respect to fuel rod thermal transients is performed. Results obtained from analysis of this "hot spot" condition represent the upper limit with respect to clad temperature and zirconium-water reaction.

   In the evaluation, the rod power at the hot spot is assumed to be 2.5 times the average rod power (i.e., $F_Q = 2.5$) at the initial core power level.

   An additional analysis is performed using the VIPRE[4] code to determine the extent of DNB in the core.

   Film Boiling Coefficient

   The film boiling coefficient is calculated in the VIPRE code using the Bishop-Sandberg-Tong film boiling correlation. The fluid properties are evaluated at film temperature (average between wall and bulk temperatures). The program calculates the film coefficient at every time step based upon the actual heat transfer conditions at the time. The neutron flux, system pressure, bulk density, and mass flow rate as a function of time are used as program input.

   For this analysis, the initial values of the pressure and the bulk density are used throughout the transient since they are the most conservative with respect to clad temperature response. For conservatism, DNB was assumed to start at the beginning of the accident.

Fuel Clad Gap Coefficient

The magnitude and time dependence of the heat transfer coefficient between fuel and clad (gap coefficient) has a pronounced influence on the thermal results. The larger the value of the gap coefficient, the more heat is transferred between pellet and clad. Based on investigations on the effect of the gap coefficient upon the maximum clad temperature during the transient, the gap coefficient was assumed to increase from a steady-state value consistent with initial fuel temperature to a very large value of 10,000 Btu/hr-ft^2-°F at the initiation of the transient. Thus, the large amount of energy stored in the fuel because of the small initial value is released to the clad at the initiation of the transient.

Zirconium Steam Reaction

The zirconium-steam reaction can become significant above 1800°F (clad temperature). The Baker-Just parabolic rate equation shown below is used to define the rate of the zirconium steam reaction.

$$\frac{d(w^2)}{dt} = 33.3 \times 10^6 \exp\left(\frac{-45{,}500}{1.986T}\right)$$

where:

w = amount reacted (mg/cm^2)

t = time (seconds)

T = temperature (Kelvin)

The reaction heat is 1,510 cal/g.

The effect of the zirconium-steam reaction is included in the calculation of the "hot spot" temperature transient.

Plant systems and equipment which are necessary to mitigate the effects of the accident are discussed in subsection 15.0.8 and listed in Table 15.0-5. No single active failure in any of these systems or equipment will adversely affect the consequences of the accident.

b. Results

The transient results for the most limiting conditions of the locked rotor and pump shaft break (Subsection 15.3.4) accidents are shown in Figure 15.3-6, sh.1, Figure 15.3-6, sh. 2, and Figure 15.3-6, sh. 3. The results of these calculations are also summarized in Table 15.3-1. The peak RCS pressure reached during the transient is less than that which would cause stresses to exceed the faulted condition stress limits. Also, the peak clad surface temperature is considerably less than 2700°F. It should be noted that the clad temperature was conservatively calculated assuming that DNB occurs at the initiation of the transient.

The calculated sequence of events for the cases analyzed is shown on Table 15.3-1. With the reactor tripped, a stable plant condition will eventually be attained. Normal plant shutdown may then proceed.

15.3.3.3 Radiological Consequences Using Alternate Source Term Methodology

The limiting radiological consequences for the locked rotor and shaft break event are associated with a loss of offsite power and are presented in Section 15.3.4.3.

15.3.3.4 Conclusions

a. Since the peak reactor coolant system pressure reached during any of the transients is less than that which would cause stresses to exceed the faulted condition stress limits, the integrity of the Primary Coolant System is not endangered.

b. Since the peak clad surface temperature calculated for the hot spot during the worst transient remains considerably less than 2700°F, the core will remain in place and intact with no loss of core cooling capability.

c. The doses which have been calculated for the locked rotor accident are below regulatory limits.

15.3.4 Reactor Coolant Pump Shaft Seizure (Locked Rotor) Including Loss of Offsite Power

15.3.4.1 Identification of Causes and Accident Description

In the event of a locked rotor/shaft break of a reactor coolant pump (RCP), the remaining three RCPs will continue to run. Analysis of the breaker coordination shows the following: under all postulated operating conditions, including maximum load of one of the 13.8 kV buses (2 RCPS, 2 circulating water pumps and the 13.8 kV substations) and minimum bus voltage, failure of one RCP (with incipient locked rotor amps) will not result in tripping of the incoming breaker to the 13.8 kV bus. Because of the separate power supply to the other 13.8 kV bus (see Figure 8.3-1), this event will have no effect on the power supply of this bus.

Offsite power will not be lost as a consequence of the event. Subsection 8.2.2.3 provides the results of stability studies showing that the loss of Seabrook Station will not cause a loss of offsite power. Figure 8.3-1 is a one-line diagram of the Electrical Distribution System showing the generator circuit breaker used for isolating the generator without affecting the normal supply to the 13.8 kV bus.

Nevertheless, a bounding evaluation of a locked rotor is provided in the analysis presented in Subsection 15.3.3, which assumed that offsite power is lost coincident with turbine trip. The transient is postulated to occur in the following manner:

a. RCP rotor locks (or shears) and flow in that loop begins to coastdown.

b. The reactor is tripped on low RCS flow in one loop.

c. Turbine-generator trips.

d. Offsite power is lost even though grid stability analyses show it will not be lost.

e. The loss of offsite power causes the three remaining RCPs to coast down.

15.3.4.2 <u>Analysis of Effects and Components</u>

<u>Method of Analysis</u>

The method of analysis used is the same as presented in Subsection 15.3.3. A bounding value of maximum reactor coolant pressure is calculated by assuming offsite power is lost coincident with turbine trip. This assumption is conservative because grid stability analyses show offsite power will not be lost.

15.3.4.3 <u>Radiological Consequences</u>

a. <u>Background</u>

This event is caused by an instantaneous seizure of a primary reactor coolant pump rotor. Flow through the affected loop is rapidly reduced, causing a reactor trip due to a low primary loop flow signal. Fuel damage may be predicted to occur as a result of this accident. Due to the pressure differential between the primary and secondary systems and assumed steam generator tube leakage, fission products are discharged from the primary into the secondary system. A portion of this radioactivity is released to the outside atmosphere from the secondary coolant system through the steam generator via the ASDVs and MSSVs. In addition, radioactivity is contained in the primary and secondary coolant before the accident and some of this activity is released to the atmosphere as a result of steaming from the steam generators following the accident.

b. Compliance with RG 1.183 Regulatory Positions

The revised Locked Rotor dose consequence analysis is consistent with the guidance provided in RG 1.183, Appendix G, "Assumptions for Evaluating the Radiological Consequences of a PWR Locked Rotor Accident," as discussed below:

1. Regulatory Position 1 – The total core inventory of the radionuclide groups utilized for determining the source term for this event is based on RG 1.183, Regulatory Position 3.1, and is provided in Table 15C-1. The inventory provided in Table 15C-1 is then adjusted for the fraction of fuel damaged and a radial peaking factor of 1.65 is applied. The fraction of fission product inventory in the gap available for release due to fuel breach is consistent with Table 3 of RG 1.183.
2. Regulatory Position 2 – Fuel damage is assumed for this event.
3. Regulatory Position 3 – Activity released from the damaged fuel is assumed to mix instantaneously and homogeneously throughout the primary coolant.
4. Regulatory Position 4 – The chemical form of radioiodine released from the damaged fuel is assumed to be 95% cesium iodide (CsI), 4.85% elemental iodine, and 0.15% organic iodide. Iodine releases from the SGs to the environment are assumed to be 97% elemental and 3% organic. These fractions apply to iodine released as a result of fuel damage and to equilibrium iodine concentrations in the RCS and secondary system.
5. Regulatory Position 5.1 – The primary-to-secondary accident induced leakage rate is apportioned between the steam generators, as specified by the Technical Specification Steam Generator Program, as 1 gpm total and 500 gallons per day to any one SG.
6. Regulatory Position 5.2 – The density used in converting volumetric leak rates to mass leak rates is consistent with the basis of surveillance tests used to show compliance with the SG leak rate Technical Specification. For the intact Steam Generators, the primary to secondary leak rate is based on a density of 1.0 gm/cc (cold liquid).
7. Regulatory Position 5.3 – The release of radioactivity is assumed to continue until shutdown cooling is in operation and releases from the steam generators are terminated.
8. Regulatory Position 5.4 – The analysis assumes a coincident loss of offsite power.

9. Regulatory Position 5.5 – All noble gas radionuclides released from the primary system are assumed released to the environment without reduction or mitigation.

10. Regulatory Position 5.6 – The steam generator tubes are assumed to remain covered throughout this event for Seabrook. Therefore, the iodine and transport model for release from the SGs is as follows:

    - Appendix E, Regulatory Position 5.5.1 – All four steam generators are used for plant cooldown. Therefore, the primary-to-secondary leakage is assumed to mix instantaneously and homogeneously with the secondary water without flashing.

    - Appendix E, Regulatory Position 5.5.2 – None of the SG tube leakage is assumed to flash for this event.

    - Appendix E, Regulatory Position 5.5.3 – All of the SG tube leakage is assumed to mix with the bulk water.

    - Appendix E, Regulatory Position 5.5.4 – The radioactivity within the bulk water is assumed to become vapor at a rate that is a function of the steaming rate and the partition coefficient. A partition coefficient of 100 is assumed for the iodine. The retention of particulate radionuclides in the unaffected SG is limited by the moisture carryover from the SG. The same partition coefficient of 100, as used for iodine, is assumed for other particulate radionuclides. This assumption is consistent with the SG carryover rate of less than 1%.

    - Appendix E, Regulatory Position 5.6 – Steam generator tube bundle uncovery is not postulated for this event for Seabrook.

c. Other Assumptions

1. RG 1.183, Section 3.6 – The assumed amount of fuel damage caused by the non-LOCA events is analyzed to determine the fraction of the fuel that reaches or exceeds the initiation temperature of fuel melt and to determine the fraction of fuel elements for which fuel clad is breached. This analysis assumes DNB as the fuel damage criterion for estimating fuel damage for the purpose of establishing radioactivity releases. For the Locked Rotor event, Table 3 of RG 1.183 specifies noble gas, alkali metal, and iodine fuel gap release fractions for the breached fuel.

2. The initial RCS activity is assumed to be at the TS limit of 1.0 μCi/gm Dose Equivalent I-131 and 100/E-bar gross activity. The ratio of radioiodines to other radionuclides, provided in Table 11.1-1, is assumed to be constant and the activities are scaled up to produce the TS limit of 1.0 μCi/gm Dose Equivalent I-131 and 100/E-bar gross activity. The initial SG activity is assumed to be at the TS 3.7.1.4 limit of 0.1 μCi/gm Dose Equivalent I-131. This analysis also conservatively assumes that the initial secondary coolant activity includes 10% of the primary coolant equilibrium concentration of alkali metals.

3. This analysis assumes that the DNB fuel damage is limited to 10% breached fuel assemblies.

d. Methodology

Input assumptions used in the dose consequence analysis of the Locked Rotor event are provided in Table 15.3-3. This event is caused by an instantaneous seizure of a primary reactor coolant pump rotor. Flow through the affected loop is rapidly reduced, causing a reactor trip due to a low primary loop flow signal. Following the reactor trip, the heat stored in the fuel rods continues to be transferred to the reactor coolant. Because of the reduced core flow, the coolant temperatures will rise. The rapid rise in primary system temperatures during the initial phase of the transient results in a reduction in the initial DNB margin and fuel damage.

For the purpose of this dose assessment, a total of 10% of the fuel assemblies are assumed damaged. A radial peaking factor of 1.65 is assumed. The activity released from the fuel is assumed to be released instantaneously and homogeneously through the primary coolant with source term from and release fractions per Appendix G of RG 1.183. Primary coolant is released to the SGs as a result of postulated primary-to-secondary accident induced leakage. Activity is released to the atmosphere via steaming from the steam generator ASDVs and MSSVs until the decay heat generated in the reactor core can be removed by the shutdown cooling system 8 hours into the event. These release assumptions are consistent with the requirements of RG 1.183.

For this event, the Control Room ventilation system cycles through three modes of operation:

- Initially the ventilation system is assumed to be operating in normal mode. The air flow distribution during this mode is 1000 cfm of unfiltered fresh air make up and an assumed value of 150 cfm of unfiltered inleakage.

- After the start of the event, the Control Room is isolated due to a high radiation reading in the Control Room ventilation system. A 30-second delay is applied to account for diesel generator start time, damper actuation time, instrument delay, and detector response time. In this emergency mode, the air flow distribution consists of 600 cfm of filtered makeup flow from the outside, 150 cfm of unfiltered inleakage, and 390 cfm of filtered recirculation flow.
- The Control Room ventilation filter efficiencies that are applied to the filtered makeup and recirculation flows are 99% for particulate, 95% for elemental iodine, and 95% for organic iodine.

e. Radiological Consequences

The atmospheric dispersion factors (*X/Qs*) used for this event for the Control Room dose are based on the location of the release and the pathway for ingress into the CR. These *X/Qs* are summarized in Table 2R-2 and Table 2R-3.

The EAB and LPZ dose consequences are determined using the *X/Q* factors provided in Appendix 2Q for the appropriate time intervals.

The radiological consequences of the Locked Rotor event are analyzed using the RADTRAD-NAI code and the inputs/assumptions previously discussed. As shown in Table 15.3-5, the results for EAB dose, LPZ dose, and Control Room dose are all within the appropriate regulatory acceptance criteria.

15.3.4.4 Conclusion

The transient analysis performed in Subsection 15.3.3 assumes a loss of offsite power coincident with turbine trip. Thus, the conclusions of Subsection 15.3.3.4 apply for a reactor coolant pump shaft seizure followed by a loss of offsite power accident.

Grid stability analyses show that offsite power will not be lost following a turbine trip. However, a conservative radiological dose calculation was performed assuming offsite power is lost at the time of turbine trip and assuming primary to secondary accident induced leakage is apportioned between the steam generators as 1 gpm total and 500 gallon per day to any one steam generator.

15.3.5 Reactor Coolant Pump Shaft Break

15.3.5.1 Identification of Causes and Accident Description

The accident is postulated as an instantaneous failure of an RCP shaft. Flow through the affected reactor coolant loop is rapidly reduced, though the initial rate of reduction of coolant flow is greater for the RCP rotor seizure event (Sections 15.3.3). With a failed shaft the pump impeller could conceivably be free to spin in the reverse direction instead of being in a fixed position. The effect of such reverse spinning is a slight decrease in the end point (steady-state) core flow.

The analysis presented in Sections 15.3.3 represents the limiting condition, assuming a locked rotor for forward flow but a free-spinning shaft for reverse flow in the affected loop.

This event is classified as an ANS Condition IV incident (a limiting fault).

15.3.5.2 Radiological Consequences

The radiological consequences of this malfunction are no worse than those calculated for the locked rotor incident (see Subsection 15.3.3).

15.3.5.3 Conclusion

The conclusions of Section 15.3.3.4 apply for a reactor coolant pump shaft break accident.

15.3.6 References

1. WCAP-14882-P-A, "RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analyses," April 1999

2. WCAP-7908-A, "FACTRAN - A FORTRAN-IV Code for Thermal Transients in a UO_2 Fuel Rod," H. G. Hargrove, December 1989

3. WCAP-7979-A, "TWINKLE - A Multi-Dimensional Neutron Kinetics Computer Code," D. H. Risher, and R. F. Barry, January 1975

4. WCAP-14565-A (Proprietary) and WCAP-15306 (Non-Proprietary), "VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis," Y. X. Sung, et al., October 1999

5. WCAP-11397-P-A, "Revised Thermal Design Procedure," A. J. Friedland and S. Ray, April 1989

6. WCAP-9226-P-A. "Reactor Core Response to Excessive Secondary Steam Releases," S. D. Hollingsworth, et al., January 1998

7. WCAP-9230, "Report on the Consequences of a Postulated Main Feedline Rupture," G. E. Lang and J. P. Cunningham, January 1978

8. WCAP-11394-P-A, "Methodology for the Analysis of the Dropped Rod Event," R. L. Haessler, et al., January 1990

9. WCAP-8424, Revision 1, "An Evaluation of Loss of Flow Accidents Caused by Power System Frequency Transients in Westinghouse PWRs," M. S. Baldwin, et al., June 1975

15.4 REACTIVITY AND POWER DISTRIBUTION ANOMALIES

A number of faults have been postulated which could result in reactivity and power distribution anomalies. Reactivity changes could be caused by control rod motion or ejection, boron concentration changes, or addition of cold water to the Reactor Coolant System (RCS). Power distribution changes could be caused by control rod motion, misalignment, ejection, or by static means such as fuel assembly mislocation. These events are discussed in this section. Detailed analyses are presented for the most limiting of these events.

Discussions of the following incidents are presented:

a. Uncontrolled Rod Cluster Control Assembly bank withdrawal from a subcritical or low power startup condition

b. Uncontrolled Rod Cluster Control Assembly bank withdrawal at power

c. Rod Cluster Control Assembly misalignment

d. Startup of an inactive reactor coolant pump at an incorrect temperature

e. Chemical and Volume Control System malfunction that results in a decrease in the boron concentration in the reactor coolant

f. Inadvertent loading and operation of a fuel assembly in an improper position

g. Spectrum of Rod Cluster Control Assembly ejection accidents.

Items a, b, d, and e above are considered to be ANS Condition II events, item f an ANS Condition III event, and item g an ANS Condition IV event. Item c entails both Condition II and III events. Item d is precluded by technical specifications which prohibit 3-loop operation. Item f is precluded by being detectable without consequence during refueling/startup physics tests. Subsection 15.0.1 contains a discussion of ANS classifications.

15.4.1 Uncontrolled Rod Cluster Control Assembly Bank Withdrawal from a Subcritical or Low Power Startup Condition

15.4.1.1 Identification of Causes and Accident Description

A Rod Cluster Control Assembly (RCCA) withdrawal accident is defined as an uncontrolled addition of reactivity to the reactor core caused by withdrawal of RCCAs, resulting in a power excursion. Such a transient could be caused by a malfunction of the Reactor Control or Rod Control Systems. This could occur with either the reactor subcritical, at Hot Zero Power or at power. The "at power" case is discussed in Subsection 15.4.2.

Procedural controls restrict rod motion if the power range nuclear instruments are inoperable. With RCA Tave less than 551°F and power range NIs inoperable, the motor generator sets can only be energized if the RCS is borated to greater than the all rods out value or if alternate means have been established to ensure that the control and shutdown rods are not capable of being withdrawn.

Although the reactor is normally brought to power from a subcritical condition by means of RCCA withdrawal, initial startup procedures with a clean core call for boron dilution on RCCA withdrawal. The maximum rate of reactivity increase in the case of boron dilution is less than that assumed in this analysis (see Subsection 15.4.6, "Chemical and Volume Control System Malfunction that Results in a Decrease in Boron Concentration in the Reactor Coolant").

The RCCA drive mechanisms are wired into preselected bank configurations which are not altered during reactor life. These circuits prevent the RCCAs from being automatically withdrawn in other than their respective banks. Power supplied to the banks is controlled such that no more than two banks can be withdrawn at the same time and in their proper withdrawal sequence. The RCCA drive mechanisms are of the magnetic latch type, and coil actuation is sequenced to provide variable speed travel. The maximum reactivity insertion rate analyzed in the detailed plant analysis is that occurring with the simultaneous withdrawal of the combination of two sequential control banks having the maximum combined worth at maximum speed.

This event is classified as an ANS Condition II incident (an incident of moderate frequency) as defined in Subsection 15.0.1.

The neutron flux response to a continuous reactivity insertion is characterized by a very fast rise, terminated by the reactivity feedback effect of the negative Doppler coefficient. This self limitation of the power excursion is of primary importance since it limits the power to a tolerable level during the delay time for protective action. Should a continuous RCCA withdrawal accident occur, the transient will be terminated by the following automatic features of the Reactor Protection System:

a. <u>Source Range High Neutron Flux Reactor Trip</u> - Actuated when either of two independent source range channels indicates a neutron flux level above a preselected manually adjusted setpoint. This trip function may be manually bypassed only after an intermediate range flux channel indicates a flux level above a specified level. It is automatically reinstated when both intermediate range channels indicate a flux level below a specified level.

b. <u>Intermediate Range High Neutron Flux Reactor Trip</u> - Actuated when either of two independent intermediate range channels indicates a flux level above a preselected manually adjustable setpoint. This trip function may be manually bypassed only after two of the four power range channels are reading above approximately 10 percent of full power, and is automatically reinstated when three of the four power range channels indicate a power level below this value.

c. Power Range High Neutron Flux Reactor Trip (Low Setting) - Actuated when two out of the four power range channels indicate a power level above approximately 25 percent of full power. This trip function may be manually bypassed when two out of the four power range channels indicate a power level above approximately 10 percent of full power, and is automatically reinstated only after three of the four channels indicate a power level below this value.

d. Power Range High Neutron Flux Reactor Trip (High Setting) - Actuated when two out of the four power range channels indicate a power level above a preset setpoint. This trip function is always active.

e. High Nuclear Flux Rate Reactor Trip - Actuated when the positive rate of change of neutron flux on two out of four nuclear power range channels indicate a rate above the preset setpoint. This trip function is always active.

In addition, control rod stops on high intermediate range flux level (one of two) and high power range flux level (one out of four) serve to discontinue rod withdrawal and prevent the need to actuate the intermediate range flux level trip and the power range flux level trip, respectively.

15.4.1.2 Analysis of Effects and Consequences

a. Method of Analysis

The analysis of the uncontrolled RCCA bank withdrawal from subcritical accident is performed in three stages: (1) an average core nuclear power transient calculation, (2) an average core heat transfer calculation, and (3) the DNBR calculation. The average core nuclear calculation is performed using spatial neutron kinetics methods, TWINKLE (Reference 3), to determine the average power generation with time, including the various total core feedback effects, i.e., Doppler reactivity and moderator reactivity. The average heat flux and temperature transients are determined by performing a fuel rod transient heat transfer calculation in FACTRAN (Reference 2). The average heat flux is next used in VIPRE (described in Reference 4) for the transient DNBR calculation. Plant characteristics and initial conditions are discussed in Subsection 15.0.3.

In order to give conservative results for a startup accident, the following assumptions are made:

1. Since the magnitude of the power peak reached during the initial part of the transient for any given rate of reactivity insertion is strongly dependent on the Doppler coefficient, a conservatively low Doppler power defect of -900 pcm was used. See Subsection 15.0.4 and Table 15.0-3.

2. Contribution of the moderator reactivity coefficient is negligible during the initial part of the transient because the heat transfer time between the fuel and the moderator is much longer than the neutron flux response time. However, after the initial neutron flux peak, the succeeding rate of power increase is affected by the moderator reactivity coefficient. The analysis assumes a moderator temperature coefficient of at least +5 pcm/°F at the zero power nominal temperature.

3. The reactor is assumed to be at Hot Zero Power. This assumption is more conservative than that of a lower initial system temperature. The higher initial system temperature yields a larger fuel water heat transfer coefficient, larger specific heats, and a less negative (smaller absolute magnitude) Doppler coefficient, all of which tend to reduce the Doppler feedback effect, thereby increasing the neutron flux peak. The initial effective multiplication factor is assumed to be 1.0 since this results in the worst nuclear power transient.

4. Reactor trip is assumed to be initiated by power range high neutron flux (low setting). The most adverse combination of instrument and setpoint errors, as well as delays for trip signal actuation and RCCA release, is taken into account. A 10 percent increase is assumed for the power range flux trip setpoint, raising it from the nominal value of 25 percent to 35 percent. Since the rise in the neutron flux is so rapid, the effect of errors in the trip setpoint on the actual time at which the rods are released is negligible. In addition, the reactor trip insertion characteristic is based on the assumption that the highest worth RCCA is stuck in its fully withdrawn position. See Subsection 15.0.5 for RCCA insertion characteristics.

5. The maximum positive reactivity insertion rate assumed is greater than that for the simultaneous withdrawal of the combination of the two sequential control banks having the greatest combined worth at maximum speed (45 inches/minute). Control rod drive mechanism design is discussed in Section 4.6.

6. The most limiting axial and radial power shapes, associated with having the two highest combined worth sequential control banks in their high worth position, is assumed in the DNB analysis.

7. The initial power level was assumed to be below the power level expected for any shutdown condition (10^{-9} of nominal power). The combination of highest reactivity insertion rate and lowest initial power produces the highest peak heat flux.

8. Two reactor coolant pumps are assumed to be in operation consistent with plant operating Mode 3 technical specification requirements. This is conservative with respect to DNB.

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

b. Results

The nuclear power, core heat flux, hot spot fuel average and clad temperature transient results are shown in Figure 15.4-1, Figure 15.4-2 and Figure 15.4-3. The DNB analysis demonstrates that the DNBR remains above the applicable safety analysis limit value at all times.

The calculated sequence of events for this accident is shown in Table 15.4-1. With the reactor tripped, the plant returns to a stable condition. The plant may subsequently be cooled down further by following normal plant shutdown procedures.

15.4.1.3 Radiological Consequences

No radiological consequences have been calculated for this postulated accident since no fuel or clad damage is predicted.

15.4.1.4 Conclusions

In the event of a RCCA withdrawal accident from the subcritical condition, the core and the Reactor Coolant System are not adversely affected, since the DNBR is greater than the limit value for all regions of the core. Thus, no fuel or clad damage is predicted as a result of DNB.

15.4.2 Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power

15.4.2.1 Identification of Causes and Accident Description

Uncontrolled Rod Cluster Control Assembly (RCCA) bank withdrawal at power results in an increase in the core heat flux. Since the heat extraction from the steam generator lags behind the core power generation until the steam generator pressure reaches the relief or safety valve setpoint, there is a net increase in the reactor coolant temperature. Unless terminated by manual or automatic action, the power mismatch and resultant coolant temperature rise could eventually result in DNB. Therefore, in order to avert damage to the fuel clad, the Reactor Protection System is designed to terminate any such transient before the DNBR falls below the safety analysis limit value.

This event is classified as an ANS Condition II incident (an incident of moderate frequency) as defined in Subsection 15.0.1.

The automatic features of the Reactor Protection System which prevent core damage following the postulated accident include:

a. Power range neutron flux instrumentation actuates a reactor trip if two out of four channels exceed an Overpower setpoint.

b. Reactor trip is actuated if any two-of-four channels exceed a rate lag setpoint on the high positive neutron flux rate setpoint.

c. Reactor trip is actuated if any two out of four ΔT channels exceed an Overtemperature ΔT setpoint. This setpoint is automatically varied with axial power imbalance, coolant temperature and pressure to protect against DNB.

d. Reactor trip is actuated if any two out of four ΔT channels exceed an Overpower ΔT setpoint. This setpoint is automatically varied with axial power imbalance and coolant temperature to protect against centerline melting.

e. A high pressurizer pressure reactor trip actuated from any two out of four pressure channels, which is set at a fixed point. This set pressure is less than the set pressure for the pressurizer safety valves.

f. A high pressurizer water level reactor trip actuated from any two out of three level channels when the reactor power is above approximately 10 percent (Permissive P7).

Figure 15.0-1 presents allowable reactor coolant loop average temperature and ΔT for the design power distribution and flow as a function of primary coolant pressure. The boundaries of operation defined by the Overpower ΔT trip and the Overtemperature ΔT trip are represented as "protection lines" on this diagram. The protection lines are drawn to include all adverse instrumentation and setpoint errors so that under nominal conditions trip would occur well within the area bounded by these lines. The utility of this diagram is that the limit imposed by any given DNBR can be represented as a line. The DNB lines represent the locus of conditions for which the DNBR equals the safety analysis limit value. All points below and to the left of a DNB line for a given pressure have a DNBR greater than the safety analysis limit value. The diagram shows that DNB is prevented for all cases if the area enclosed with the maximum protection lines is not traversed by the applicable DNBR line at any point.

The area of permissible operation (power, pressure and temperature) is bounded by the combination of the following reactor trips: high neutron flux (fixed setpoint); high pressurizer pressure (fixed setpoint); low pressurizer pressure (fixed setpoint); Overpower and Overtemperature ΔT (variable setpoints), and the opening of the steam generator safety valves.

15.4.2.2 **Analysis of Effects and Consequences**

a. Method of Analysis

This transient is analyzed by the RETRAN Code (Reference 15). This code simulates the neutron kinetics, Reactor Coolant System, pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generator, and steam generator safety valves. The code computes pertinent plant variables including temperatures, pressure, and power level. The core limits as illustrated in Figure 15.0-1 are used as input to RETRAN to determine the minimum DNBR during the transient.

This accident is analyzed with the revised thermal design procedure, as described in WCAP-11397[5]. In order to obtain conservative results for an uncontrolled rod withdrawal at power accident, the following assumptions are made:

1. Initial reactor power, pressure, and RCS temperatures are assumed to be at their nominal values. Uncertainties in initial conditions are included in the limit DNBR as described in Reference 5.

2. Reactivity coefficients - Two cases are analyzed:

   (a) Minimum reactivity feedback

   A positive moderator coefficient of reactivity is assumed corresponding to the beginning of core life. A least negative Doppler power coefficient is assumed.

   (b) Maximum reactivity feedback

   A conservatively large positive moderator density coefficient and a most negative Doppler power coefficient are assumed, corresponding to the end of core life.

3. The reactor trip on high neutron flux is assumed to be actuated at a conservative value of 118 percent of nominal full power. The overtemperature ΔT trip includes all adverse instrumentation and setpoint errors with maximum delays for trip actuation.

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

5. The maximum positive reactivity insertion rate is greater than that for the simultaneous withdrawal of the combination of the two control banks having the maximum combined worth, at maximum speed.

The effect of RCCA movement on the axial core power distribution is accounted for by the f(ΔI) penalty function, which decreases the overtemperature ΔT setpoint proportional to the decrease in margin to DNB.

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

b. Results

Figure 15.4-4, sh.1, Figure 15.4-4, sh.2, and Figure 15.4-4, sh.3, the transient response for a rapid RCCA withdrawal incident starting from full power. Reactor trip on high neutron flux occurs shortly after the start of the accident. Since the neutron flux increase is rapid with respect to the thermal time constant, small changes in coolant temperature and pressure result and margin to DNB is maintained.

The transient response for a slow RCCA withdrawal from full power is shown in Figure 15.4-5, sh.1, Figure 15.4-5, sh.2, and Figure 15.4-5, sh.3. Reactor trip on overtemperature ΔT occurs after a longer period and the rise in temperature and pressure is consequently larger than for rapid RCCA withdrawal. Again, the minimum DNBR is greater than the limit value.

Figure 15.4-6 shows the minimum DNBR as a function of reactivity insertion rate from initial full power operation for minimum and maximum reactivity feedback. It can be seen that two reactor trip channels provide protection over the whole range of reactivity insertion rates. These are the high neutron flux and overtemperature ΔT channels. The minimum DNBR is never less than the limit value.

Figure 15.4-7 and Figure 15.4-8 show the minimum DNBR as a function of reactivity insertion rate for RCCA withdrawal incidents starting at 60 and 10 percent power respectively. The results are similar to the 100 percent power case, except as the initial power is decreased, the range over which the overtemperature ΔT trip is effective is increased. In both cases the DNBR remains above the limit value.

The shape of the curves of minimum DNBR versus reactivity insertion rate in the referenced figures is due both to reactor core and coolant system transient response and to protection system action in initiating a reactor trip.

Referring to Figure 15.4-7, the 60 percent power minimum feedback case, it is noted that:

1. For high reactivity insertion rates (i.e., between ~ 12 pcm/sec and 110 pcm/sec) reactor trip is initiated by the high neutron flux trip for the minimum reactivity feedback cases. The neutron flux level in the core rises rapidly for these insertion rates while core heat flux and coolant system temperature lag behind due to the thermal capacity of the fuel and coolant system fluid. Thus, the reactor is tripped prior to significant increase in heat flux or water temperature with resultant high minimum DNB ratios during the transient. As reactivity insertion rate decreases, core heat flux and coolant temperatures can remain more nearly in equilibrium with the neutron flux; minimum DNB ratio during the transient thus decreases with decreasing insertion rate.

2. The overtemperature ΔT channels initiate a reactor trip when measured coolant ΔT exceeds a setpoint based on measured reactor coolant system average temperature and pressure. It is important in this context to note that the average temperature contribution to the circuit as well as the measured ΔT that is compared to the setpoint are lead-lag compensated in order to decrease the effect of the thermal capacity of the RCS in response to power increases.

3. With further decrease in reactivity insertion rate, the overtemperature ΔT and high neutron flux trips become equally effective in terminating the transient (i.e. at ~ 12 pcm/sec reactivity insertion rate).

For reactivity insertion rates less than ~ 12 pcm/sec, the effectiveness of the overtemperature ΔT trip increases (in terms of increased minimum DNBR) due to the fact that with lower insertion rates the power increase rate is slower, the rate of rise of average coolant temperature is slower and the system lags and delays become less significant.

Figure 15.4-6, Figure 15.4-7, and Figure 15.4-8 illustrate the minimum DNBR calculated for minimum and maximum reactivity feedback at 100, 60, and 10 percent power, respectively.

Since the RCCA withdrawal at power incident is an overpower transient, the fuel temperatures rise during the transient until after reactor trip occurs. For high reactivity insertion rates, the overpower transient is fast with respect to the fuel rod thermal time constant, and the core heat flux lags behind the neutron flux response. Due to this lag, the peak core heat flux does not exceed 118 percent of its nominal value (i.e., the high neutron flux trip setpoint assumed in the analysis). Taking into account the effect of the RCCA withdrawal on the axial core power distribution, the peak fuel temperature will still remain below the fuel melting temperature.

For slow reactivity insertion rates, the core heat flux remains more nearly in equilibrium with the neutron flux. The overpower transient is terminated by the Overtemperature ΔT reactor trip before a DNB condition is reached. The peak heat flux again is maintained below 118 percent of its nominal value. Taking into account the effect of the RCCA withdrawal on the axial core power distribution, the peak fuel centerline temperature will remain below the fuel melting temperature.

Since DNB does not occur at any time during the RCCA withdrawal at power transient, the ability of the primary coolant to remove heat from the fuel rod is not reduced. Thus, the fuel cladding temperature does not rise significantly above its initial value during the transient.

The calculated sequence of events for this accident for a transient initiated at full power is shown on Table 15.4-1. With the reactor tripped, the plant eventually returns to a stable condition. The plant may subsequently be cooled down further by following normal plant shutdown procedures.

15.4.2.3 Radiological Consequences

No radiological consequences have been calculated for this postulated accident since no fuel or clad damage is predicted.

15.4.2.4 Conclusions

The high neutron flux and overtemperature ΔT trip channels provide adequate protection over the entire range of possible reactivity insertion rates, such that the minimum value of DNBR remains above the limit value.

15.4.3 Rod Cluster Control Assembly Misoperation (System Malfunction or Operator Error)

15.4.3.1 Identification of Causes and Accident Description

Rod Cluster Control Assembly (RCCA) misalignment accidents include:

a. One or more dropped RCCAs within the same group

b. A dropped RCCA bank

c. Statically misaligned RCCA

d. Withdrawal of a single RCCA.

Each RCCA has a position indicator channel which displays position of the assembly. The displays of assembly positions are grouped for the operator's convenience. Fully inserted assemblies are further indicated by a rod at bottom signal, which actuates a local alarm and a control room annunciator. Group demand position is also indicated.

Full length RCCAs are always moved in preselected banks, and the banks are always moved in the same preselected sequence. Each bank of RCCAs is divided into two groups. The two groups in a bank move sequentially such that the first group is always within one step of the second group in the bank. A definite schedule of actuation (or deactuation of the stationary gripper, movable gripper, and lift coils of a mechanism) is required to withdraw the RCCA attached to the mechanism. Since the stationary gripper, movable gripper, and lift coils associated with the four RCCAs of a rod group are driven in parallel, any single failure which would cause rod withdrawal would affect a minimum of one group. Mechanical failures are in the direction of insertion, or immobility.

The dropped RCCA, dropped RCCA bank, and statically misaligned assembly events are classified as ANS Condition II incidents (incidents of moderate frequency) as defined in Subsection 15.0.1. However the single RCCA withdrawal incident is classified as an ANS Condition III event, as discussed below.

No single electrical or mechanical failure in the Rod Control System could cause the accidental withdrawal of a single RCCA from the inserted bank at full power operation. The operator could withdraw a single RCCA in the control bank since this feature is necessary to retrieve an assembly should one be accidentally dropped. The event analyzed must result from multiple wiring failures or multiple significant operator errors and subsequent and repeated operator, disregard of event indication. The probability of such a combination of conditions is so low that the limiting consequences may include slight fuel damage.

Thus, consistent with the philosophy and format of ANSI N18.2, the event is classified as a Condition III event. By definition, "Condition III occurrences include incidents, any one of which may occur during the lifetime of a particular plant," and "shall not cause more than a small fraction of fuel elements in the reactor to be damaged..."

This selection of criterion is in accordance with General Design Criterion (GDC) 25 which states, "The protection system shall be designed to assure that specified acceptable fuel design limits are not exceeded for any <u>single</u> malfunction of the reactivity control systems, such as accidental withdrawal (not ejection or dropout) of control rods." (Emphasis has been added.) It has been shown that single failures resulting in RCCA bank withdrawals do not violate specified fuel design limits. Moreover, no single malfunction can result in the withdrawal of a single RCCA. Thus, it is concluded that the criterion established for the single rod withdrawal at power is appropriate and in accordance with GDC 25.

A dropped RCCA or RCCA bank is detected by:

a. Sudden drop in the core power level as seen by the Nuclear Instrumentation System

b. Asymmetric power distribution as seen on out-of-core neutron detectors or core exit thermocouples

c. Rod at bottom signal

d. Rod deviation alarm or

e. Rod position indication.

Misaligned RCCAs are detected by:

a. Asymmetric power distribution as seen on out-of-core neutron detectors or core exit thermocouples

b. Rod deviation alarm or

c. Rod position indicators.

The resolution of the rod position indicator channel is ± 1.7 percent of span (± 2.5 inches). Deviation of any RCCA from its group by twice this distance (3.4 percent of span, or 5 inches) will not cause power distributions worse than the design limits. The deviation alarm alerts the operator to rod deviation with respect to the group position in excess of 5.1 percent of span. If the rod deviation alarm is not operable, the operator is required to take action as required by the technical specifications.

If one or more rod position indicator channels should be out of service, detailed operating instructions shall be followed to assure the alignment of the nonindicated RCCA. The operator is also required to take action as outlined by the Technical Specifications.

In the extremely unlikely event of simultaneous electrical failures which could result in single RCCA withdrawal, rod deviation and rod control urgent failure would both be displayed on the plant annunciator, and the rod position indicators would show the relative positions of the assemblies in the bank. The urgent failure alarm also inhibits automatic rod motion in the group in which it occurs. Withdrawal of a single RCCA by operator action, whether deliberate or by a combination of errors, would result in activation of the same alarm and the same visual indications. Withdrawal of a single RCCA results in both positive reactivity insertion tending to increase core power, and an increase in local power density in the core area associated with the RCCA. Automatic protection for this event is provided by the Overtemperature ΔT reactor trip, although due to the increase in local power density it is not possible in all cases to provide assurance that the core safety limits will not be violated.

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

15.4.3.2 Analysis of Effects and Consequences

a. Dropped RCCAs, Dropped RCCA Bank, and Statically Misaligned RCCA

1. Method of Analysis

(a) One or More Dropped RCCAs from the Same Group

The LOFTRAN[1] is used to calculate the transient system response for the evaluation of the dropped RCCA event. The code simulates the neutron kinetics, RCS, pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generator, and steam generator safety valves. The code computes pertinent plant variables including temperatures, pressures, and power level.

Transient statepoints (temperature, pressure and power) are calculated by LOFTRAN and nuclear models are used to obtain a hot channel factor consistent with the primary system conditions and reactor power. By incorporating the primary conditions from the transient analysis and the hot channel factor from the nuclear analysis, the DNB design basis is shown to be met using dropped rod limit lines developed with the Westinghouse version of VIPRE-01 code (VIPRE)[4]. The transient response analysis, nuclear peaking factor analysis, and performance of the DNB design basis confirmation are performed in accordance with the methodology described in WCAP-11394[8]. Note that the analysis does not take credit for the negative flux rate reactor trip.

A generic statepoint analysis for this event, which was performed in 1986 to bound a number of four-loop PWRs, was evaluated and determined to remain applicable to Seabrook. With the generic statepoints being applicable, the effects of the power uprate are accounted for in the DNB analysis, which is performed on a cycle specific basis.

(b) Dropped RCCA Bank

A dropped RCCA bank results in a symmetric power change in the core. As discussed in reference 8, assumptions made for the dropped RCCA(s) analysis provide a bounding analysis for the dropped RCCA bank.

(c) Statically Misaligned RCCA

Steady-state power distributions are analyzed using appropriate nuclear physics computer codes. The peaking factors are then compared to peaking factor limits developed using the VIPRE code, which are based on meeting the DNBR design criterion. The analysis examines the following cases: the worst rod withdrawn with bank D inserted at the insertion limit, the worst rod dropped with bank D inserted at the insertion limit, and the worst rod dropped with all other rods out, all with the reactor at full power. The analysis assumes this incident to occur at the time in core life with maximum predicted peaking factors.

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. Either reactivity feedback or control bank withdrawal will reestablish power. Partially dropped RCCA results are bounded by the fully dropped RCCA results. The operator may manually retrieve the RCCA by following approved operating procedures.

Following a dropped rod event in manual rod control, the plant will establish a new equilibrium condition. Without control system interaction, a new equilibrium is achieved at a reduced power level and reduced primary temperature. Thus, the automatic rod control mode of operation is the limiting case.

For a dropped RCCA event in the automatic rod control mode, 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. Figure 15.4-9, sh.1, and Figure 15.4-9, sh.2 show a typical transient response to a dropped RCCA (or RCCAs) in the automatic rod control mode. 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 large negative reactivity insertion. The core is not adversely affected during the insertion period, since power is decreasing rapidly. The transient will proceed as described for a dropped RCCA above. In most cases, the negative reactivity worth of a dropped RCCA bank is greater than the available positive reactivity worth associated with the automatic withdrawal of control bank D from its full power insertion limit. However, in the case of a relatively low worth dropped RCCA bank, such as Shutdown Bank A, the available positive reactivity worth from automatic control bank D withdrawal may exceed the negative reactivity worth of the dropped bank by a small amount. Therefore, a power overshoot may still occur in the case of a dropped RCCA bank, due to the combined effects of both automatic withdrawal of bank D and moderator temperature reactivity feedback. However, the magnitude of the possible power overshoot is smaller with a dropped RCCA bank, than it is with single or multiple dropped RCCAs, due to the greater worth of the entire dropped bank, when compared to the available D-bank worth. In addition, the power distribution associated with a dropped RCCA bank is symmetric, resulting in lower peaking factors, when compared to the asymmetric power distributions associated with single or multiple dropped RCCAs. Therefore, the minimum DNBR for a dropped RCCA bank event is bounded by the DNBR associated with single or multiple dropped RCCAs. 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 inserted to its insertion limit with one RCCA fully withdrawn. Multiple independent alarms, including a bank insertion limit alarm, alert the operator well before the postulated conditions are approached. The bank can be inserted to its insertion limit with any one assembly fully withdrawn without the DNBR falling below the limit value. Any action required of the operator to maintain the plant in a stabilized condition will be in a time frame in excess of ten minutes following the incident.

The insertion limits in the technical specifications may vary from time to time depending on a number of limiting criteria. The full power insertion limits on control bank D must be chosen to meet minimum DNBR and peaking factor criterion under normal and misaligned rod conditions. However, the actual insertion limit is usually dictated by other criterion. Detailed results will vary from cycle to cycle depending on fuel arrangements.

The RCCA misalignment cases are analyzed using the revised thermal design procedure as described in WCAP-11397[5]. The initial reactor power, pressure, and RCS temperatures are at their nominal values, but with the increased radial peaking factor associated with the misaligned RCCA. Uncertainties in the initial conditions are included in the limit DNBR value.

For the RCCA misalignment case with bank D inserted to its full power insertion limit and one RCCA fully withdrawn, DNBR does not fall below the limit value.

Calculations have not been performed specifically for RCCAs misaligned from other control banks, which are permitted to be either fully or partially inserted at part power conditions. However, it has been determined on a generic basis that the increase in radial peaking factor necessary to reach the DNBR limit at reduced power conditions, is greater than the credible increase in radial peaking factors associated with reduced thermal power levels and deeper permitted control bank insertion. Therefore, the full power case discussed above with bank D at the insertion limit is more limiting than any credible part power RCCA misalignment scenario involving rods at the insertion limit.

For RCCA misalignments with one RCCA fully inserted, the DNBR does not fall below the limit value.

DNB does not occur for the RCCA misalignment incident and thus the ability of the primary coolant to remove heat from the fuel rod is not reduced. The peak fuel temperature corresponds to a linear heat generation rate based on the radial peaking factor penalty associated with the misaligned RCCA and the design axial power distribution. The resulting linear heat generation is well below that which would cause fuel melting.

Following the identification of an RCCA misalignment condition by the operator, the operator is required to take action as required by the plant technical specifications and operating instructions.

b. Single RCCA Withdrawal

1. Method of Analysis

Core power distributions simulating a single RCCA withdrawal event are calculated using the computer code ANC. The case of the worst rod withdrawn from bank D inserted at the insertion limit, with the reactor initially at full power, is identified and analyzed. The purpose of this calculation is to confirm that the number of fuel rods that go through DNB is less than the safety analysis limit of 5%. The ANC calculated peaking factors are compared to the design peaking factor used to set the overtemperature ΔT trip. Overtemperature ΔT trip setpoints are established to prevent exceeding DNBR limits. If the calculated peaking factors are above the design peaking factor limit, including appropriate calculational uncertainty, a fuel census is generated for the most limiting case to determine the percentage of rods in the core which exceed the design peaking factor. All rods which exceed the design peaking factor are assumed to undergo DNB prior to reaching the power and coolant conditions that would trip the plant on overtemperature ΔT.

The ANC calculations are performed at the time in core life which has the highest peak $F_{\Delta H}$. Power distributions are generated for all unique combinations of bank D inserted to the full power insertion limit, with one bank D RCCA fully withdrawn. Xenon reconstruction is used to skew the axial flux difference to the upper allowable limit. The most limiting configuration is determined by the case that produces the highest peaking factors under these conditions.

2. Results

For the single rod withdrawal event, two cases have been considered as follows:

(a) If the reactor is in the manual control mode, continuous withdrawal of a single RCCA results in both an increase in core power and coolant temperature, and an increase in the local hot channel factor in the area of the withdrawing RCCA. In terms of the overall system response, this case is similar to those presented in Subsection 15.4.2; however, the increased local power peaking in the area of the withdrawn RCCA may result in lower minimum DNBRs than for the withdrawn bank cases. Depending on initial bank insertion and location of the withdrawn RCCA, automatic reactor trip may not occur sufficiently fast to prevent the minimum core DNB ratio from falling below the limit value. Evaluation of this case at the power and coolant conditions at which the Overtemperature ΔT trip would be expected to trip the plant shows that an upper limit for the number of rods with a DNBR less than the limit value is 5 percent.

(b) If the reactor is in the automatic control mode, the multiple failures that result in the withdrawal of a single RCCA will result in the immobility of the other RCCAs in the controlling bank. The transient will then proceed in the same manner as Case (a) described above.

For such cases as above, a reactor trip will ultimately ensue, although not sufficiently fast in all instances to prevent a minimum DNBR in the core of less than the limit value. Following reactor trip, normal shutdown procedures are followed.

15.4.3.3 Radiological Consequences

No radiological consequences have been calculated for these postulated accidents since no significant fuel or clad damage is predicted. The case of the accidental withdrawal of a single RCCA has an upper limit potential of some clad damage; however, the radiological releases and offsite doses are bounded by the results of Subsection 15.4.8.3 (radiological consequences for the spectrum of rod ejection accidents).

15.4.3.4 Conclusions

For cases of dropped RCCAs (including partially dropped RCCAs) or dropped banks, the DNBR remains above the limit value and core damage does not occur.

For all cases of any RCCA fully inserted, or bank D inserted to its rod insertion limits with any single RCCA in that bank fully withdrawn (static misalignment), the DNBR remains greater than the limit value.

For the case of the accidental withdrawal of a single RCCA, with the reactor in the automatic or manual control mode and initially operating at full power with bank D at the insertion limit, an upper bound of the number of fuel rods experiencing DNB is 5 percent of the total fuel rods in the core.

15.4.4 Startup of an Inactive Reactor Coolant Pump at an Incorrect Temperature

15.4.4.1 Identification of Causes and Accident Description

If the plant were allowed to operate with one pump out of service, there would be reverse flow through the inactive loop due to the pressure difference across the reactor vessel. The cold leg temperature in an inactive loop is identical to the cold leg temperature of the active loops (the reactor core inlet temperature). If the reactor is operated at power, and assuming the secondary side of the steam generator in the inactive loop is not isolated, there is a temperature drop across the steam generator in the inactive loop and, with the reverse flow, the hot leg temperature of the inactive loop is lower than the reactor core inlet temperature.

Starting of an idle reactor coolant pump without bringing the inactive loop hot leg temperature close to the core inlet temperature would result in the injection of cold water into the core, which would cause a reactivity insertion and subsequent power increase.

This event is classified as an ANS Condition II incident (a fault of moderate frequency) as defined in Subsection 15.0.1.

Should the startup of an inactive reactor coolant pump accident occur, the transient will be terminated automatically by a reactor trip on low coolant loop flow when the power range neutron flux (two out of four channels) exceeds the P-8 setpoint, which has been previously reset for three loop operation.

15.4.4.2 Analysis of Effects and Consequences

Three loop operation at Seabrook Station is prohibited by technical specifications. Therefore this event was not analyzed.

15.4.5 A Malfunction or Failure of the Flow Controller in a BWR Loop That Results in an Increased Reactor Coolant Flow Rate

Not applicable to Seabrook.

15.4.6 Chemical and Volume Control System Malfunction that Results in a Decrease in the Boron Concentration in the Reactor Coolant

15.4.6.1 Identification of Causes and Accident Description

Reactivity can be added to the core by feeding primary grade water into the RCS via the CVCS. Boron dilution is a manual operation under strict administrative controls with procedures calling for a limit on the rate and duration of dilution. A boric acid blend system is provided to permit the operator to match the boron concentration of reactor coolant makeup water during normal charging to that in the RCS. The boric acid from the boric acid tank is blended with primary grade water in the blender and the composition is determined by the preset flow rates of the boric acid and primary grade water on the control board. The CVCS is designed to limit, even under various postulated failure modes, the potential rate of dilution to a value which, after indication through alarms and instrumentation, provides the operator sufficient time to correct the situation in a safe and orderly manner.

The inadvertent opening of the Reactor Makeup Water (RMW) control valve in conjunction with a failure in the blend system permitting 0 ppm water to flow from the discharge of a single RMW pump to the charging pump suction is considered the limiting ANS Condition II boron dilution event for all modes of operation. In order for makeup water to be added to the RCS at pressure, at least one charging pump must be running in addition to an RMW pump.

Information on the status of the RMW is continuously available to the operator. Lights are provided on the control board to indicate the operating condition of the pumps in the CVCS. Alarms are actuated to warn the operator if boric acid or demineralized water flowrates deviate from preset values as a result of system malfunction.

The inadvertent dilution from this source can readily be terminated by closing the reactor makeup control valve or stopping the RMW pump.

The rate of unborated makeup water addition to the RCS for this worst-case scenario is limited to the discharge flow capacity of a single RMW pump to the CVCS boric acid blender (150 gpm).

An additional source of unborated water which can dilute the reactor coolant is the Boron Thermal Regeneration System (BTRS). Borated RCS water is depleted of boron as it passes through the BTRS.

The BTRS is capable of supplying diluent at a rate comparable to that of one RMW pump. However, water from the BTRS is passed to the CVCS Volume Control Tank (VCT) where it mixes with water maintained at or nearly equal to the RCS boron concentration. Because of the size of the VCT and the mixing of BTRS diluent with water in the VCT, inadvertent operation of the BTRS is capable of creating only a mild boron dilution transient, which is bounded by the limiting scenario discussed above. The BTRS is excluded as a source of unborated water during refueling, cold shutdown, and hot shutdown since Technical Specifications require the BTRS be rendered inoperable in these modes.

Regardless of the cause of a dilution event, numerous alarms and indications including a shutdown monitor system alarm will alert the operator to a potential loss of shutdown margin. The Shutdown Monitor System augments the source range nuclear instrumentation by monitoring for statistically significant increases in the excore neutron flux, as an indication of a potential return to criticality. Specifically, when the neutron count rate increases by more than a preset ratio an alarm is generated. Further description of the Shutdown Monitor System is provided in Section 7.6.11.

The boron dilution event is classified as an ANS Condition II incident (a fault of moderate frequency) as defined in Subsection 15.0.1.

15.4.6.2 Method of Analysis

To cover all phases of plant operation, boron dilution during refueling, cold and hot shutdown, hot standby, startup and power operation are considered in this analysis.

a. Dilution during Refueling

The following conditions are assumed for an uncontrolled boron dilution during refueling:

1. The maximum boron concentration to lose all shutdown margin conservatively bounds the condition of All Rods In (ARI) most reactive time in life, no xenon, with $T_{avg} \leq 140°F$.

2. The boron concentration of all filled portions of the Reactor Coolant System and the refueling canal shall be maintained uniform and sufficient to ensure that the more restrictive of the following reactivity conditions is met:

   a. A K_{eff} of 0.95 or less; or

   b. A boron concentration of greater than or equal to 2,100 ppm.

3. Dilution flow is assumed to be 150 gpm.

4. Mixing of the reactor coolant is accomplished by the operation of at least one residual heat removal pump.

5. A minimum water volume (3,395 ft^3) in the Reactor Coolant System is used. This is the minimum volume of the RCS for residual heat removal system operation. The water in the reactor vessel is assumed to be drained so that the nozzles are half-filled. The total volume includes the reactor vessel up to the nozzle centerline, one hot leg half filled up to the RHR connection, two cold legs half filled up to the RHR connections, and the active volume of one RHR loop.

6. The density of RCS fluid is assumed to be 61.4 lb/ft^3.

b. <u>Dilution during Cold Shutdown (with Filled Loops)</u>

1. The maximum boron concentration required to lose all shutdown margin conservatively bounds the condition of All Rods In (ARI) less the highest worth assembly, most reactive time in life, no xenon, with $T_{avg} \leq 200°F$.

2. Technical Specifications and the Core Operating Limits Report require a minimum shutdown margin. The assumed initial boron concentration is the minimum boron concentration required to meet this shutdown margin.

3. The assumed dilution flowrate is 150 gpm.

4. Mixing of the reactor coolant is accomplished by the operation of one residual heat removal pump.

5. A minimum water volume of 3,992 ft^3 in the Reactor Coolant System is used. The total volume includes the reactor vessel excluding the upper head region, one hot leg up to the RHR connection, two cold legs up to the RHR connections, and the active volume of the smaller RHR loop.

6. The density of RCS fluid is assumed to be 60.1 lb/ft^3.

c. <u>Dilution during Cold Shutdown (with Drained Loops)</u>

1. Technical Specifications require that 2000 ppm be maintained in this condition. The initial boron concentration is assumed to be 2000 ppm.

2. The maximum boron concentration to lose all shutdown margin is identical to the case with filled loops.

3. The assumed dilution flowrate is 150 gpm.

4. Mixing of the reactor coolant is accomplished by operation of at least one residual heat removal pump.

5. A minimum water volume of 3,395 ft^3 in the Reactor Coolant System is used. This is the minimum volume of the RCS for Residual Heat Removal System operation as described under "Dilution During Refueling."

6. The density of RCS fluid is assumed to be 60.1 lb/ft^3.

d. Dilution during Hot Shutdown

1. The maximum boron concentration to lose all shutdown margin conservatively bounds the condition of zero power, ARI less the highest worth rod, most reactive time in life, no xenon, with 200°F ≤ T_{avg} ≤ 350°F.
2. Technical Specifications and the Core Operating Limits Report require a minimum shutdown margin. The assumed initial boron concentration is the minimum boron concentration required to meet this shutdown margin.
3. The assumed dilution flowrate is 150 gpm.
4. A minimum water volume of 3,992 ft^3 in the RCS is used.
5. The density of RCS fluid is assumed to be 55.6 lb/ft^3 (350°F, saturated condition conservatively used. RCS is maintained 50°F subcooled).

e. Dilution during Hot Standby

For the bounding case in this operational mode, the reactor is assumed to be initially subcritical with all rods in less the highest worth rod and with the Technical Specification and the Core Operating Limits Report requirement for shutdown margin met using soluble boron.

1. The maximum boron concentration to lose all shutdown margin conservatively bounds the condition of zero power, ARI less the highest worth rod, most reactive time in life, no xenon, with 350°F ≤ T_{avg} ≤ 557°F.
2. Technical Specifications and the Core Operating Limits Report require a minimum shutdown margin. The assumed initial boron concentration is the minimum boron concentration required to meet this shutdown margin.
3. Dilution flow is assumed to be limited to the capacity of one RMW pump (150 gpm).
4. A minimum water volume (8,645.9 ft^3) in the Reactor Coolant System is used. This volume corresponds to the active volume of the Reactor Coolant System, minus the pressurizer and surge line volumes.
5. Mixing of the reactor coolant is accomplished by operation of the reactor coolant pumps.
6. The density of the RCS fluid is assumed to be 46.4 lb/ft^3 (557°F and 2250 psia).

f. Dilution During Startup (Mode 2)

The following conditions are assumed for an uncontrolled boron dilution during startup.

1. The dilution flow rate is assumed to be limited to the capacity of one RMW pump with the reactor coolant system at pressure (approximately 150 gpm).

2. A minimum water volume (9818.3 ft^3) in the RCS is used. This is a conservative estimate of the active volume of the RCS minus the pressurizer volume, and accounts for 10% steam generator tube plugging.

3. The initial condition in the analysis is assumed to be during the dilution, corresponding to a critical, hot zero power condition with the control rods at the rod insertion limits. A reactor trip on source range high neutron flux is assumed to occur at this condition, alerting the operator to the dilution in progress. The maximum boron concentration at which the reactor will again attain criticality with all rods inserted less the most reactive RCCA stuck out of the core is taken as 1750 ppm. The minimum change from this condition to the initial condition at the rod insertion limits is taken as 200 ppm.

g. Dilution During Power Operation (Mode 1)

The following conditions are assumed for an uncontrolled boron dilution during power operation.

1. During power operation, the plant may be operated two ways: under manual operator control or under automatic rod control. While in manual or automatic rod control, the dilution flow rate is assumed to be the maximum flow capacity of a single RMW pump or 150 gpm.

2. A minimum water volume (9818.2 ft^3) in the RCS is used. This is a conservative estimate of the active volume of the RCS minus the pressurizer volume, and accounts for 10% steam generator tube plugging.

3. For the case of manual reactor control, the initial condition in the analysis is assumed to correspond to a critical, hot full power condition with the control rods at the rod insertion limits. Dilution causes the power and RCS temperature to rise, resulting in a reactor trip on overtemperature ΔT or high neutron flux, alerting the operator to the dilution in progress. The maximum boron concentration at which the reactor will again attain criticality at hot zero power with all rods inserted less the most reactive RCCA stuck out of the core is taken as 1750 ppm. The minimum change from this condition to the initial condition of hot full power at the rod insertion limits is taken as 200 ppm.

For the case of automatic reactor control, the initial condition in the analysis is assumed to correspond to a critical, hot full power condition with the control rods at the rod insertion limits. The operator will be alerted to the dilution in progress by the low-low rod insertion limit alarm. The maximum boron concentration at which the reactor will attain criticality at hot zero power with all rods inserted less the most reactive RCCA stuck out of the core is taken as 1750 ppm. The minimum change from this condition to the initial condition of hot full power at the rod insertion limits is taken as 200 ppm.

15.4.6.3 Results of Analysis

a. Dilution during Refueling

For dilution during refueling, the minimum time required for the shutdown margin to be lost after a Shutdown Monitor System alarm or other alarm will allow at least a half hour for the operator to prevent a loss of all shutdown margin.

b. Dilution during Cold Shutdown (with Filled Loops)

For dilution during cold shutdown with filled loops, the minimum time required for the shutdown margin to be lost after a Shutdown Monitor System alarm or other alarm will allow at least 15 minutes for the operator to prevent a loss of all shutdown margin.

c. Dilution during Cold Shutdown (with Drained Loops)

For dilution during cold shutdown with drained loops, the minimum time required for the shutdown margin to be lost after a Shutdown Monitor System alarm or other alarm will allow at least 15 minutes for the operator to prevent a loss of all shutdown margin.

d. Dilution during Hot Shutdown

For dilution during hot shutdown, the minimum time required to lose all shutdown margin after a Shutdown Monitor System alarm or other alarm will allow at least 15 minutes for the operator to prevent a loss of all shutdown margin.

e. Dilution during Hot Standby

For dilution during hot standby, the minimum time required to lose all shutdown margin after a Shutdown Monitor System alarm or other alarm will allow at least 15 minutes for the operator to prevent a loss of all shutdown margin.

f. Dilution During Startup (Mode 2)

In the event of an unplanned approach to criticality or dilution during power escalation while in the startup mode, the operator has at least 15 minutes following reactor trip on source range high neutron flux until the loss of shutdown margin.

g. Dilution During Power Operation (Mode 1)

During full power operation with the reactor in manual control, the operator has at least 15 minutes following reactor trip on overtemperature ΔT until the loss of shutdown margin. The maximum reactivity insertion rate resulting from the boron dilution is 1.4 pcm/sec.

During full power operation with the reactor in automatic control, the operator has at least 15 minutes following the low-low rod insertion limit alarm until the loss of shutdown margin.

15.4.6.4 Radiological Consequences

No radiological consequences have been calculated for this postulated event since no fuel or clad damage is predicted.

15.4.6.5 Conclusions

The results presented above show that for all the operating modes, there is adequate time for the operator to terminate an unplanned boron dilution event prior to loss of all shutdown margin. Following termination of the dilution flow, the reactor will be in a stable condition with no fuel damage. The calculated sequence of events for the limiting cases described above is shown in Table 15.4-1.

15.4.7 Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position

15.4.7.1 Identification of Causes and Accident Description

Fuel and core loading errors, such as can arise from the inadvertent loading of one or more fuel assemblies into improper positions, loading a fuel rod during manufacture with one or more pellets of the wrong enrichment or the loading of a full fuel assembly during manufacture with pellets of the wrong enrichment, will lead to increased heat fluxes if the error results in placing fuel in core positions calling for fuel of lesser enrichment.

Any error in enrichment, beyond the normal manufacturing tolerances, can cause power shapes which are more peaked than those calculated with the correct enrichments. There is a 5 percent uncertainty margin included in the design value of power peaking factor assumed in the analysis of Condition I and Condition II transients. Successful completion of the reload startup physics tests provides assurance that the plant can be operated as designed.

To reduce the probability of core loading errors, strict administrative controls are placed on the entire core loading sequence. Then, using the core loading patterns from the just completed cycle and the ensuing cycle along with the spent fuel pool map, a core loading sequence is developed. The core loading sequence provides the step-by-step instructions necessary to end up with the desired core configuration. Lastly, as part of the reload process, fuel assembly identification numbers and component types are verified during the reload process. This positive identification along with the multiple independent verification programs utilized during the placement in the reactor vessel provides additional assurance that the fuel is loaded in accordance with the core loading pattern.

The power distortion due to any combination of misplaced fuel assemblies would significantly raise peaking factors and would be readily observable with fixed incore detectors located in about one third of the fuel assemblies in the core. Each fixed incore detector also includes a core exit thermocouple which would also indicate any abnormally high coolant enthalpy rise. Incore flux measurements are taken during the startup subsequent to every refueling operation.

This event is classified as an ANS Condition III incident (an infrequent incident) as defined in Subsection 15.0.1.

15.4.7.2 Radiological Consequences

Any localized fuel or clad damage that may result for this postulated accident or from enrichment errors is assumed to result in radiological consequences which are less severe than those presented in Subsection 15.4.8.3 (radiological consequences for the spectrum of rod ejection accidents).

15.4.7.3 Conclusions

Fuel assembly enrichment errors would be prevented by administrative procedures implemented in fabrication.

In the event that a single pin or pellet has a higher enrichment than the nominal value, the consequences in terms of reduced DNBR and increased fuel and clad temperatures will be limited to the incorrectly loaded pin or pins and perhaps the immediately adjacent pins.

Fuel assembly loading errors are prevented by administrative procedures implemented during core loading. In the unlikely event that a loading error occurs, the resulting power distribution effects will either be readily detected by the reload startup test program or will cause a sufficiently small perturbation to be acceptable within the uncertainties allowed between nominal and design power shapes.

15.4.8 Spectrum of Rod Cluster Control Assembly Ejection Accidents

15.4.8.1 Identification of Causes and Accident Description

This accident is defined as the mechanical failure of a control rod mechanism pressure housing resulting in the ejection of a RCCA and drive shaft. The consequence of this mechanical failure is a rapid positive reactivity insertion together with an adverse core power distribution, possibly leading to localized fuel rod damage.

a. Design Precautions and Protection

Certain features in the Seabrook pressurized water reactor are intended to preclude the possibility of a rod ejection accident, or to limit the consequences if the accident were to occur. These include a sound, conservative mechanical design of the rod housings, together with a thorough quality control (testing) program during assembly, and a nuclear design which lessens the potential ejection worth of RCCAs, and minimizes the number of assemblies inserted at high power levels.

1. Mechanical Design

The mechanical design is discussed in Section 4.6. Mechanical design and quality control procedures intended to preclude the possibility of a RCCA drive mechanism housing failure are listed below:

(a) Each full length control rod drive mechanism housing is completely assembled and shop tested at 4100 psi.

(b) The mechanism housings are individually hydrotested after they are attached to the head adapters in the reactor vessel head, and checked during the hydrotest of the completed Reactor Coolant System.

(c) Stress levels in the mechanism are not affected by anticipated system transients at power, or by the thermal movement of the coolant loops. Moments induced by the design earthquake can be accepted within the allowable primary working stress range specified by the American Society of Mechanical Engineers (ASME) Code, Section III, for Class I components.

(d) The latch mechanism housing and rod travel housing are each a single length of forged Type 304 stainless steel. This material exhibits excellent notch toughness at all temperatures which will be encountered.

A significant margin of strength in the elastic range together with the large energy absorption capability in the plastic range gives additional assurance that gross failure of the housing will not occur. The joints between the latch mechanism housing and head adapter, and between the latch mechanism housing and rod travel housing, are threaded joints reinforced by canopy-type rod welds. Administrative regulations require periodic inspections of these (and other) welds.

2. Nuclear Design

Even if a rupture of a RCCA drive mechanism housing is postulated, the operation of a plant utilizing chemical shim is such that the severity of an ejected RCCA is inherently limited. In general, the reactor is operated with the RCCAs inserted only far enough to permit load follow. Reactivity changes caused by core depletion and xenon transients are compensated by boron concentration changes. Further, the location and grouping of control RCCA banks are selected during the nuclear design to lessen the severity of an RCCA ejection accident. Therefore, should an RCCA be ejected from its normal position during full power operation, only a minor reactivity excursion, at worst, could be expected to occur.

However, it may be occasionally desirable to operate with larger than normal insertions. For this reason, a rod insertion limit is defined as a function of power level. Operation with the RCCAs above this limit guarantees adequate shutdown capability and acceptable power distribution. The position of all RCCAs is continuously indicated in the control room. An alarm will occur if a bank of RCCAs approaches its insertion limit or if one RCCA deviates from its bank.

Operating instructions require boration at the low insertion limit level alarm and emergency boration at the low-low level alarm.

3. Reactor Protection

The reactor protection in the event of a rod ejection accident has been described in Reference 10. The protection for this accident is provided by high neutron flux trip (high and low setting) and high rate of neutron flux increase trip. These protection functions are described in detail in Section 7.2.

Procedural controls restrict rod motion if the power range nuclear instruments are inoperable. With RCA Tave less than 551°F and power range NIs inoperable, the motor generator sets can only be energized if the RCS is borated to greater than the all rods out value or if alternate means have been established to ensure that the control and shutdown rods are not capable of being withdrawn.

4. Effects on Adjacent Housings

Disregarding the remote possibility of the occurrence of an RCCA mechanism housing failure, investigations have shown that failure of a housing due to either longitudinal or circumferential cracking would not cause damage to adjacent housings.

5. Effects of Rod Travel Housing Longitudinal Failures

If a longitudinal failure of the rod travel housing should occur, the region of the position indicator assembly opposite the break would be stressed by the reactor coolant pressure of 2250 psia. The most probable leakage path would be provided by the radial deformation of the position indicator coil assembly, resulting in the growth of axial flow passages between the rod travel housing and the hollow tube along which the coil assemblies are mounted.

If failure of the position indicator coil assembly should occur, the resulting free radial jet from the failed housing could cause it to bend and contact adjacent rod housings. If the adjacent housings were on the periphery, they might bend outward from their bases. The housing material is quite ductile; plastic hinging without cracking would be expected. Housings adjacent to a failed housing, in locations other than the periphery, would not bend because of the rigidity of multiple adjacent housings.

6. Effects of Rod Travel Housing Circumferential Failures

If circumferential failure of a rod travel housing should occur, the broken-off section of the housing would be ejected vertically because the driving force is vertical and the position indicator coil assembly and the drive shaft would tend to guide the broken-off piece upwards during its travel. Travel is limited by the missile shield, thereby limiting the projectile acceleration. When the projectile reached the missile shield, it would partially penetrate the shield and dissipate its kinetic energy. The water jet from the break would continue to push the broken-off piece against the missile shield.

If the broken-off piece of the rod travel housing was short enough to clear the break when fully ejected, it would rebound after impact with the missile shield. The top end plates of the position indicator coil assemblies would prevent the broken piece from directly hitting the rod travel housing of a second drive mechanism. Even if a direct hit by the rebounding piece was to occur, the low kinetic energy of the rebounding projectile would not cause significant damage.

7. Possible Consequences

From the above discussion, the probability of damage to an adjacent housing must be considered remote. However, even if damage is postulated, it would not be expected to lead to a more severe transient since RCCAs are inserted in the core in symmetric patterns, and control rods immediately adjacent to worst ejected rods are not in the core when the reactor is critical. Damage to an adjacent housing could, at worst, cause the RCCA not to fall on receiving a trip signal; however, this is already taken into account in the analysis by assuming a stuck rod adjacent to the ejected rod.

8. Summary

The considerations given above lead to the conclusion that failure of a control rod housing, due either to longitudinal or circumferential cracking, would not cause damage to adjacent housings that would increase severity of the initial accident.

b. Limiting Criteria

This event is classified as an ANS Condition IV incident. See Subsection 15.0.1 for a discussion of ANS classifications. Due to the extremely low probability of a RCCA ejection accident, some fuel damage could be considered an acceptable consequence.

Comprehensive studies of the threshold of fuel failure and of the threshold of significant conversion of the fuel thermal energy to mechanical energy, have been carried out as part of the SPERT project by the Idaho Nuclear Corporation (Reference 11).

Extensive tests of UO_2 zirconium clad fuel rods representative of those in pressurized water reactor-type cores have demonstrated failure thresholds in the range of 240 to 257 cal/gm. However, other rods of a slightly different design have exhibited failures as low as 225 cal/gm.

These results differ significantly from the TREAT (Reference 12) results, which indicated a failure threshold of 280 cal/gm. Limited results have indicated that this threshold decreases by about 10 percent with fuel burnup. The clad failure mechanism appears to be melting for zero burnup rods and brittle fracture for irradiated rods. Also important is the conversion ratio of thermal to mechanical energy. This ratio becomes marginally detectable above 300 cal/gm for unirradiated rods and 200 cal/gm for irradiated rods; catastrophic failure, (large fuel dispersal, large pressure rise) even for irradiated rods, did not occur below 300 cal/gm.

In view of the above experimental results, criteria are applied to ensure that there is little or no possibility of fuel dispersal in the coolant, gross lattice distortion, or severe shock waves. These criteria are:

1. Average fuel pellet enthalpy at the hot spot below 200 cal/gm for unirradiated and irradiated fuel.

2. Peak reactor coolant pressure less than that which could cause stresses to exceed the faulted condition stress limits.

3. Fuel melting will be limited to less than 10 percent of the fuel volume at the hot spot even if the average fuel pellet enthalpy is below the limits of criterion 1 above.

It should be noted that the original FSAR included an additional criterion that the average clad temperature at the hot spot must remain below 2700°F. The elimination of this criterion as a basis for evaluating the RCCA ejection accident results is consistent with the revised Westinghouse acceptance criteria for this event[(13)].

15.4.8.2 Analysis of Effects and Consequences

a. Method of Analysis

The calculation of the RCCA ejection transient is performed in two stages, first an average core channel calculation and then a hot region calculation. The average core calculation is performed using spatial neutron kinetics methods to determine the average power generation with time including the various total core feedback effects; i.e., Doppler reactivity and moderator reactivity. Enthalpy and temperature transients in the hot spot are then determined by multiplying the average core energy generation by the hot channel factor and performing a fuel rod transient heat transfer calculation. The power distribution calculated without feedback is pessimistically assumed to persist throughout the transient. A detailed discussion of the method of analysis can be found in Reference 10.

b. Average Core Analysis

The spatial kinetics computer code, TWINKLE[3], is used for the average core transient analysis. This code solves the two-group neutron diffusion theory kinetic equation in one, two, or three spatial dimensions (rectangular coordinates) for six delayed neutron groups and up to 8000 spatial points. The computer code includes a detailed multiregion, transient fuel clad coolant heat transfer model for calculation of pointwise Doppler and moderator feedback effects. In this analysis, the code is used as a one-dimensional axial kinetics code since it allows a more realistic representation of the spatial effects of axial moderator feedback and RCCA movement. However, since the radial dimension is missing, it is still necessary to employ very conservative methods (described below) of calculating the ejected rod worth and hot channel factor.

c. Hot Spot Analysis

In the hot spot analysis, the initial heat flux is equal to the nominal times the design hot channel factor. During the transient, the heat flux hot channel factor is linearly increased to the transient value in 0.1 second, the time for full ejection of the rod. Therefore, the assumption is made that the hot spot before and after ejection are coincident. This is very conservative since the peak after ejection will occur in or adjacent to the assembly with the ejected rod, and prior to ejection the power in this region will necessarily be depressed.

The hot spot analysis is performed using the detailed fuel and clad transient heat transfer computer code, FACTRAN[2]. This computer code calculates the transient temperature distribution in a cross section of a metal clad UO_2 fuel rod, and the heat flux at the surface of the rod, using as input the nuclear power versus time and the local coolant conditions. The zirconium-water reaction is explicitly represented, and all material properties are represented as functions of temperature. A conservative radial power distribution is used within the fuel rod.

FACTRAN uses the Dittus-Boelter or Jens-Lottes correlation to determine the film heat transfer before DNB, and the Bishop-Sandberg-Tong correlation to determine the film boiling coefficient after DNB. The Bishop-Sandberg-Tong correlation is conservatively used assuming zero bulk fluid quality. The DNB ratio is not calculated, instead the code is forced into DNB by specifying a conservative DNB heat flux. The gap heat transfer coefficient can be calculated by the code; however, it is adjusted in order to force the full power steady-state temperature distribution to agree with the fuel heat transfer design codes.

d. System Overpressure Analysis

Because safety limits for fuel damage specified earlier are not exceeded, there is little likelihood of fuel dispersal into the coolant. The pressure surge may, therefore, be calculated on the basis of conventional heat transfer from the fuel and prompt heat generation in the coolant.

The pressure surge is calculated by first performing the fuel heat transfer calculation to determine the average and hot spot heat flux versus time. Using this heat flux data, a thermal hydraulic calculation is conducted to determine the volume surge. Finally, the volume surge is simulated in the NSSS plant transient computer code. This code calculates the pressure transient taking into account fluid transport in the RCS and heat transfer to the steam generators. No credit is taken for the possible pressure reduction caused by the assumed failure of the control rod pressure housing. The system overpressure is generically addressed in Reference 10.

Calculation of Basic Parameters

Input parameters for the analysis are conservatively selected on the basis of values calculated for this type of core. The more important parameters are discussed below. Table 15.4-2 presents the parameters used in this analysis

a. Ejected Rod Worths and Hot Channel Factors

The values for ejected rod worths and hot channel factors are calculated using either three-dimensional static methods or by a synthesis method employing one-dimensional and two-dimensional calculations. Standard nuclear design codes are used in the analysis. No credit is taken for the flux flattening effects of reactivity feedback. The calculation is performed for the maximum allowed bank insertion at a given power level, as determined by the rod insertion limits. Adverse xenon distributions are considered in the calculation to provide worst case results.

Appropriate margins are added to the ejected rod worth and hot channel factors to account for any calculational uncertainties.

Power distribution before and after ejection for a "worst case" can be found in Reference 10. During plant startup physics testing, ejected rod worths and power distributions are measured in the zero and full power configurations and compared to values used in the analysis. Experience has shown that the ejected rod worth and power peaking factors are consistently overpredicted in the analysis.

b. Reactivity Feedback Weighting Factors

The largest temperature rises, and hence the largest reactivity feedbacks, occur in channels where the power is higher than average. Since the weight of a region is dependent on flux, these regions have high weights. This means that the reactivity feedback is larger than that indicated by a simple channel analysis. Physics calculations have been carried out for temperature changes with a flat temperature distribution, and with a large number of axial and radial temperature distributions. Reactivity changes were compared and effective weighting factors determined. These weighting factors take the form of multipliers which, when applied to single channel feedbacks, correct them to effective whole core feedbacks for the appropriate flux shape. In this analysis, since a one dimensional (axial) spatial kinetics method is employed, axial weighting is not necessary if the initial condition is made to match the ejected rod configuration. In addition, no weighting is applied to the moderator feedback. A conservative radial weighting factor is applied to the transient fuel temperature to obtain an effective fuel temperature as a function of time accounting for the missing spatial dimension. These weighting factors have also been shown to be conservative compared to three-dimensional analysis.

c. Moderator and Doppler Coefficient

The critical boron concentrations at the beginning of life and end of life are adjusted in the nuclear code in order to obtain moderator density coefficient curves which are conservative compared to actual design conditions for the plant. As discussed above, no weighting factor is applied to these results.

The Doppler reactivity defect is determined as a function of power level using a one-dimensional steady-state computer code with a Doppler weighting factor of 1.0. The Doppler weighting factor will increase under accident conditions, as discussed above.

d. Delayed Neutron Fraction, βeff

Calculations of the effective delayed neutron fraction (βeff) typically yield values no less than 0.70% at beginning of life and 0.50% at end of life for the first cycle. The accident is sensitive to βeff if the ejected rod worth is equal to or greater than βeff as in zero power transients. In order to allow for future cycles, pessimistic estimates of 0.54% at beginning of cycle and 0.44% at end of cycle were used in the analysis.

e. Trip Reactivity Insertion

The trip reactivity insertion assumed is given in Table 15.4-2 and includes the effect of one stuck RCCA adjacent to the ejected rod. These values are reduced by the ejected rod reactivity. The shutdown reactivity was simulated by dropping a rod of the required worth into the core. The start of rod motion occurred 0.5 second after the high neutron flux trip point is reached. This delay is assumed to consist of 0.2 second for the instrument channel to produce a signal, 0.15 second for the trip breaker to open, and 0.15 second for the coil to release the rods. A curve of trip rod insertion versus time was used which assumed that insertion to the dashpot does not occur until 2.4 seconds after the start of fall. The choice of such a conservative insertion rate means that there is over one second after the trip point is reached before significant shutdown reactivity is inserted into the core. This is a particularly important conservatism for hot full power accidents.

The minimum design shutdown margin available for this plant at hot zero power (HZP) may be reached only at end of life in the equilibrium cycle. This value includes an allowance for the worst stuck rod, an adverse xenon distribution, conservative Doppler and moderator defects, and an allowance for calculational uncertainties. Physics calculations have shown that the effect of two stuck RCCAs (one of which is the worst ejected rod) is to reduce the shutdown by about an additional 1%Δk. Therefore, following a reactor trip resulting from an RCCA ejection accident, the reactor will be subcritical when the core returns to HZP.

Depressurization calculations have been performed assuming the maximum possible size break (2.75 inch diameter) located in the reactor pressure vessel head. The results show a rapid pressure drop and a decrease in system water mass due to the break. The emergency core cooling system (ECCS) is actuated on low pressurizer pressure within 1 minute after the break. The RCS pressure continues to drop and reaches saturation (1100 to 1300 psi depending on the system temperature) in about 2 to 3 minutes. Due to the large thermal inertia of primary and secondary system, there has been no significant decrease in the RCS temperature below no-load by this time, and the depressurization itself has caused an increase in shutdown margin by about 0.2%Δk due to the pressure coefficient. The cooldown transient could not absorb the available shutdown margin until more than 10 minutes after the break. The addition of borated safety injection flow starting one minute after the break is much more than sufficient to ensure that the core remains subcritical during the cooldown.

f. Reactor Protection

Reactor protection for a rod ejection is provided by high neutron flux trip (high and low setting) and high rate of neutron flux increase trip. These protection functions are part of the reactor trip system. No single failure of the reactor trip system will negate the protection functions required for the rod ejection accident, or adversely affect the consequences of the accident.

g. Results

Cases are presented for both beginning and end of life at zero and full power.

(1) Beginning of Cycle, Full Power

Control bank D was assumed to be inserted to its insertion limit. The worst ejected rod worth and hot channel factor were conservatively calculated to be 0.25%Δρ and 6.0 respectively. The maximum fuel stored energy was 164 cal/gm. The peak hot spot fuel center temperature reached melting, conservatively assumed at 4900°F. However, melting was restricted to less than 10% of the fuel pellet.

(2) Beginning of Cycle, Zero Power

For this condition, control bank D was assumed to be fully inserted and banks B and C were at their insertion limits. The worst ejected rod is located in control bank D and has a worth of 0.78%$\Delta\rho$ and a hot channel factor of 11.5. The maximum fuel stored energy was 141 cal/gm. The peak fuel center temperature was 3835°F.

(3) End of Cycle, Full Power

Control bank D was assumed to be inserted to its insertion limit. The ejected rod worth and hot channel factors were conservatively calculated to be 0.25%$\Delta\rho$ and 7.0 respectively. The maximum fuel stored energy was 164 cal/gm. The peak fuel center temperature was 4850°F.

(4) End of Cycle, Zero Power

The ejected rod worth and hot channel factor for this case were obtained assuming control bank D to be fully inserted and banks B and C at their insertion limits. The results were 0.85%$\Delta\rho$ and 26.0 respectively. The maximum fuel stored energy was 149 cal/gm. The peak fuel center temperature was 3938°F. The Doppler weighting factor for this case is significantly higher than that of the other cases due to the very large transient hot channel factor.

A summary of the cases presented above is given in Table 15.4-2. The nuclear power and hot spot fuel and clad temperature transients for the worst cases (beginning of life full power and end of life zero power) are presented in Figure 15.4-10 and Figure 15.4-11. The calculated sequence of events for these worst case rod ejection accidents is presented in Table 15.4-1. For all cases, reactor trip occurs very early in the transient, after which the nuclear power excursion is terminated. As discussed previously, the reactor will remain subcritical following reactor trip.

The ejection of an RCCA constitutes a break in the RCS, located in the reactor pressure vessel head. Following the RCCA ejection, the operator would follow the same emergency instructions as for any other LOCA to recover from the event.

h. Fission Product Release

It is assumed that fission products are released from the gaps of all rods entering DNB. In all cases considered, less than 10% of the rods entered DNB based on a generic analysis. Although limited fuel melting at the hot spot was predicted for the beginning-of-life full power case, melting is not expected since the analysis conservatively assumed that the hot spots before and after ejection were coincident.

i. Pressure Surge

A detailed calculation of the pressure surge for an ejection worth of one dollar at beginning of life, hot full power, indicates that the peak pressure does not exceed that which would cause stress to exceed the faulted condition stress limits[(10)]. Since the severity of the present analysis does not exceed the "worst case" analysis, the accident for this plant will not result in an excessive pressure rise or further damage to the RCS.

j. Lattice Deformations

A large temperature gradient will exist in the region of the hot spot. Since the fuel rods are free to move in the vertical direction, differential expansion between separate rods cannot produce distortion. However, the temperature gradients across individual rods may produce a differential expansion tending to bow the midpoint of the rods toward the hotter side of the rod. Calculations have indicated that this bowing would result in a negative reactivity effect at the hot spot since Westinghouse cores are under-moderated, and bowing will tend to increase the under moderation at the hot spot. Since the 17x17 fuel design is also under-moderated, the same effect would be observed. In practice, no significant bowing is anticipated, since the structural rigidity of the core is more than sufficient to withstand the forces produced. Boiling in the hot spot region would produce a net flow away from that region. However, the heat from the fuel is released to the water relatively slowly, and it is considered inconceivable that cross flow will be sufficient to produce significant lattice forces. Even if massive and rapid boiling, sufficient to distort the lattice, is hypothetically postulated, the large void fraction in the hot spot region would produce a reduction in the total core moderator to fuel ratio, and a large reduction in this ratio at the hot spot. The net effect would, therefore, be a negative feedback.

It can be concluded that no conceivable mechanism exists for a net positive feedback resulting from lattice deformation. In fact, a small negative feedback may result. The effect is conservatively ignored in the analysis.

15.4.8.3 Radiological Consequences using Alternate Source Term Methodology

a. Background

This event consists of the mechanical failure of a control rod mechanism pressure housing resulting in the ejection of a rod cluster control assembly and drive shaft. This event is the same as the Rod Ejection event referred to in RG 1.183. The RCCA Ejection results in a reactivity insertion that leads to a core power level increase and subsequent reactor trip. Following the reactor trip, plant cooldown is effected by steam release from the SG MSSVs/ASDVs. Two RCCA Ejection cases are considered. The first case assumes that 100% of the activity released from the damaged fuel is instantaneously and homogeneously mixed throughout the containment atmosphere. The second case assumes that 100% of the activity released from the damaged fuel is completely dissolved in the primary coolant and is available for release to the secondary system.

b. Compliance with RG 1.183 Regulatory Positions

The RCCA Ejection dose consequence analysis is consistent with the guidance provided in RG 1.183 Appendix H, "Assumptions for Evaluating the Radiological Consequences of a PWR Rod Ejection Accident," as discussed below:

1. Regulatory Position 1 – The total core inventory of the radionuclide groups utilized for determining the source term for this event is based on RG 1.183, Regulatory Position 3.1, and is provided in Table 15C-1. The inventory provided in Table 15C-1 is adjusted for the fraction of fuel damaged and a radial peaking factor of 1.65 is applied. The release fractions provided in RG 1.183 Table 3 are adjusted to comply with the specific RG 1.183 Appendix H release requirements. For both the containment and secondary release cases, the activity available for release from the fuel gap for fuel that experiences DNB is assumed to be 10% of the noble gas and iodine inventory in the DNB fuel. For the containment release case for fuel that experiences fuel centerline melt (FCM), 100% of the noble gas and 25% of the iodine inventory in the melted fuel is assumed to be released to the containment. For the secondary release case for fuel that experiences FCM, 100% of the noble gas and 50% of the iodine inventory in the melted fuel is assumed to be released to the primary coolant.

2. Regulatory Position 2 – Fuel damage is assumed for this event.

3. Regulatory Position 3 – For the containment release case, 100% of the activity released from the damaged fuel is assumed to mix instantaneously and homogeneously in the containment atmosphere. For the secondary release case, 100% of the activity released from the damaged fuel is assumed to mix instantaneously and homogeneously in the primary coolant and be available for leakage to the secondary side of the SGs.

4. Regulatory Position 4 – The chemical form of radioiodine released from the damaged fuel to the containment is assumed to be 95% cesium iodide (CsI), 4.85% elemental iodine, and 0.15% organic iodide.

5. Regulatory Position 5 – The chemical form of radioiodine released from the SGs to the environment is assumed to be 97% elemental iodine, and 3% organic iodide.

6. Regulatory Position 6.1 – For the containment leakage case, natural deposition in the containment is credited. In addition, the Secondary Containment ventilation filtration is credited. Containment spray is not credited.

7. Regulatory Position 6.2 – The containment is assumed to leak at the TS maximum allowable rate of 0.15% for the first 24 hours and 0.075% for the remainder of the event.

8. Regulatory Position 7.1 – The primary-to-secondary accident induced leakage rate is apportioned between the SGs as specified by the Technical Specification Steam Generator Program (1.0 gpm total, 500 gallons per day to any one SG).

9. Regulatory Position 7.2 – The density used in converting volumetric leak rates to mass leak rates is consistent with the basis of surveillance tests used to show compliance with the SG leak rate TS.

10. Regulatory Position 7.3 – All of the noble gas released to the secondary side is assumed to be released directly to the environment without reduction or mitigation.

11. Regulatory Position 7.4 – Compliance with Appendix E Sections 5.5 and 5.6 is discussed below:

    - Appendix E, Regulatory Position 5.5.1 – All four steam generators are used for plant cooldown. Therefore, the primary-to-secondary leakage is assumed to mix with the secondary water without flashing.

- Appendix E, Regulatory Position 5.5.2 – None of the SG tube leakage is assumed to flash for this event.
- Appendix E, Regulatory Position 5.5.3 – All of the SG tube leakage is assumed to mix with the bulk water.
- Appendix E, Regulatory Position 5.5.4 – The radioactivity within the bulk water is assumed to become vapor at a rate that is a function of the steaming rate and the partition coefficient. A partition coefficient of 100 is assumed for the iodine. The retention of particulate radionuclides in the SGs is limited by the moisture carryover from the SGs. The same partition coefficient of 100, as used for iodine, is assumed for other particulate radionuclides. This assumption is consistent with the SG carryover rate of less than 1%.
- Appendix E, Regulatory Position 5.6 – Steam generator tube bundle uncovery is not postulated for this event for Seabrook.

c. <u>Other Assumptions</u>

1. RG 1.183 does not address secondary coolant activity. This analysis assumed that the equilibrium specific activity on the secondary side of the steam generators is equal to the TS 3.7.1.4 limit of 0.1 μCi/gm Dose Equivalent I-131. This analysis also conservatively assumes that the initial secondary coolant activity includes 10% of the primary coolant equilibrium concentration of alkali metals.
2. The steam mass release rates for the SGs are provided in Table 15.4-4.
3. This evaluation assumed that the RCS mass remains constant throughout the event.
4. The SG secondary side mass in the SGs is assumed to remain constant throughout the event.
5. Steam releases from the SGs are postulated to occur from the MSSV or ASDV with the most limiting atmospheric dispersion factors. For the RCCA Ejection inside of containment release case, releases are assumed to leak out of the containment via the same containment release points as used for the LOCA.

d. Methodology

Input assumptions used in the dose consequence analysis of the RCCA Ejection are provided in Table 15.4-3. The postulated accident consists of two cases. One case assumes that 100% of the activity released from the damaged fuel is instantaneously and homogeneously mixed throughout the containment atmosphere, and the second case assumes that 100% of the activity released from the damaged fuel is completely dissolved in the primary coolant and is available for release to the secondary system.

For the containment release case, 100% of the activity is released instantaneously to the containment. The releases from the containment correspond to the same leakage points as used for the LOCA. Natural deposition of the released activity inside of containment is credited. In addition, the secondary containment building ventilation and filtration system is credited. Removal of activity via containment spray is not credited.

For the secondary release case, primary coolant activity is released into the SGs by leakage across the SG tubes. The activity on the secondary side is then released via steaming from the SG MSSVs/ASDVs until the decay heat generated in the reactor core can be removed by the Shutdown Cooling (SDC) system 8 hours into the event. Additional activity, based on the secondary coolant initial iodine concentration is assumed to be equal to the maximum value of 0.1 μCi/gm DE I-131 permitted by TS 3.7.1.4. Activity is released to the environment from the steam generator as a result of the postulated primary-to-secondary accident induced leakage and the postulated activity levels of the primary and secondary coolants, until the steam generator steam release is terminated (at 8 hours for SDC initiation). These release assumptions are consistent with the requirements of RG 1.183.

The RCCA Ejection is evaluated with the assumption that 0.375% of the fuel experiences FCM and 15.0% of the fuel experiences DNB. The activity released from the damaged fuel corresponds to the requirements set out in Regulatory Position 1 of Appendix H to RG 1.183. A radial peaking factor of 1.65 is applied in the development of the source terms.

For this event, the Control Room ventilation system modes of operation are:

- Initially the ventilation system is assumed to be operating in normal mode. The air flow distribution during this mode is 1000 cfm of unfiltered fresh air make up and an assumed value of 150 cfm of unfiltered inleakage assumed for the secondary side release or 190 cfm unfiltered inleakage assumed for the primary release.

- After the start of the event, the Control Room is isolated due to a high radiation reading in the Control Room ventilation system. A 30-second delay is applied to account for diesel generator start time, damper actuation time, instrument delay, and detector response time. In this emergency mode, the air flow distribution consists of 600 cfm of filtered makeup flow from the outside, 150 cfm of unfiltered inleakage assumed for the secondary side release or 190 cfm unfiltered inleakage assumed for the primary release and 390 cfm of filtered recirculation flow.
- The Control Room ventilation filter efficiencies that are applied to the filtered makeup and recirculation flows are 99% for particulate, 95% for elemental iodine, and 95% for organic iodine.

e. Radiological Consequences

The atmospheric dispersion factors (*X/Qs*) used for this event for the Control Room dose are based on the postulated release locations and the pathway into the control room. These *X/Qs* are summarized in Table 2R-2 and Table 2R-3.

For the RCCA secondary side release case, releases from the SGs are assumed to occur from the MSSV/ASDV that produces the most limiting *X/Qs*. For the RCCA Ejection containment release case, the *X/Qs* for containment leakage are assumed to be identical to those for the LOCA.

For the EAB and the LPZ dose the *X/Q* factors are provided in Appendix 2Q.

The radiological consequences of the RCCA Ejection are analyzed using the RADTRAD-NAI code and the inputs/assumptions previously discussed. As shown in Table 15.4-5, the results of both cases for EAB dose, LPZ dose, and Control Room dose are all within the appropriate regulatory acceptance criteria.

15.4.8.4 Conclusions

Even on a pessimistic basis, the analyses indicate that the described fuel and clad limits are not exceeded. It is concluded that there is no danger of sudden fuel dispersal into the coolant. Since the peak pressure does not exceed that which would cause stresses to exceed the faulted condition stress limits, it is concluded that there is no danger of further consequential damage to the RCS. The analyses have demonstrated the fission product release as a result of fuel rods entering DNB is limited to less than 10% of the fuel rods in the core.

15.4.9 Spectrum of Rod Drop Accidents in a BWR

Not applicable to Seabrook.

15.4.10 References

1. WCAP-7907-P-A, "LOFTRAN Code Description," T. W. T. Burnett, et al., April 1984
2. WCAP-7908-A, "FACTRAN - A FORTRAN-IV Code for Thermal Transients in a UO_2 Fuel Rod," H. G. Hargrove, December 1989
3. WCAP-7979-A, "TWINKLE - A Multi-Dimensional Neutron Kinetics Computer Code," D. H. Risher, and R. F. Barry, January 1975
4. WCAP-14565, "VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis," Y. X. Sung, et al., April 1997
5. WCAP-11397-P-A, "Revised Thermal Design Procedure," A. J. Friedland and S. Ray, April 1984
6. WCAP-9226-P-A. "Reactor Core Response to Excessive Secondary Steam Releases," S. D. Hollingsworth, et al., January 1998
7. WCAP-9230, "Report on the Consequences of a Postulated Main Feedline Rupture," G. E. Lang and J. P. Cunningham, January 1978
8. WCAP-11394-P-A, "Methodology for the Analysis of the Dropped Rod Event," R. L. Haessler, et al., January 1990
9. WCAP-8424, Revision 1, "An Evaluation of Loss of Flow Accidents Caused by Power System Frequency Transients in Westinghouse PWRs," M. S. Baldwin, et al., June 1975
10. WCAP-7588, Revision 1-A, "An Evaluation of the Rod Ejection Accident in Westinghouse Pressurized Water Reactors Using Spatial Kinetics Methods," D. H. Risher, January 1975
11. IN-1370, "Annual Report - SPERT Project, October 1968 - September 1969 Edition (Idaho Nuclear Corporation)," T. G. Taxelius, June 1970
12. ANL-7225, "Studies in TREAT of Zircaloy-2-Clad, UO_2-Core Simulated Fuel Elements," R. C. Liimantaninen and F. J. Testa, p. 177, November 1966
13. Letter from W. J. Johnson (Westinghouse) to R. C. Jones (USNRC), "Use of 2700°F PCT Acceptance Limit in Non-LOCA Accidents," NS-NRC-89-3466, October 1989

14. ANSI/ANS-5.1-1979, "American National Standard for Decay Heat Power in Light Water Reactors," August 1979Light Water Reactors," August 1979

15. WCAP-14882-P-A, "RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analyses," D.S. Huegel, et al., April 1999

15.5 INCREASE IN REACTOR COOLANT INVENTORY

Discussion and analysis of the following events is presented in this section:

a. Inadvertent operation of Emergency Core Cooling System during power operation.

b. Chemical and volume control system malfunction that increases reactor coolant inventory.

These events, considered to be ANS Condition II, cause an increase in reactor coolant inventory. Subsection 15.0.1 contains a discussion of ANS classifications.

15.5.1 Inadvertent Operation of Emergency Core Cooling System during Power Operation

15.5.1.1 Identification of Causes and Accident Description

Spurious Emergency Core Cooling System (ECCS) operation at power could be caused by operator error or a false electrical actuation signal. A spurious signal may originate from any of the safety injection actuation channels as described in Section 7.3.

Following the actuation signal, the suction of the coolant charging pumps is diverted from the volume control tank to the refueling water storage tank. The charging pumps then force highly concentrated (2600 ppm) boric acid solution into the cold leg of each loop. The safety injection pumps also start automatically but provide no flow when the Reactor Coolant System (RCS) is at normal RCS pressure.

A Safety Injection System (SIS) signal normally results in a reactor trip followed by a turbine trip. However, it cannot be assumed that any single fault that actuates the SIS will also produce a reactor trip. If a reactor trip is generated by the spurious SIS signal, the operator should determine if the spurious signal was transient or steady-state in nature. The operator must also determine if the safety injection signal should be blocked. For a spurious occurrence, the operator would stop the safety injection and maintain the plant in the hot shutdown condition. If the ECCS actuation instrumentation must be repaired, future plant operation will be in accordance with the Technical Specifications.

If the Reactor Protection System does not produce an immediate trip as a result of the spurious SIS signal, the reactor experiences a negative reactivity excursion due to the injected boron causing a decrease in reactor power. The power mismatch causes a drop in T_{avg} and consequent coolant shrinkage, pressurizer pressure and water level drop. Load will decrease due to the effect of reduced steam pressure on load after the turbine throttle valve is fully open. If automatic rod control is used, these effects will be lessened until the rods have moved out of the core. The transient is eventually terminated by the reactor protection system low pressure trip or by manual trip.

The time of trip is affected by initial operating conditions including core burnup history which affects initial boron concentration, rate of change of boron concentration, Doppler and moderator coefficients.

Recovery from this second case is made in the same manner as described for the case where the SIS signal results directly in a reactor trip. The only difference is the lower T_{avg} and pressure associated with the power mismatch during the transient. Since the negative reactivity from the injected boron causes reactor power to decrease, the time at which reactor trip occurs has little effect on DNBR.

A second issue associated with this event is the possibility of the pressurizer overfilling, especially a possible condition where the pressurizer is water-solid and its pressure reaches the setpoint of the pressurizer safety relief valves. In this condition, water would pass through these safety relief valves, which could damage the valves and challenge the ability to ensure the RCS boundary can be isolated. The analysis focuses on the pressurizer filling aspects of the event and demonstrates that the pressurizer does not become water-solid, and therefore, there is no water flow through the PORV's or pressurizer safety valves.

This event is classified as a Condition II incident (an incident of moderate frequency) as defined in Subsection 15.0.1.

15.5.1.2 Analysis of Effects and Consequences

a. Method of Analysis

Because of the power and temperature reduction during the transient, operating conditions do not approach the core limits for DNBR. The spurious operation of the ECCS is analyzed by employing the RETRAN computer code. The RETRAN computer code is discussed in UFSAR Section 15.0.11.

The analysis employs several plant parameters at maximum and minimum values to maximize the rate for pressurizer filling. For example, the analysis employs maximum reactivity feedback, maximum initial pressurizer water level, and the minimum initial reactor coolant temperature to maximize the rate of pressurizer filling.

The assumptions are as follows:

1. Initial Operating Conditions

   The impact of the full power RCS Tavg window was considered. The upper end of the Tavg window was determined to be more limiting. The higher corresponding pressurizer level turned out to be more limiting than the benefit gained by the lower initial mass.

   The impact of the feedwater temperature window was also analyzed and the upper end of the feedwater temperature window was determined to be slightly more limiting.

   Initial reactor power is assumed to be at the maximum value, initial reactor coolant temperature is assumed to be at the minimum value, initial pressurizer pressure is assumed to be at its minimum value, and the initial pressurizer water level is assumed to be at its maximum value, consistent with steady-state full power operation including allowances for calibration and instrument errors.

2. Moderator and Doppler Coefficients of Reactivity

   A most-negative moderator temperature coefficient was used. A high (absolute value) Doppler power coefficient was assumed.

3. Reactor Control

   The reactor was assumed to be in manual control.

4. Pressurizer Control

   Pressurizer heaters and spray are assumed to be operable to increase the rate of pressurizer filling. The pressurizer sprays act to reduce the RCS pressure, thus increasing ECCS injection. The pressurizer heaters act to add energy to the pressurizer fluid, thus increasing the pressurizer fluid volume through thermal expansion.

5. Boron Injection

   At time zero, two charging pumps conservatively inject 2,600 ppm borated water into the cold leg of each loop.

6. Reactor Trip

   The reactor and turbine are assumed to trip upon receipt of the SI signal. Assuming reactor and turbine trip on SI minimizes the heat removal capability of the RCS, thereby maximizing the RCS inventory increase through SI flow and thermal expansion of the RCS fluid.

7. Pressurizer PORVs

   No credit is taken for any pressurizer PORV operation.

Plant systems and equipment, which are available to mitigate the effects of the accident, are discussed in Subsection 15.0.8 and listed in Table 15.0-5. No single active failure in any of these systems or equipment will adversely affect the consequences of the accident. Safety injection termination is defined as the stopping of all mass injection into the RCS.

b. Results

Figure 15.5-1 shows the transient response to inadvertent operation of the ECCS during power operation. The calculated sequence of events is shown on Table 15.5-1.

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 transient to rapidly turn around. Pressurizer water level then increases throughout the transient. Per Emergency Operating Procedures, RCS temperature (T_{avg}) is maintained at 557°F through heat removal from the steam generators using the steam generator atmospheric steam dump valves, and flow from all but one centrifugal charging pump is terminated early in the event. The analysis credits heat removal through the steam generators using the atmospheric steam dump valves, stopping of all but one centrifugal charging pump at 9 minutes after the beginning of the event, and termination of all charging flow at 13 minutes into the event based on analyzed maximum fill rate. The results of the revised analysis indicate that at no time does the pressurizer become water-solid.

15.5.1.3 Radiological Consequences

No radioactivity releases are anticipated as a direct result of this malfunction. Consequently, no radiological consequences are predicted.

15.5.1.4 Conclusions

Results of the analysis of the inadvertent ECCS initiation at power event demonstrate that there is no hazard to the integrity of the RCS. The approach to terminating this event is consistent with Option II of Westinghouse NSAL 93-013.

For this event, the DNBR is never less than the initial value. Thus, there will be no cladding damage and no release of fission products to the RCS.

Operator action terminating safety injection flow is sufficient to preclude a pressurizer water-solid condition and prevent actuation of the pressurizer PORVs and safety valves. By demonstrating that sufficient time is available for the appropriate operator actions to preclude a pressurizer water-solid condition, the pressurizer valve integrity can be maintained for the inadvertent ECCS initiation at power event. No credit for operation of the pressurizer PORVs is assumed. Therefore, the ability to isolate the RCS and maintain the integrity of the RCS pressure boundary confirms that this event does not lead to a more serious plant condition, hence demonstrating acceptability of the Condition II acceptance criteria.

15.5.2 Chemical and Volume Control System Malfunction that Increases Reactor Coolant Inventory

Transients due to CVCS malfunctions that increase the reactor coolant inventory can be divided into three categories:

Category 1	CVCS malfunctions that result in the injection of water with a boron concentration greater than the RCS boron concentration.
Category 2	CVCS malfunctions that result in the injection of water with a boron concentration less than the RCS boron concentration.
Category 3	CVCS malfunctions that result in the injection of water with a boron concentration equal to the RCS boron concentration.

There are two possible criteria for evaluating these transients: core integrity and overfilling of the pressurizer. Transients of the type listed in Category 1 are bounded by the "inadvertent operation of emergency core cooling system analysis" presented in Subsection 15.5.1. Transients of the type listed in Category 2 are bounded by the "CVCS malfunction that results in a decrease in boron concentration in the reactor coolant" presented in Subsection 15.4.6.

CVCS malfunctions of the type described under Category 3 will not result in any significant nuclear power or RCS temperature transient; this type of transient may result in filling the pressurizer. An analysis of the CVCS malfunction that results in injection of water with a boron concentration equal to the RCS boron concentration are presented in this section.

<u>CVCS Malfunctions that Result in the Injection of Water with a Boron Concentration Equal to the RCS Boron Concentration</u>

a. <u>Identification of Causes and Accident Description</u>

The most limiting case would result if charging was in automatic control and the pressurizer level channel being used for charging control failed in a low direction. This would cause maximum charging flow to be delivered to the RCS and letdown flow would be isolated. The worst single failure for this event would be another pressurizer level channel failing in an as is condition or a low condition. This will defeat the reactor trip on 2 out of 3 high pressurizer level channels. To prevent filling the pressurizer, the operator must be relied upon to terminate charging. This event is classified as a Condition II incident (an incident of moderate frequency).

b. <u>Analysis of Effects and Consequences</u>

The CVCS malfunction is analyzed by employing the detailed digital computer program RETRAN[1]. The code simulates the neutron kinetics, RCS, pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generator, steam generator safety valves, and the ECCS. The program computes pertinent plant variables, including temperatures, pressures, and power level.

The assumptions incorporated in the analyses were as follows:

1. Initial Operating Conditions

Pressurizer pressure is assumed to be at its minimum value. Pressurizer water level is assumed to be at the high end of the range of the values consistent with its programmed level. The initial reactor power and RCS temperature are at their full power values with uncertainties.

The impact of the full power RCS T_{avg} window was considered. The upper end of the T_{avg} window was determined to be more limiting. The higher corresponding pressurizer level turned out to be more limiting than the benefit gained by the lower initial mass.

The impact of the feedwater temperature window was also analyzed and the upper end of the feedwater temperature window was determined to be slightly more limiting.

2. Reactivity Coefficients

   Maximum reactivity feedback case

   The most negative moderator temperature coefficient and a most negative Doppler coefficient.

3. Reactor Control

   Both manual and automatic control have been analyzed.

4. Charging System

   Maximum charging system flow based on RCS back pressure from one centrifugal pump is delivered to the RCS.

5. Reactor Trips

   The transient is initiated by the pressurizer level channel which is used for control purpose failing low. As a worst single failure, another pressurizer level channel fails low, defeating the two out of three high pressurizer level trip. No reactor trips are used.

c. <u>Results</u>

Figure 15.5-2 shows the transient response due to the charging system malfunction. In all the cases analyzed, core power and RCS average temperature remain relatively unchanged.

The calculated sequence of events is shown in Table 15.5-2.

d. <u>Conclusions</u>

The sequence of events presented in Table 15.5-2 shows that the operator has sufficient time to take corrective action.

15.5.3 References

1. WCAP-14882-P-A, "RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analyses," D.S. Huegel, et al., April 1999.

2. Westinghouse Nuclear Safety Advisory Letter NSAL-93-013, "Inadvertent ECCS Actuation at Power," dated June 30, 1993.

15.6 DECREASE IN REACTOR COOLANT INVENTORY

Events that result in a decrease in reactor coolant inventory, as discussed in this section, are as follows:

a. Inadvertent opening of a pressurizer safety or relief valve

b. Break in an instrument line or other lines from the reactor coolant pressure boundary that penetrate Containment

c. Steam generator tube failure

d. Loss-of-coolant accident resulting from a spectrum of postulated piping breaks within the reactor coolant pressure boundary.

15.6.1 Inadvertent Opening of a Pressurizer Safety or Relief Valve

15.6.1.1 Identification of Causes and Accident Description

An accidental depressurization of the Reactor Coolant System (RCS) could occur as a result of an inadvertent opening of a pressurizer relief or safety valve. Since a safety valve is sized to relieve approximately twice the steam flow rate of a relief valve, and will therefore allow a much more rapid depressurization upon opening, the most severe core conditions resulting from an accidental depressurization of the RCS are associated with an inadvertent opening of a pressurizer safety valve. Initially the event results in a rapidly decreasing RCS pressure which could reach the hot leg saturation pressure if a reactor trip did not occur. The pressure continues to decrease throughout the transient. The effect of the pressure decrease is to decrease power via the moderator density feedback (positive MTC), but the reactor control system (if in the automatic mode) functions to maintain the power and average coolant temperature until reactor trip occurs. Pressurizer level increases initially due to expansion caused by depressurization and then decreases following reactor trip.

The reactor may be tripped by the following reactor protection system signals:

a. Overtemperature ΔT

b. Pressurizer low pressure.

An inadvertent opening of a pressurizer safety or relief valve is classified as an ANS Condition II event, a fault of moderate frequency. See Subsection 15.0.1 for a discussion of Condition II events.

15.6.1.2 Analysis of Effects and Consequences

a. Method of Analysis

The accidental depressurization transient is analyzed by employing the detailed digital computer code RETRAN (Reference 4). The code simulates the neutron kinetics, RCS, pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generator, and steam generator safety valves. The code computes pertinent plant variables including temperatures, pressures, and power level.

This accident is analyzed with the revised thermal design procedure described in WCAP-11397 (Reference 36).

In order to give conservative results in calculating the Departure from Nucleate Boiling Ratio (DNBR) during the transient, the following assumptions are made:

1. Initial reactor power, pressure, and RCS temperature are assumed to be at their nominal values. Uncertainties in initial conditions are included in the limit DNBR as described in WCAP-11397 (Reference 36).
2. The most positive MTC is assumed.
3. The least negative Doppler coefficient of reactivity is assumed such to maximize power increase prior to the reactor trip.
4. The pressurizer safety valve flowrate is assumed to be 120% of the design capacity of the valve.

Plant systems and equipment which are necessary to mitigate the effects of RCS depressurization caused by an inadvertent safety valve opening are discussed in Subsection 15.0.8 and listed in Table 15.0-5.

Normal reactor control systems are not required to function. The rod control system is assumed to be in the manual mode in order to prevent rod insertion due to an increase in RCS temperature prior to reactor trip. The reactor protection system functions to trip the reactor on the appropriate signal. No single active failure will prevent the reactor protection system from functioning properly.

b. Results

The system response to an inadvertent opening of a pressurizer safety valve is shown in Figure 15.6-1 and Figure 15.6-2. Figure 15.6-1 illustrates the nuclear power transient following the depressurization. Nuclear power increases slowly from the initial value until reactor trip occurs on overtemperature ΔT. The pressure transient and average coolant temperature transient following the accident are given in Figure 15.6-2. The DNBR decreases initially, but increases rapidly following the trip, as shown in Figure 15.6-1. The DNBR remains above the limit value throughout the transient.

The calculated sequence of events for the inadvertent opening of a pressurizer safety valve incident is shown on Table 15.6-1.

15.6.1.3 Radiological Consequences

No radiological consequences have been calculated for this postulated accident since no fuel or clad damage is predicted.

15.6.1.4 Conclusions

The results of the analysis show that the Low Pressurizer Pressure and the Overtemperature ΔT reactor protection system signals provide adequate protection against the RCS depressurization event. No fuel or clad damage is predicted for this accident.

15.6.2 Failure of Small Lines Carrying Primary Coolant Outside Containment

15.6.2.1 Identification of Causes and Accident Description

The sample lines from the hot legs of reactor coolant loops 1 and 3, and from the steam and liquid space of the pressurizer, and the chemical and volume control system (CVCS) letdown and excess letdown lines penetrate the Containment. The sample lines are provided with normally closed isolation valves on both sides of the containment wall, as sampling requirements dictate only intermittent daily use. The CVCS letdown line is provided with normally open containment isolation valves on both sides of the containment wall that are designed in accordance with the requirements of General Design Criteria 55. The excess letdown line is normally isolated and is also provided with two normally open containment isolation valves. The temperature of this fluid leaving the Containment is a maximum of 380°F.

The most severe pipe rupture with regard to radioactivity release during normal power operation is a complete severance of the 3-inch letdown line outside the Containment between the outboard containment isolation valve and the letdown heat exchanger (see Figure 9.3-26, Figure 9.3-27, Figure 9.3-28, Figure 9.3-29 and Figure 9.3-31).

15.6.2.2 Analysis of Effects and Consequences

The occurrence of a complete severance of the letdown line would result in a loss-of-reactor coolant at the rate of about 140 gpm (referenced to a density of 62 lb/ft^3). This release rate would not result in actuation of any Engineered Safety Features Systems. Area radiation and leakage detection instrumentation provide the primary means for detection of a letdown line rupture (see Subsection 5.2.5). Frequent operation of the CVCS Reactor Makeup Control System and the other CVCS instrumentation will aid the operator in diagnosing a letdown line rupture.

The time required for the operator to identify the accident and isolate the rupture is expected to be within 30 minutes of the rupture. Once the rupture is identified, the operator would isolate the letdown line rupture by closing the high pressure letdown valves, followed by closing the pressurizer low level isolation valves. Alternatively, the operator could close the letdown line containment isolation valves to isolate the rupture. All valves are provided with control switches at the main control board. There are no single failures that would prevent isolation of the letdown line rupture.

15.6.2.3 Radiological Consequences using Alternate Source Term Methodology

a. Background

This event is a rupture of a primary coolant letdown line outside of containment. Since RG 1.183 does not provide specific guidance to the analysis of this type of event, the general guidance of the Regulatory Guide will be supplemented with guidance of the Standard Review Plan (SRP) section 15.6.2 and consideration of the current licensing basis for this event. In accordance with SRP 15.6.2, the source term for this calculation will assume an accident-generated or concurrent iodine spike. A reactor trip is not predicted for this event. The dose assessment for this event is comprised of two separate release paths. Path 1 defines the leakage from the double ended rupture of the Letdown line in the Plant Auxiliary Building (PAB) outside of containment. Path 2 defines the release of activity through the secondary side steam release from the condenser.

b. Compliance with RG 1.183 Regulatory Positions

Since Regulatory Guide 1.183 does not provide any direct guidance regarding analysis of a Letdown Line Rupture, Standard Review Plan (SRP) Section 15.6.2 is used as the primary source of guidance for this analysis. In accordance with SRP 15.6.2, this analysis assumes an accident-generated or concurrent iodine spike in combination with the maximum leakage of primary fluid through the SG tubes into the secondary side. The RG 1.183 guidance provided for other events is applied to this event as applicable and appropriate.

The revised Letdown Line Rupture event dose consequence analysis is consistent with the guidance provided in RG 1.183, as discussed below:

1. Regulatory Position 2.2 of Appendix E – This guidance is used to define the concurrent iodine spike of 500 times the release rate corresponding to the iodine concentration at the equilibrium value (1.0 μCi/gm DE I-131).
2. Regulatory Position 3 of Appendix E – The activity released from the fuel is assumed to be released instantaneously and homogeneously through the primary coolant.

3. Regulatory Position 4 of Appendix E – The chemical form of radioiodine released from the fuel is assumed to be 95% cesium iodide (CsI), 4.85% elemental iodine, and 0.15% organic iodide. Iodine releases from the faulted SG and the unaffected SG to the environment (or containment) are assumed to be 97% elemental and 3% organic.

4. Regulatory Position 5.1 of Appendix E – The SGs are modeled as a single component with all SG tube leakage modeled into that component.

5. Regulatory Position 5.2 of Appendix E – The density used is converting volumetric leak rates to mass leak rates is consistent with the basis of surveillance tests used to show compliance with the SG leak rate TS. For the intact Steam Generators, the primary to secondary leak rate is based on a density of 1.0 gm/cc (cold liquid).

6. Regulatory Position 5.3 of Appendix E – Since a reactor trip is not predicted for this event, the primary-to-secondary leak rate is assumed to continue throughout the 30 day duration of the analysis.

7. Regulatory Position 5.4 of Appendix E – All noble gas radionuclides released from the primary system are assumed to be released to the environment without reduction or mitigation. All of the noble gas released from the primary system to the SGs is assumed to be released directly to the environment.

8. Regulatory Position 5.5.1 of Appendix E – For the steam generators used for plant cooldown, the primary-to-secondary leakage is assumed to mix with the secondary water without flashing.

9. Regulatory Position 5.5.4 of Appendix E – It is conservatively assumed that the decontamination prescribed for the SGs in Regulatory Guide 1.183 is not applicable to the SGs under power operation. Therefore, no partition factor is applied to the activity as it is transferred from the SG to the turbine. Consistent with the pre-trip treatment of the secondary steam release during the current Steam Generator Tube Rupture at Seabrook, an iodine Decontamination Fraction of 99% will be assigned for the release from the condenser. It is similarly reasonable to assume that the 99% is equally applicable to all particulate released from the condenser. Therefore, the SG tube leakage will be modeled as a release from the RCS to the environment at the condenser location with a 99% filter efficiency for all particulates, and elemental and organic iodine.

10. Regulatory Position 5.6 of Appendix E – Steam generator tube bundle uncovery is not postulated for the SG's for Seabrook.

c. Other Assumptions

1. RG 1.183 does not address secondary coolant activity. This analysis assumes that the equilibrium specific activity on the secondary side of the steam generators is equal to the TS 3.7.1.4 limit of 0.1 μCi/gm Dose Equivalent I-131.

2. For a Letdown Line Rupture event outside of containment, releases from the faulted line are postulated to occur from the Primary Auxiliary Building at the location with the most limiting atmospheric dispersion factors. Releases from the secondary side are postulated to occur from the condenser.

d. Methodology

The dose assessment for this event is comprised of two separate release paths. Path 1 defines the leakage from the double ended rupture of the letdown line in the Primary Auxiliary Building outside of containment with a direct release to the environment. Path 2 defines the release of RCS tube leakage through the secondary side via steam release through the condenser. Since RG 1.183 does not provide any direct guidance regarding analysis of a Letdown Line Rupture, Standard Review Plan (SRP) Section 15.6.2 is used as the primary source of guidance for this analysis. In accordance with SRP 15.6.2, this analysis assumes an accident-generated or concurrent iodine spike in combination with the maximum leakage of primary fluid through the SG tubes into the secondary side.

The accident generated appearance rate for the concurrent iodine spike is computed using the input in Table 15.6-3, with a 500 times multiplier on the normal appearance rate. The modeling of this spike is identical to that modeled for the MSLB concurrent spike case.

The Letdown Line Rupture flow rate is modeled as 140 gpm (at 62 lb_m/ft^3) for 30 minutes with a flashing fraction of 0.1815 as computed using the RG 1.183 guidance from position 5.4 of Appendix A for ECCS leakage for leakage at 380°F and 2235 psia. All of the noble gas in the letdown line rupture flow is released to the environment and the non-noble gas activity in the 0.1815 flashing fraction is assumed to be released (consistent with SRP 15.6.2 guidance).

For this event, the Control Room ventilation system modes of operation are:

- Initially the ventilation system is assumed to be operating in normal mode. The air flow distribution during this mode is 1000 cfm of unfiltered fresh air make up and an assumed value of 300 cfm of unfiltered inleakage.

- After the start of the event, the Control Room is isolated due to a high radiation reading in the Control Room ventilation system. A 30-second delay is applied to account for diesel generator start time, damper actuation time, instrument delay, and detector response time. In this emergency mode, the air flow distribution consists of 600 cfm of filtered makeup flow from the outside, 300 cfm of unfiltered inleakage, and 390 cfm of filtered recirculation flow.
- The Control Room ventilation filter efficiencies that are applied to the filtered makeup and recirculation flows are 99% for particulate, 95% for elemental iodine, and 95% for organic iodine.

e. <u>Radiological Consequences</u>

The atmospheric dispersion factors (*X/Qs*) used for this event for the Control Room dose are based on the postulated release locations and the pathway into the control room. These *X/Qs* are summarized in Table 2R-2 and Table 2R-3.

For the secondary side release case, releases from the SGs are assumed to occur from the condenser.

For the EAB and the LPZ dose the *X/Q* factors are provided in Appendix 2Q.

Reg. Guide 1.183 does not provide any direct guidance for the acceptance criteria for this event. However, the SRP states that the acceptance criteria is "a small fraction" of the 10 CFR 100 values which is further described as 10% of the limit. In applying the AST methodology to the letdown line break that same 10% interpretation is applied to the 10 CFR part 50.67 limits for the LPZ and EAB dose. The acceptable dose limit for the Control Room (CR) is that specified in 10 CFR 50.67. For a Letdown Line Rupture, these are interpreted as:

Area	Dose Criteria	
EAB	2.5 rem TEDE	(for the worst two hour period)
LPZ	2.5 rem TEDE	(for 30 days)
Control Room	5 rem TEDE	(for 30 days)

The radiological consequences of the Letdown Line Rupture event are analyzed using the RADTRAD-NAI code and the inputs and assumptions previously discussed. As shown in Table 15.6-5, the radiological consequences of the Letdown Line Rupture event are all within the appropriate acceptance criteria.

15.6.2.4 Conclusion

The doses which have been calculated for the accident of a small line break outside the Containment are within regulatory limits.

15.6.3 Steam Generator Tube Rupture

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 ANS Condition IV event, a limiting fault (see Subsection 15.0.1). The accident is assumed to take place at power with the reactor coolant activity corresponding to continuous operation with a limited amount of defective fuel rods. The accident leads to an increase in activity in the secondary system due to leakage of radioactive coolant from the Reactor Coolant System. In the event of a coincident loss of offsite power, or failure of the condenser dump system, discharge of activity to the atmosphere takes place via the steam generator safety and/or power-operated relief valves.

Since the steam generator tube material is Inconel, a highly ductile material, the assumption of a complete severance is conservative. 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 unit operation.

The operator will determine that a steam generator tube rupture has occurred, and will identify and isolate the ruptured steam generator to minimize contamination of the secondary system and ensure termination of radioactive release to the atmosphere from the ruptured unit. 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, (References 38, 39). Sufficient indications and controls are provided so the operator can carry out these functions satisfactorily.

Assuming normal operation of the various plant control systems, 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 there is a steam flow/feedwater flow mismatch before trip as feedwater flow to the affected steam generator is reduced due to the additional break flow which is now being supplied to that unit.

b. Continued loss of reactor coolant inventory leads to a reactor trip signal generated by Overtemperature ΔT, low pressurizer pressure, or manual operator action. Resultant plant cooldown following reactor trip leads to a rapid change of pressurizer level, and the safety injection signal, initiated manually by the operator or automatically by low pressurizer pressure, follows soon after the reactor trip. The safety injection signal automatically terminates normal feedwater supply and initiates emergency feedwater addition.

c. The steam generator blowdown liquid monitor and the condenser off-gas radiation monitor will alarm, indicating a sharp increase in radioactivity in the secondary system.

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 and atmosphere. In the event of a coincident station blackout, the steam dump valves would automatically close to protect the condenser. The steam generator pressure would rapidly increase resulting in steam discharge to the atmosphere (when the pressure reaches to the setpoint) through the steam generator safety and/or power-operated relief valves.

e. Following reactor trip, the continued action of emergency feedwater supply and borated safety injection flow (supplied from the refueling water storage tank) provide a heat sink which absorbs some of the decay heat. Thus, steam bypass to the condenser, or in the case of loss of offsite power, steam relief to the atmosphere, is attenuated during the time interval in which the recovery procedure leading to identification of the ruptured steam generator is being carried out.

f. Safety injection flow results in increasing pressurizer water level. The time after trip at which the operator can clearly see returning level in the pressurizer is dependent upon the amount of operating auxiliary equipment.

The sequence of events for the steam generator tube rupture thermal and hydraulics analysis is given in Table 15.6-7.

The time dependent parameters for the Steam Generator Tube Rupture thermal and hydraulics analysis which is bounded by the radiological analysis presented in Section 15.6.3.3 are listed in the following figures:

Figure 15.6-50 - Pressurizer Pressure

Figure 15.6-51 - Reactor Coolant System Temperature

Figure 15.6-52 - Steam Generator Pressure (Ruptured Steam Generator)

Figure 15.6-53 - Primary Coolant Flashing (Ruptured Steam Generator)

Figure 15.6-54 - Pressurizer Water Level

Figure 15.6-55 - Steam Flow Rate (Ruptured Steam Generator)

Figure 15.6-56 - Feedwater Flow to Ruptured Steam Generator

Figure 15.6-57 - Ruptured Steam Generator Steam Flow Rate to Atmosphere

Figure 15.6-58 - Ruptured Steam Generator Break Flow Rate

Figure 15.6-59 - Steam Generator Mass

Figure 15.6-60 - Ruptured Steam Generator Liquid Volume

15.6.3.2 Analysis of Effects and Consequences

a. Method of Analysis

Two scenarios are considered, one leading to minimum margin to overfill of the ruptured steam generator, and the other to maximum radiological consequences.

In estimating the mass transfer from the Reactor Coolant System through the broken tube for the scenario with maximum radiological consequences the following assumptions are made:

1. Reactor trip occurs automatically as a result of an OTDT signal.
2. Following the initiation of the safety injection signal, two centrifugal, high head safety injection and two charging pumps are actuated and continue to deliver flow until the emergency instructions for a tube rupture accident indicate that the operator should switch off all but one pump when he has identified the accident and has pressurizer level indication. The analysis considers high head safety injection and charging pumps.
3. The power-operated relief valve on the main steam line from the ruptured steam generator fails in the full open position during the initial attempt by the operator to isolate steam flow from the ruptured steam generator.
4. The operators identify the open power-operated relief valve and manually isolate it by locally closing the upstream block valve within 20 minutes of the initial attempt to close the valve. The implementation of further recovery procedure actions is delayed until the power-operated relief valve on the ruptured steam generator has been isolated.
5. The break flow is terminated by cooldown of the reactor coolant system opening the power-operated relief valves on the intact steam generators, reducing reactor coolant system pressure to a pressure below the pressure of the ruptured steam generator at the end of the cooldown by opening one of the power-operated relief valves on the pressurizer, and stopping safety injection flow.

The assumptions for the overfill scenario are similar except that no degradation of the high head safety injection or charging pumps is assumed. A power-operated relief valve on a main steam line from one of the intact steam generators is assumed to fail to open on demand, reducing the rate of reactor coolant system cooldown. The power-operated relief valve on the main steam line from the ruptured steam generator is assumed to function normally and close upon operator demand when the operators attempt to isolate steam flow from the ruptured steam generator.

b. Recovery Procedure

Symptoms of a tube rupture such as falling pressurizer pressure and level and increased charging pump flow are also symptoms of small steamline breaks and loss-of-coolant accidents. It is therefore important to determine that the accident is a rupture of a steam generator tube to carry out the correct recovery procedure. The accident under discussion can be identified by the following method. In the event of a complete tube rupture, the level in one steam generator will rise more rapidly than in the others. This is a unique indication of a tube rupture accident. Also, this accident could be identified by a steam generator blowdown radiation alarm. The recovery procedure includes isolation of the ruptured steam generator and unit cooldown.

After the Residual Heat Removal System is placed in operation, the condensate accumulated in the secondary system can be analyzed and processed as required.

There is ample time available to carry out the above recovery procedures so that isolation of the affected steam generator is established before water level rises into the main steam pipes. The available time scale is improved by the termination of emergency feedwater flow to the ruptured steam generator and the regulation of pressurizer water level with only one charging pump operating. Normal operator vigilance therefore assures that excessive water level will not be attained.

c. Results

The results of the scenario leading to minimum margin to overfill of the ruptured steam generator show that operator implementation of the steam generator tube rupture recovery procedures results in termination of the break flow before water level in the ruptured steam generator rises into the main steam pipes. The results of the radiological consequences analysis are described in Section 15.6.3.3 below. The thermal hydraulic results of this accident are less severe than that for a LOCA small break (see Subsection 15.6.5).

15.6.3.3 Radiological Consequences using Alternate Source Term Methodology

a. Background

This event is assumed to be caused by the instantaneous rupture of a Steam Generator tube that relieves to the lower pressure secondary system. No melt or clad breach is postulated for the Seabrook SGTR event.

A single ASDV is assumed to stick open in the Seabrook SGTR analysis. Two stuck open ASDV scenarios are considered. Case 1 assumes that a single ASDV fails open when level reaches 33% in the affected SG. Case 2 assumes that a single ASDV fails open 3 minutes following reactor trip. The failed open ASDV is assumed to be reclosed 20 minutes after failing open.

b. Compliance with RG 1.183 Regulatory Positions

The SGTR dose consequence analysis is consistent with the guidance provided in RG 1.183, Appendix F, "Assumptions for Evaluating the Radiological Consequences of a PWR Steam Generator Tube Rupture Accident," as discussed below:

1. Regulatory Position 1 – No fuel damage is postulated to occur for the Seabrook SGTR event.
2. Regulatory Position 2 – No fuel damage is postulated to occur for the Seabrook SGTR event. Two cases of iodine spiking are assumed.
3. Regulatory Position 2.1 – One case assumes a reactor transient prior to the postulated SGTR that raises the primary coolant iodine concentration to the maximum allowed by Tech Specs, which is a value of 60.0 μCi/gm DE I-131 for the analyzed conditions. This is the pre-accident spike case.
4. Regulatory Position 2.2 – One case assumes the transient associated with the SGTR causes an iodine spike. The spiking model assumes the primary coolant activity is initially at the Tech Spec limit of 1.0 μCi/gm DE I-131. Iodine is assumed to be released from the fuel into the RCS at a rate of 335 times the iodine equilibrium release rate for a period of 8 hours. This is the accident-induced spike case.
5. Regulatory Position 3 – The activity released from the fuel is assumed to be released instantaneously and homogeneously through the primary coolant.
6. Regulatory Position 4 – Iodine releases from the steam generators to the environment are assumed to be 97% elemental and 3% organic.

7. Regulatory Position 5.1 – The primary-to-secondary accident induced leakage rate is apportioned between the SGs as specified by the Technical Specification Steam Generator Program (1.0 gpm total, 500 gpd to any one SG). The tube leakage is conservatively apportioned as 313.33 gpd to the faulted SG and 1126.67 gpd total to the other three SGs in order to maximize dose consequences.

8. Regulatory Position 5.2 – The density used in converting volumetric leak rates to mass leak rates is consistent with the basis of surveillance tests used to show compliance with the SG leak rate TS. For the intact Steam Generators, the primary-to-secondary leak rate is based on a density of 1.09 gm/cc (cold liquid).

9. Regulatory Position 5.3 – The primary-to-secondary leak rate is assumed to continue until the temperature of the leakage is less than 212°F at 48 hours. The release of radioactivity from the SGs is assumed to continue until shutdown cooling is in operation and steam release from the SGs is terminated (RHR initiation at 8 hours).

10. Regulatory Position 5.4 – The release of fission products from the secondary system is evaluated with the assumption of a coincident loss of offsite power (LOOP).

11. Regulatory Position 5.5 – All noble gases released from the primary system are assumed to be released to the environment without reduction or mitigation.

12. Regulatory Position 5.6 – Regulatory Position 5.6 refers to Appendix E, Regulatory Positions 5.5 and 5.6. The iodine transport model for release from the steam generators is as follows:

    - Appendix E, Regulatory Position 5.5.1 – Tube uncovery is not postulated for this event; therefore, the primary-to-secondary leakage is assumed to mix with the secondary water without flashing for all steam generators.

    - Appendix E, Regulatory Position 5.5.2 – A portion of the primary-to-secondary ruptured tube flow through the SGTR is assumed to flash to vapor, based on the thermodynamic conditions in the reactor and secondary. The portion that flashes immediately to vapor is assumed to rise through the bulk water of the SG, enter the steam space, and be immediately released to the environment with no mitigation; i.e., no reduction for scrubbing within the SG bulk water is credited.

- Appendix E, Regulatory Position 5.5.3 – All of the SG tube leakage and ruptured tube flow that does not flash is assumed to mix with the bulk water.
- Appendix E, Regulatory Position 5.5.4 – The radioactivity within the bulk water is assumed to become vapor at a rate that is a function of the steaming rate and the partition coefficient. A partition coefficient of 100 is assumed for the iodine. The retention of particulate radionucludes in the SGs is limited by the moisture carryover from the SGs. The same partition coefficient of 100, as used for iodine, is assumed for other particulate radionuclides. This assumption is consistent with the SG carryover rate of less than 1%.
- Appendix E, Regulatory Position 5.6 – Steam generator tube bundle uncovery is not postulated for this event for Seabrook.

c. <u>Other Assumptions</u>

1. RCS and SG volume are assumed to remain constant throughout both the pre-accident and the accident-induced iodine spike SGTR events.
2. Data used to calculate the iodine equilibrium appearance rate are provided in Table 15.6-10.

d. <u>Methodology</u>

Input assumptions used in the dose consequence analysis of the SGTR event are provided in Table 15.6-6. This event is assumed to be caused by the instantaneous rupture of a steam generator tube releasing primary coolant to the lower pressure secondary system. In the unlikely event of a concurrent loss of power, the loss of circulating water through the condenser would eventually result in the loss of condenser vacuum. Valves in the condenser bypass lines would automatically close to protect the condenser thereby causing steam relief directly to the atmosphere from the ASDVs or MSSVs. This direct steam relief continues until it is terminated by initiation of RHR cooling (8 hours).

A thermal-hydraulic analysis is performed to determine a conservative maximum break flow, break flashing flow, and steam release inventory through the faulted SG relief valves. Additional activity, based on the proposed primary-to-secondary leakage limits, is released via the MSSVs or ASDVs until the RHR system is placed in operation to continue heat removal from the primary system.

No fuel melt or clad breach is postulated for the SGTR event. Consistent with RG 1.183 Appendix F, Regulatory Position 2, if no or minimal fuel damage is postulated for the limiting event, the activity release is assumed as the maximum allowed by Technical Specifications for two cases of iodine spiking: (1) maximum pre-accident iodine spike, and (2) maximum accident-induced, or concurrent, iodine spike.

For the case of a pre-accident iodine spike, a reactor transient is assumed to have occurred prior to the postulated SGTR event. The primary coolant iodine concentration is increased to the maximum value of 60 μCi/gm DE I-131 permitted by Technical Specification 3.4.8. The iodine activities for the pre-accident spike case are presented in Table 15.6-9. Primary coolant is released into the ruptured SG by the tube rupture and by a fraction of the total proposed allowable primary-to-secondary leakage. Activity is released to the environment from the ruptured SG via direct flashing of a fraction of the released primary coolant from the tube rupture and also via steaming from the ruptured SG ASDVs. The unaffected SGs are used to cool down the plant during the SGTR event. Primary-to-secondary tube leakage is also postulated into the intact SGs. Activity is released via steaming from the SG MSSVs/ASDVs until the decay heat generated in the reactor core can be removed by the RHR system at 8 hours into the event. These release assumptions are consistent with the requirements of RG 1.183.

For the case of the accident-induced spike, the postulated SGTR event induces an iodine spike. The RCS activity is initially assumed to be 1.0 μCi/gm DE I-131 as allowed by Technical Specifications. Iodine is released from the fuel into the RCS at a rate of 335 times the iodine equilibrium release rate for a period of 8 hours. The iodine activities for the accident-induced (concurrent) iodine spike case are presented in Table 15.6-11. All other release assumptions for this case are identical to those for the pre-accident spike case.

For this event, the Control Room ventilation system cycles through two modes of operation:

- Initially the ventilation system is assumed to be operating in normal mode. The air intake to the Control Room during this mode is 1000 cfm of unfiltered fresh air.

- After the start of the event, the Control Room normal air intake is conservatively assumed to isolate on a CR intake radiation monitor signal. A 30-second delay is conservatively applied to account for the time to reach the signal, the diesel generator start time, load sequencing, and damper actuation and positioning time. After isolation of the Control Room normal air intake, the air flow distribution consists of 600 cfm of filtered makeup flow through the worst of the two emergency intakes, 300 cfm of unfiltered inleakage and 390 cfm of filtered recirculation flow.
- 20 cfm of unfiltered inleakage was assumed to enter the Control Room via the CR fire exit and 280 cfm was assumed to enter via the Diesel Building.
- The Control Room ventilation filter efficiencies that are applied to the filtered makeup and recirculation flows are 99% for particulates/aerosols, and 95% for elemental and organic iodine.

e. <u>Radiological Consequences</u>

The atmospheric dispersion factors (*X/Qs*) used for this event for the Control Room dose are based on the postulated release locations and the operational mode of the control room ventilation system. These *X/Qs* are summarized in Table 2R-2 and Table 2R-3.

Releases from the intact and faulted SGs are assumed to occur from the MSSV/ASDV that produces the most limiting *X/Qs* when combined with the limiting applicable control room intake.

For the EAB and LPZ dose analysis, the *X/Q* factors for the appropriate time intervals are used. These *X/Q* factors are provided in Appendix 2Q.

The radiological consequences of the SGTR Accident are analyzed using the RADTRAD-NAI code and the inputs/assumptions previously discussed. Two activity release cases corresponding to the RCS maximum pre-accident iodine spike and the accident-induced iodine spike, based on Tech Spec limits, are analyzed. In addition, two ASDV failure cases are analyzed. As shown in Table 15.6-12, the radiological consequences of the Seabrook SGTR event for EAB dose, LPZ dose, and Control Room dose are all within the appropriate regulatory acceptance criteria.

15.6.3.4 Conclusions

The offsite doses from a postulated steam generator tube rupture at Seabrook Station are well within the exposure guideline values. Thus, the occurrence of this postulated accident will not result in an undue hazard to the general public.

15.6.4 Spectrum of BWR Steam System Piping Failures Outside of Containment

Not applicable to Seabrook.

15.6.5 Loss-of-Coolant Accidents Resulting from a Spectrum of Postulated Piping Breaks within the Reactor Coolant Pressure Boundary

15.6.5.1 Identification of Causes and Frequency Classification

A LOCA is the result of a pipe rupture of the RCS pressure boundary (see Section 5.2). For the analyses reported here, a major pipe break (large break) is defined as a rupture with a total cross-sectional area equal to or greater than 1.0 ft^2. This event is considered an American Nuclear Society (ANS) Condition IV event, a limiting fault, in that it is not expected to occur during the lifetime of the plant but is postulated as a conservative design basis (see Section 15.0.1).

Ruptures of small cross-section will cause expulsion of the coolant at a rate which can be accommodated by the high head safety injection pumps and which would maintain an operational water level in the pressurizer permitting the operator to execute an orderly shutdown. The coolant which would be released to containment contains the fission products present in it.

A minor pipe break (small break), as considered in this section, is defined as a rupture of the RCS piping with a cross-sectional area less than 1.0 ft^2, in which the normally operating charging system flow is not sufficient to sustain pressurizer level and pressure. This event is considered an American Nuclear Society (ANS) Condition III event, which is a fault which may occur very infrequently during the life of the plant.

It must be demonstrated that there is a high level of probability that the Acceptance Criteria for the LOCA as described in the 10 CFR 50.46 (Reference 1) are met.

1. There is a high level of probability that the peak cladding temperature (PCT) shall not exceed 2200°F.
2. The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel were to react.
3. The maximum calculated local oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation.
4. Calculated changes in core geometry shall be such that the core remains amenable to cooling. This requirement is met by demonstrating that the PCT does not exceed 2200°F, the maximum local oxidation does not exceed 17%, and the seismic and LOCA forces are not sufficient to distort the fuel assemblies to the extent that the core cannot be cooled.

5. After any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long lived radioactivity remaining in the core.

These criteria were established to provide a significant margin in Emergency Core Cooling System (ECCS) performance following a LOCA. Reference 2 presents a study in regards to the probability of occurrence of RCS pipe ruptures.

In all cases, small breaks (less than 1.0 ft^2) yield results with more margin to the Acceptance Criteria limits than large breaks.

15.6.5.2 Major Reactor Coolant System Pipe Ruptures (Loss-of-Coolant-Accident)

The analysis specified by 10 CFR 50.46 (Reference 1), "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Power Reactors", is presented in this section. The results of the Best-Estimate large break loss-of-coolant accident (LOCA) analysis are summarized in Table 15.6-32, and show compliance with the acceptance criteria.

For the purpose of ECCS analyses, Westinghouse (W) defines a large break loss-of-coolant accident (LOCA) as a rupture 1.0 ft^2 or larger of the reactor coolant system piping including the double ended rupture of the largest pipe in the reactor coolant system or of any line connected to that system.

Should a major break occur, rapid depressurization of the Reactor Coolant System (RCS) to a pressure nearly equal to the containment pressure occurs in approximately 27.5 seconds, with a nearly complete loss of system inventory. Rapid voiding in the core shuts down reactor power. A safety injection system signal is actuated when the low pressurizer pressure setpoint is reached. These countermeasures will limit the consequences of the accident in two ways:

1. Borated water injection complements void formation in causing rapid reduction of power to a residual level corresponding to fission product decay heat. An average RCS/sump mixed boron concentration is calculated to ensure that the post-LOCA core remains subcritical. However, no credit is taken for the insertion of control rods to shut down the reactor in the large break analysis.
2. Injection of borated water provides heat transfer from the core and prevents excessive cladding temperatures.

Before the break occurs, the reactor is assumed to be in a full power equilibrium condition, i.e., the heat generated in the core is being removed through the steam generator secondary system. At the beginning of the blowdown phase, the entire RCS contains sub-cooled liquid (except for pressurizer, which is at T_{sat}) which transfers heat from the core by forced convection with some fully developed nucleate boiling. During blowdown, heat from fission product decay, hot internals and the vessel, continues to be transferred to the reactor coolant. After the break develops, the time to departure from nucleate boiling is calculated. Thereafter, the core heat transfer is unstable, with both nucleate boiling and film boiling occurring. As the core becomes voided, both transition boiling and forced convection are considered as the dominant core heat transfer mechanisms. Heat transfer due to radiation is also considered.

The heat transfer between the RCS and the secondary system may be in either direction, depending on the relative temperatures. In the case of the large break LOCA, the primary pressure rapidly decreases below the secondary system pressure and the steam generators are an additional heat source. In the Seabrook Station Large Break LOCA analysis using the Best-Estimate methodology, the steam generator secondary is conservatively assumed to be isolated (main feedwater and steam line) at the initiation of the event to maximize the secondary side heat load.

15.6.5.2.1 Performance Criteria for Emergency Core Cooling System

The reactor is designed to withstand thermal effects caused by a loss-of-coolant accident including the double-ended severance of the largest reactor cooling system cold leg pipe. The reactor core and internals together with the Emergency Core Cooling System (ECCS) are designed so that the reactor can be safely shut down and the essential heat transfer geometry of the core preserved following the accident. Long-term coolability is maintained.

When the RCS depressurizes to approximately 634.7 psia, the accumulators begin to inject borated water into the reactor coolant loops. Borated water from the accumulator in the broken loop is assumed to spill to containment and be unavailable for core cooling for breaks in the cold leg of the RCS. Flow from the accumulators in the intact loops may not reach the core during depressurization of the RCS due to the fluid dynamics present during the ECCS bypass period. ECCS bypass results from the momentum of the fluid flow up the downcomer due to a break in the cold leg, which entrains ECCS flow out toward the break. Bypass of the ECCS diminishes as mechanisms responsible for the bypassing are calculated to be no longer effective.

The blowdown phase of the transient ends when the RCS pressure reaches approximately 40 psia. After the end of the blowdown, refill of the reactor vessel lower plenum begins. Refill is completed when emergency core cooling water has filled the lower plenum of the reactor vessel, which is bounded by the bottom of the active fuel region of the fuel rods (called bottom of core (BOC) recovery time).

The reflood phase of the transient is defined as the time period lasting from BOC recovery until the reactor vessel has been filled with water to the extent that the core temperature rise has been terminated. From the latter stage of blowdown and on into the beginning of reflood, the intact loop accumulator tanks rapidly discharges borated cooling water into the RCS. Although a portion injected prior to end of bypass is lost out the cold leg break, the accumulators eventually contributes 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 from the centrifugal charging pump (CCP), the safety injection pump (SIP) and the residual heat removal pump (RHR) aids in the filling of the downcomer and core and subsequently supply water to help 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 (RWST) reaches a minimum allowable value, coolant for long-term cooling of the core is obtained by switching from the injection mode to the sump recirculation mode of ECCS operation. Spilled borated water is drawn from the engineered safety features (ESF) containment sumps by the RHR pumps and returned to the RCS cold legs. Figure 15.6-3 contains a schematic of a representative sequence of events for the Seabrook Station Best-Estimate large break LOCA transient.

For the Best-Estimate large break LOCA analysis, one ECCS train, including one centrifugal charging pump (CCP), one safety injection pump (SIP) and one low head pump (RHR), starts and delivers flow through the injection lines. The accumulator and safety injection flows from the broken loop were assumed to spill to containment. All emergency diesel generators (EDGs) are assumed to start in the modeling of the containment spray pumps. Modeling full containment heat removal systems operation is required by Branch Technical Position CSB 6-1 (Reference 9) and is conservative for the large break LOCA.

To minimize delivery to the reactor, the CCP, SIP and RHR branch line chosen to spill is selected as the one with the minimum resistance.

15.6.5.2.2 Large Break LOCA Analytical Model

In 1988, as a result of the improved understanding of LOCA thermal-hydraulic phenomena gained by extensive research programs, the NRC staff amended the requirements of 10 CFR 50.46 and Appendix K, "ECCS Evaluation Models," so that a realistic evaluation model may be used to analyze the performance of the ECCS during a hypothetical LOCA (Reference 1). Under the amended rules, best-estimate thermal-hydraulic models may be used in place of models with Appendix K features. The rule change also requires, as part of the analysis, an assessment of the uncertainty of the best-estimate calculations. It further requires that this analysis uncertainty be included when comparing the results of the calculations to the prescribed acceptance limits. Further guidance for the use of best-estimate codes was provided in Regulatory Guide 1.157 (Reference 5).

To demonstrate use of the revised ECCS rule, the NRC and its consultants developed a method called the Code Scaling, Applicability, and Uncertainty (CSAU) evaluation methodology (Reference 6). This method outlined an approach for defining and qualifying a best-estimate thermal-hydraulic code and quantifying the uncertainties in a LOCA analysis.

A LOCA evaluation methodology for three- and four-loop PWR plants based on the revised 10 CFR 50.46 rules was developed by Westinghouse with the support of EPRI and Consolidated Edison and was approved by the NRC (Reference 7). The methodology is documented in WCAP-12945, "Code Qualification Document (CQD) for Best Estimate LOCA Analysis" (Reference 8).

A thermal-hydraulic computer code which was reviewed and approved for the calculation of fluid and thermal conditions in the PWR during a large break LOCA is WCOBRA/TRAC Version MOD7A, Rev. 1 (Reference 8).

WCOBRA/TRAC combines two-fluid, three-field, multi-dimensional fluid equations used in the vessel with one-dimensional drift-flux equations used in the loops to allow a complete and detailed simulation of a PWR. This best-estimate computer code contains the following features:

- Ability to model transient three-dimensional flows in different geometries inside the vessel.
- Ability to model thermal and mechanical non-equilibrium between phases
- Ability to mechanistically represent interfacial heat, mass, and momentum transfer in different flow regimes.
- Ability to represent important reactor components such as fuel rods, steam generators, reactor coolant pumps, etc.

The reactor vessel is modeled with the three-dimensional, three-field fluid model, while the loop, major loop components, and safety injection points are modeled with the one-dimensional fluid model.

The basic building block for the vessel is the channel, a vertical stack of single mesh cells. Several channels can be connected together by gaps to model a region of the reactor vessel. Regions that occupy the same level form a section of the vessel. Vessel sections are connected axially to complete the vessel mesh by specifying channel connections between sections. Heat transfer surfaces and solid structures that interact significantly with the fluid can be modeled with rods and unheated conductors. The fuel parameters are generated using the Westinghouse fuel performance code (PAD, 4.0, Reference 4).

One-dimensional components are connected to the vessel. Special purpose components exist to model specific components such as the steam generator and pump.

A typical calculation using WCOBRA/TRAC begins with the establishment of a steady-state initial condition with all loops intact. The input parameters and initial conditions for this steady-state calculation are discussed in the next section.

Following the establishment of an acceptable steady-state condition, the transient calculation is initiated by introducing a break into one of the loops. The evolution of the transient through blowdown, refill, and reflood follows continuously, using the same computer code (WCOBRA/TRAC) and the same modeling assumptions. Containment pressure is modeled with the BREAK component using a time dependent pressure table. Containment pressure is calculated using the COCO code (References 3 and 11) and mass and energy releases from the WCOBRA/TRAC calculation. The parameters used in the containment analysis to determine this pressure curve are presented in Tables 15.6-26 through 15.6-28.

The methods used in the application of WCOBRA/TRAC to the large break LOCA are described in Reference 8. A detailed assessment of the computer code WCOBRA/TRAC was made through comparisons to experimental data. These assessments were used to develop quantitative estimates of the code's ability to predict key physical phenomena in a PWR large break LOCA. Modeling of a PWR introduces additional uncertainties which are identified and quantified in the plant-specific analysis. The final step of the best-estimate methodology is to combine all the uncertainties related to the code and plant parameters and estimate the PCT at the 95th percentile ($PCT^{95\%}$). The steps taken to derive the PCT uncertainty estimate are summarized below:

1. Plant Model Development

   In this step, WCOBRA/TRAC model of the Seabrook Station is developed. A high level of noding detail is used, in order to provide an accurate simulation of the transient. However, specific guidelines are followed to assure that the model is consistent with models used in the code validation. This results in a high level of consistency among plant models, except for specific areas dictated by hardware differences such as in the upper plenum of the reactor vessel or the ECCS injection configuration.

2. Determination of Plant Operating Conditions

   In this step, the expected or desired range or the plant operating conditions to which the analysis applies is established. The parameters considered are based on a "key LOCA parameters" list that was developed as part of the methodology. A set of these parameters, at mostly nominal values, is chosen for input as initial conditions to the plant model. A transient is run utilizing these parameters and is known as the "initial transient." Next, several confirmatory turns are made, which vary a subset of the key LOCA parameters over their expected operating range in one-at-a-time sensitivities. The most limiting input conditions, based on these confirmatory runs, are then combined into a single transient, which is then called the "reference transient."

3. PWR Sensitivity Calculations

   A series of PWR transients are performed in which the initial fluid conditions and boundary conditions are ranged around the nominal conditions used in the reference transient.

   Next, a series of transients are performed which vary the power distribution, taking into account all possible power distributions during normal plant operation.

   Finally, a series of transients are performed which vary parameters that affect the overall system response ("global" parameters) and local fuel rod response ("local" parameters).

4. Response Surface Calculations

   The results from the power distribution and global model WCOBRA/TRAC runs performed in Step 3 are fit by regression analyses into equations known as response surfaces. The results of the initial conditions run matrix are used to generate a PCT uncertainty distribution.

5. Uncertainty Evaluation

   The total PCT uncertainty from the initial conditions, power distribution, and model calculations is derived using the approved methodology (Reference 8). The uncertainty calculations assume certain plant operating ranges which may be varied depending on the results obtained. These uncertainties are then combined to determine the initial estimate of the total PCT uncertainty distribution for the guillotine and limiting split breaks. The results of these initial estimates of the total PCT uncertainty are compared to determine the limiting break type. If the split break is limiting, an additional set of split transients are performed which vary overall system response ("global" parameters) and local fuel rod response ("local" parameters). Finally, an additional series of runs is made to quantify the bias and uncertainty due to assuming that the above three uncertainty categories are independent. The final PCT uncertainty distribution is then calculated for the limiting break type, and the 95th percentile PCT (PCT$^{95\%}$) is determined, as described in Section 15.6.5.2.3.6 (Uncertainty Evaluation).

6. Plant Operating Range

   The plant operating range over which the uncertainty evaluation applies is defined. Depending on the results obtained in the above uncertainty evaluation, this range may be the desired range established in step 2, or may be narrower for some parameters to gain additional PCT margin.

There are three major uncertainty categories or elements:

- Initial conditions bias and uncertainty
- Power distribution bias and uncertainty
- Model bias and uncertainty

Conceptually, these elements may be assumed to affect the reference transient PCT as shown below

$$PCT_i = PCT_{REF,i} + \Delta PCT_{IC,i} + \Delta PCT_{PD,i} + \Delta PCT_{MOD,i} \qquad (15.6.5.2\text{-}1)$$

Where,

$PCT_{REF,i}$ = **Reference transient PCT:** The reference transient PCT is calculated using WCOBRA/TRAC at the nominal conditions identified in Table 15.6-29, for the blowdown, first and second reflood periods.

$\Delta PCT_{IC,i}$ = **Initial condition bias and uncertainty:** This bias is the difference between the reference transient PCT, which assumes several nominal or average initial conditions, and the average PCT taking into account all possible values of the initial conditions. This bias takes into account plant variations which have a relatively small effect on PCT. The elements which make up this bias and its uncertainty are plant-specific.

$\Delta PCT_{PD,i}$ = **Power distribution bias and uncertainty:** This bias is the difference between the reference transient PCT, which assumes a nominal power distribution, and the average PCT taking into account all possible power distributions during normal plant operation. Elements which contribute to the uncertainty of this bias are calculational uncertainties, and variations due to transient operation of the reactor.

$\Delta PCT_{MOD,i}$ = **Model bias and uncertainty:** This component accounts for uncertainties in the ability of the WCOBRA/TRAC code to accurately predict phenomena which affect the overall system response ("global" parameters) and the local fuel rod response ("local" parameters). The code and model bias is the difference between the reference transient PCT, which assumes nominal values for the global and local parameters, and the average PCT taking into account all possible values of global and local parameters.

The separability of the bias and uncertainty components in the manner described above is an approximation, since the parameters in each element may be affected by parameters in other elements. The bias and uncertainty associated with this assumption is quantified as part of the overall uncertainty methodology and included in the final estimates of the $PCT^{95\%}$.

15.6.5.2.3 Large Break LOCA Analysis Results

A series of WCOBRA/TRAC calculations were performed using the Seabrook Station input model, to determine the effect of variations in several key LOCA parameters on peak cladding temperature (PCT). From these studies, an assessment was made of the parameters that had a significant effect as will be described in the following sections.

15.6.5.2.3.1 Large Break LOCA Reference Split Break Transient Description

The plant-specific analysis performed for the Seabrook Station confirmed that the split break is more limiting than the double-ended cold leg guillotine (DECLG) break. Because split break is limiting, this split break transient (CD = 2.0) will now be known as the reference split break transient. The plant conditions used in the reference split break transient are listed in Table 15.6-29. Since many of these parameters are at their bounded values, the calculated results are a conservative representation of the response to a large break LOCA. The following is a description of the reference split break reference transient.

The LOCA transient can be conveniently divided into a number of time periods in which specific phenomena are occurring. For a typical large break, the blowdown period can be divided into the critical heat flux (CHF) phase, the upward core flow phase, and the down-ward core flow phase. These are followed by the refill, reflood 1, reflood 2 and long term cooling phases. The important phenomena occurring during each of these phases are discussed for the reference split break transient. The results are shown in Figures 15.6-7 through 15.6-19.

Criteria Heat Flux (CHF) Phase (~ 0-2 seconds)

Immediately following the cold leg rupture, the break discharge is subcooled and high flow rate, the core flow reverses, the fuel rods go through departure from nucleate boiling (DNB) and the cladding rapidly heats up while core power shuts down. Figure 15.6-7 shows the maximum cladding temperature in the core, as a function of time. The hot water in the core and upper plenum flashes to steam during this period. This phase is terminated when the water in the lower plenum and downcomer begins to flash. The mixture swells and the intact loop pumps, still rotating in single-phase liquid, push this two-phase mixture into the core.

Upward Core Flow Phase (~ 2-11.5 seconds)

Heat transfer is improved as the two-phase mixture is pushed into the core. This phase may be enhanced if the pumps are not degraded, and the break discharge rate is low because the fluid is saturated at the break. Figure 15.6-8 shows the break flowrate for the reference split break transient. This phase ends as lower plenum mass is depleted, the loops become two-phase, and the pump head degrades. If pumps are highly degraded or the break flow is large, the cooling effect due to upward flow may not be significant. Figure 15.6-9 shows the void fraction for one intact loop pump and the broken loop pump. The intact loop pump remains in single-phase liquid flow for several seconds, while the broken loop pump is in two-phase and steam flow soon after the break.

Downward Core Flow Phase (~ 11.5-28 seconds)

The loop flow is pushed into the vessel by the intact loop pumps and decreases as the pump flow becomes two-phase. The break flow begins to dominate and pulls flow down through the core. Figures 15.6-10 and 15.6-11 show the vapor flow at the mid-core of channels 13 and 15. While liquid and entrained liquid flows also provide core cooling, the vapor flow in the core best illustrates this phase of core cooling. This period is enhanced by flow from the upper head. As the system pressure continues to fall, the break flow and consequently the core flow, are reduced. The core begins to heat up as the system reaches containment pressure and the vessel begins to fill with Emergency Core Cooling System (ECCS) water.

Refill Phase (~ 28-33 seconds)

The core experiences a nearly adiabatic heatup as the lower plenum fills with ECCS water. Figure 15.6-12 shows the lower plenum liquid level. This phase ends when the ECCS water enters the core and entrainment begins, with a resulting improvement in heat transfer. Figure 15.6-13 and Figure 15.6-14 shows the liquid flows from the accumulator and the safety injection from an intact loop (Loop 1).

First Reflood Phase (~ 33-50 seconds)

The accumulators are emptying and nitrogen enters into the system (Figure 15.6-13). This forces water into the core which then boils as the lower core region begins to quench, causing repressurization. The repressurization is best illustrated by the reduction in pumped SI flow (Figure 15.6-14). During this time, core cooling may be increased.

Second Reflood Phase (~ 50 seconds – end)

The system then settles into a gravity driven reflood which exhibits lower core heat transfer. Figures 15.6-15 and 15.6-16 show the core and downcomer liquid levels. Figure 15.6-17 shows the vessel fluid mass. As the quench front progresses further into the core, the peak cladding temperature (PCT) location moves higher in the top core region. Figure 15.6-18 shows the movement of the PCT location. As the vessel continues to fill, the PCT location is cooled and the PCT heatup is terminated (Figures 15.6-7 and 15.6-9).

Long Term Core Cooling

At the end of the WCOBRA/TRC calculation, the core and downcomer levels are increasing as the pumped safety injection flow exceeds the break flow. The core and downcomer levels would be expected to continue to rise, until the downcomer mixture levels approaches the loop elevation. At that point, the break flow would increase, until it roughly matches the injection flowrate. The core would continue to be cooled until the entire core is eventually quenched.

The reference split break transient resulted in a first reflood PCT of 1570°F and a second reflood PCT of 1567°F.

15.6.5.2.3.2 Confirmatory Sensitivity Studies

A number of sensitivity calculations were carried out to investigate the effect of the key LOCA parameters, and to determine the reference transient. In the sensitivity studies performed, LOCA parameters were varied one at a time. For each sensitivity study, a comparison between the base case and the sensitivity case transient results was made.

The results of the sensitivity studies are summarized in Tables 15.6-30 and 15.6-31. The results of these analyses lead to the following conclusions:

1. Modeling maximum steam generator tube plugging (10%) results in a higher PCT than minimum steam generator tube plugging (0%).
2. Modeling loss-of-offsite-power (LOOP) results in a higher PCT than no loss-of-offsite-power (no-LOOP).
3. Modeling the maximum value of vessel average temperature (T_{avg} = 589.1°F) results in a higher PCT than minimum value of vessel average temperature (T_{avg} = 571.0°F).
4. Modeling the minimum power fraction (P_{LOW} = 0.2) in the low power/periphery channel of the core results in a higher PCT than maximum power fraction. (P_{LOW} = 0.6).
5. For the split break confirmatory study, it was determined that the limiting split break area is 2 times the area of a cold leg pipe (C_D = 2.0).

15.6.5.2.3.3 Initial Conditions Sensitivity Studies

Several calculations were performed to evaluate the effect of change in the initial conditions on the calculated LOCA transient. These calculations analyzed key initial plant conditions over their expected range of operation. These studies included effects of ranging RCS conditions (pressure and temperature), safety injection temperature, and accumulator conditions (pressure, temperature and water volume).

The calculated results were used to develop initial condition uncertainty distributions for the blowdown and reflood peaks. These distributions are then used in the uncertainty evaluation to predict the PCT uncertainty component resulting from initial conditions uncertainty ($\Delta PCT_{IC,i}$).

15.6.5.2.3.4 Power Distribution Sensitivity Studies

Several calculations were performed to evaluate the effect of power distribution on the calculated LOCA transient. The power distribution attributes which were analyzed are the peak linear heat rate, the maximum relative rod power, the relative power in the bottom third of the core (P_{BOT}), and the relative power in the middle third of the core (P_{MID}). The choice of these variables and their ranges are based on the expected range of plant operation.

The power distribution parameters used for the reference transient are biased to yield a relatively high PCT. The reference transient uses a lightly higher $F_{\Delta H}$ value (1.683) than the Tech Spec $F_{\Delta H}$ value (1.65), a skewed to the top power distribution, and a F_Q (2.2) at the midpoint of the sample range.

A run matrix was developed in order to vary the power distribution attributes singly and in combination. The sensitivity results indicated that power distributions with peak powers shifted towards the top of the core produced higher PCTs.

The calculated results were used to develop response surfaces, as described in Step 4 of Section 15.6.5.2.2, which could be used to predict the change in PCT for various changes in the power distributions for the blowdown and reflood peaks. These were then used in the uncertainty evaluation, to predict the PCT uncertainty component resulting from uncertainties in power distribution parameters, ($\Delta PCT_{PD,i}$).

15.6.5.2.3.5 Global Model Sensitivity Studies

Several calculations were performed to evaluate the effect of broken loop resistance, break discharge coefficient, and condensation rate on the PCT for the guillotine break. As in the power distribution study, these parameters were varied singly and in combination in order to obtain a database which could be used for response surface generation. The run matrix and ranges of the break flow parameters are described in Reference 8. The limiting split break was also identified using the methodology described in Reference 8. The results of these studies indicated that the split break calculation with an area equal to 2 times the cold leg area results in the highest PCT. This requires that the effect of broken loop resistance and condensation must be reevaluated for the limiting split area.

The calculated results were used to develop response surfaces as described in Section 15.6.5.2.2, which could be used to predict the change in PCT for various changes in the flow conditions. These were then used in the uncertainty evaluation to predict the PCT uncertainty component resulting from uncertainties in global model parameters ($\Delta PCT_{MOD,i}$).

15.6.5.2.3.6 Uncertainty Evaluation and Results

The PCT equation was presented in Section 15.6.5.2.2. Each element of uncertainty is initially considered to be independent of the other. Each bias component is considered a random variable, whose uncertainty and distribution is obtained directly, or is obtained from the uncertainty of the parameters of which the bias is a function. For example, $\Delta PCT_{PD,i}$ is a function of F_Q, $F_{\Delta H}$, P_{BOT}, and P_{MID}. Its distribution is obtained by sampling the plant F_Q, $F_{\Delta H}$, P_{BOT}, and P_{MID} distributions and using a response surface to calculate $\Delta PCT_{PD,i}$. Since ΔPCT_i is the sum of these biases, it also becomes a random variable. Separate initial PCT frequency distributions are constructed as follows for the guillotine break and the limiting split break size:

1. Generate a random value of each ΔPCT element.
2. Calculate the resulting PCT using Equation 15.6.5.2-1.

3. Repeat the process many times to generate a histogram of PCTs.

For the Seabrook Station, the results of this assessment showed the split break to potentially be limiting. Additional split break calculations were then performed, a more detailed description of $\Delta PCT_{MOD,i}$ was developed, and steps 1 through 3 repeated for the limiting split break. This analysis confirmed the split break to be the limiting break type. As the result of this analysis, the split break is used in the final verification step and the limiting split break transient (CD = 2.0) becomes the reference split break transient for the Seabrook Station.

A final verification step is performed in which additional calculations (known as "superposition" calculations) are made with WCOBRA/TRAC, simultaneously varying several parameters which were previously assumed independent (for example, power distributions and global models). Predictions using Equation 15.6.5.2-1 are compared to this data, and additional biases and uncertainties are applied.

The estimate of the PCT 95th percent probability is determined by finding that PCT below which 95th percent of the calculated PCTs reside. This estimate is the licensing basis PCT, under the revised ECCS rule.

The results for the Seabrook Station are given in Table 15.6-32, which shows the limiting first reflood 95th percentile PCT ($PCT^{95\%}$) of 1789°F. As expected, the difference between the 95th percent value and the average value increases with increasing time, as more parameter uncertainties come into play.

15.6.5.2.3.7 Evaluation

The base analysis discussed in Sections 15.6.5.2.3.1 to 15.6.5.2.3.6 is for non-IFBA fuel. An analysis of IFBA fuel was performed independently, utilizing the HOTSPOT code and a high PCT case. The analysis result indicated that IFBA fuel is bounded by non-IFBA fuel.

15.6.5.2.4 Large Break LOCA Conclusions

It must be demonstrated that there is a high level of probability that the limits set forth in 10 CFR 50.46 are met. The demonstration that these limits are met for the Seabrook Station is as follows:

1. There is a high level of probability that the peak cladding temperature (PCT) shall not exceed 2200°F. The results presented in Table 15.6-32 indicate that this regulatory limit has been met with a reflood $PCT^{95\%}$ of 1789°F.

2. The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel were to react. The total amount of hydrogen generated, based on this conservative assessment is 0.003 times the maximum hypothetical amount, which meets the regulatory limit.

3. The maximum calculated local oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation. The approved Best-Estimate LOCA methodology assesses this requirement using a plant-specific transient which has a PCT in excess of the estimated 95 percentile PCT ($PCT^{95\%}$). Based on this conservative calculation, a maximum local oxidation of 3.53 percent is calculated, which meets the regulatory limit.

4. Calculated changes in core geometry shall be such that the core remains amenable to cooling. This requirement is met by demonstrating that the PCT does not exceed 2200°F, the maximum local oxidation does not exceed 17%, and the seismic and LOCA forces are not sufficient to distort the fuel assemblies to the extent that the core cannot be cooled. The BE methodology (Reference 8) specifies that the effects of LOCA and seismic loads on core geometry do not need to be considered unless grid crush extends to in-board assemblies. Fuel assembly structural analyses performed for Seabrook Station indicate that this condition does not occur. Therefore, this regulatory limit is met.

5. After any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long lived radioactivity remaining in the core. The conditions at the end of the WCOBRA/TRAC calculations indicate that the transition to long term cooling is underway even before the entire core is quenched.

15.6.5.2.5 Plant Operating Range

The expected PCT and its uncertainty developed above are valid for a range of plant operating conditions. In contrast to current Appendix K calculations, many parameters in the base case calculation are at nominal values. The range of variation of the operating parameters has been accounted for in the estimated PCT uncertainty. Table 15.6-33 summarizes the operating ranges for the Seabrook Station. If operation is maintained within these ranges, the LOCA analysis is considered to be valid.

15.6.5.3 Small Break LOCA

a. Sequence of Events and Systems Operations

The Seabrook Small Break Loss-of-Coolant Accident (SBLOCA) analysis was performed to support the Power Uprate[29]. Pertinent analysis assumptions include: licensed core power of 3659 MWt (including calorimetric uncertainty), 10% uniform SGTP, maximum peaking factor ($F_Q(Z)$) envelope of 2.50, hot channel enthalpy rise factor $F_{\Delta H}$ of 1.65 and RFA (w/IFMs) fuel. A break spectrum of 3 inch, 4 inch, and 6 inch breaks was analyzed, resulting in the 4 inch case being limiting with a peak cladding temperature (PCT) of 1373°F[29].

The results of this Small Break ECCS analysis, utilizing the currently approved NOTRUMP Evaluation Model [16, 17, 30], have shown that Seabrook remains in compliance with the requirements of 10 CFR 50.46.

Should a small break LOCA occur, depressurization of the RCS causes fluid to flow into the loops from the pressurizer resulting in a pressure and level decrease in the pressurizer. The reactor trip signal subsequently occurs when the pressurizer low-pressure trip setpoint is reached. Loss-of-Offsite Power (LOOP) is assumed to occur coincident with reactor trip. A safety injection signal is generated when the pressurizer low-low-pressure setpoint is reached. These countermeasures will limit the consequences of the accident in two ways:

- Reactor trip and borated water injection, together with void formation, cause a rapid reduction of nuclear power to a residual level corresponding to delayed fission and fission product decay. No credit is taken in the LOCA analysis for the boron content of the injection water. However, an average RCS/sump mixed boron concentration is calculated to ensure that the post-LOCA core remains subcritical. In addition, in the small break LOCA analysis, credit is taken for the insertion of Rod Cluster Control Assemblies (RCCAs) subsequent to the reactor trip signal, while assuming the most reactive RCCA is stuck in the full out position.
- Injection of borated water ensures sufficient flooding of the core to prevent excessive clad temperatures.

For small break LOCAs, the most limiting single active failure is the one that results in the minimum emergency core cooling system (ECCS) flow delivered to the RCS. This has been determined to be the loss of an emergency power train which results in the loss of one complete train of ECCS components. This means that credit can be taken for only one centrifugal charging pump (CCP), one safety injection pump (SIP), and one residual heat removal (RHR) (or low head) pump. During the small break transient, one ECCS train is assumed to start and deliver flow through the injection lines (one of each loop) with one branch injection line (SIP and CCP) spilling to the RCS backpressure. RHR flow is not modeled for small break LOCAs because the pressure will not fall below the RHR cut-in pressure before the end of the transient. To minimize delivery to the reactor, the branch line chosen to spill is selected as the one with the minimum resistance. In addition, the SIP and CCP performance curves were degraded by 10%.

1. <u>Description of Transient</u>

   The sequence of events following a small break LOCA are presented in Table 15.6-1.

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. After the small break LOCA is initiated, reactor trip occurs due to a low pressurizer signal. 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 the pumps coast down following LOOP. Upward flow through the core is maintained. However, the core flow is not sufficient to prevent a partial core uncovery. Subsequently, the ECCS provides sufficient core flow to cover the core.

During blowdown, heat from fission product decay, hot internals, and the vessel continues to be transferred to the RCS. The heat transfer between the RCS and the secondary system may be in either direction depending on the relative temperatures. Continued heat addition to the secondary system results in increased secondary system pressure which leads to steam relief via the main steam safety valves. Makeup to the secondary is automatically provided by the emergency feedwater pumps. The safety injection signal isolates normal feedwater flow by closing the main feedwater isolation, control, and bypass valves and initiates emergency feedwater flow by starting the emergency feedwater pumps. The secondary flow aids in the reduction of RCS pressure.

When the RCS depressurizes to approximately 600 psia, the cold leg accumulators begin to inject borated water into the reactor coolant loops. However, for most small breaks the vessel mixture level starts to increase, covering the fuel with ECCS pumped injection before accumulator injection begins.

b. <u>Core and System Performance</u>

1. <u>Evaluation Model</u>

The requirements of an acceptable ECCS evaluation model are presented in Appendix K of 10 CFR 50 (Reference 1).

For small breaks (less than 1.0 ft^2) the NOTRUMP digital computer code (References 16, 17, and 30) is employed to calculate the transient depressurization of the RCS as well as to describe the mass and energy of the fluid flow through the break. The NOTRUMP computer code is a one-dimensional general network code incorporating a number of advanced features. Among these are calculation of thermal non-equilibrium in all fluid volumes, flow regime-dependent drift flux calculations with counter-current flooding limitations, mixture level tracking logic in multiple-stacked fluid nodes and regime-dependent drift flux calculations with multiple-stacked fluid nodes and regime-dependent heat transfer correlations. Also, safety injection into the broken loop is modeled using the COSI condensation model [30]. The NOTRUMP small break LOCA ECCS evaluation model was developed to determine the RCS response to design basis small break LOCAs, and to address NRC concerns expressed in NUREG-0611 (Reference 18).

The RCS model is nodalized into volumes interconnected by flowpaths. The broken loop is modeled explicitly, while the intact loops are lumped into a single second loop. The transient behavior of the system is determined from the governing conservation equations of mass, energy, and momentum. The multimode capability of the program enables explicit, detailed spatial representation of various system components which, among other capabilities, enables a proper calculation of the behavior of the loop seal during a LOCA. The reactor core is represented as heated control volumes with associated phase separation models to permit transient mixture height calculations.

Clad thermal analyses are performed with the LOCTA-IV code (Reference 19) using the NOTRUMP calculated core pressure, fuel rod power history, uncovered core steam flow and mixture heights as boundary conditions. (Figure 15.6-6).

Figure 15.6-4 depicts the hot rod axial power shape used to perform the small break LOCA analysis presented here. The shape was chosen because it represents a distribution with power concentrated in the upper regions of the core. Such a distribution is limiting for small break LOCA because it minimizes coolant level swell, while maximizing vapor superheating and fuel rod heat generation at the uncovered elevations. The small break analysis assumes that the core continues to operate at full power until the control rods are completely inserted. For conservatism, it is assumed that the most reactive RCCA does not insert.

The safety injection performance, as modeled in the small break analysis, is presented in Figure 15.6-49. Conservatively, 10% head degradation is assumed for the charging and safety injection pumps.

Schematic representation of the computer code interface is given in Figure 15.6-6.

2. Input Parameters and Initial Conditions

Table 15.6-13 lists important input parameters and initial conditions used in the analysis.

The bases used to select the numerical values that are important parameters to the analysis have been conservatively determined from extensive sensitivity studies (See References 13, 14, and 15). In addition, the requirements of Appendix K regarding specific model features were met by selecting models which provide a significant overall conservatism in the analysis. The assumptions which were made pertain to the conditions of the reactor and associated safety system equipment at the time that the LOCA occurs, and include such items as the core peaking factors, the containment pressure, and the performance of the ECCS. Decay heat generated throughout the transient is also conservatively calculated.

3. Results

NUREG-0737 Section II.K.3.31 (Reference 20) requires a plant specific small break LOCA analysis using an Evaluation Model revised per Section II.K.3.30. In accordance with NRC generic letter 83-35 (Reference 21), generic analyses using, NOTRUMP (References 16 and 17) were performed and are presented in Reference 22. Those results demonstrate that in a comparison of cold leg, hot leg and pump suction leg break locations, the cold leg break of less than 10 inches in diameter is limiting.

Therefore, a range of small break analyses is presented which establishes the limiting break size. The results of these analyses are summarized in Table 15.6-1 and Table 15.6-15.

It was determined that, because of the low calculated PCT, rod burst and blockage effects would not have a significant effect on the small break results for Seabrook Station. Therefore, a fuel assembly burnup sensitivity study was not required.

Figures 15.6-4, 15.6-34 through 15.6-45, 15.6-49 present the principal parameters of interest for the small break LOCA ECCS analyses. For all cases analyzed the following transient parameters are presented:

(a) RCS pressure

(b) Core mixture height

(c) Hot spot clad temperature

For the limiting break analyzed, the following additional transient parameters are presented:

(a) Core steam flow rate

(b) Core heat transfer coefficient

(c) Hot spot fluid temperature

The limiting break PCT is 1373°F, which is less than the Acceptance Criteria limit of 2200°F of 10 CFR 50.46.

The small break LOCA results are well below all Acceptance Criteria limits of 10 CFR 50.46 and in all cases are not limiting when compared to the results presented for large breaks.

15.6.5.4 Radiological Consequences using Alternate Source Term Methodology

a. Background

This event is assumed to be caused by an abrupt failure of the main reactor coolant pipe and the ECCS fails to prevent the core from experiencing significant degradation (i.e., melting). This sequence cannot occur unless there are multiple failures, and thus goes beyond the typical design basis accident that considers a single active failure. Activity is released from the containment and from there, released to the environment by means of containment leakage and leakage from the ECCS.

b. Compliance with RG 1.183 Regulatory Positions

The LOCA dose consequence analysis is consistent with the guidance provided in RG 1.183, Appendix A "Assumptions for Evaluating the Radiological Consequences of a LWR Loss-of-Coolant Accident," as discussed below:

1. Regulatory Position 1 – The total core inventory of the radionuclide groups utilized for determining the source term for this event is based on RG 1.183, Regulatory Position 3.1, at 102% of core thermal power and is provided in Table 15C-1. The core inventory release fractions for the gap release and early in-vessel damage phases of the LOCA are consistent with Regulatory Position 3.2 and Table 2 of RG 1.183.

2. Regulatory Position 2 – The sump pH is controlled at a value greater than 7.0 per UFSAR Section 6.5.2.2. Therefore, the chemical form of the radioiodine released to the containment is assumed to be 95% cesium iodide (CsI), 4.85% elemental iodine, and 0.15% organic iodide. With the exception of elemental and organic iodine and noble gases, fission products are assumed to be in particulate form.

3. Regulatory Position 3.1 – The activity released from the fuel is assumed to mix instantaneously and homogeneously throughout the free air volume of the containment. The release into the containment is assumed to terminate at the end of the early in-vessel phase.

4. Regulatory Position 3.2 – Reduction of the airborne radioactivity in the containment by natural deposition is credited. A natural deposition removal coefficient for elemental iodine is calculated per SRP 6.5.2 as 2.23 hr^{-1}. This removal is credited in the sprayed and unsprayed regions. A natural deposition removal coefficient of 0.1 hr^{-1} is assumed (based on the Industry Degraded Core Rulemaking Program Technical Report 11.3, "Fission Product Transport in Degraded Core Accidents," Atomic Industrial Forum, December 1983) for all aerosols in the unsprayed regions with no credit of natural deposition of aerosols in the sprayed regions. No removal of organic iodine by natural deposition is assumed.

5. Regulatory Position 3.3 – Containment spray provides coverage to 85.4% of the containment. Therefore, the Seabrook containment building atmosphere is not considered to be a single, well-mixed volume. The containment is divided into sprayed and unsprayed regions. A mixing rate of two turnovers of the unsprayed region per hour is assumed.

The SRP limits the spray removal coefficient for elemental iodine to 20 hr^{-1}; therefore, although a higher value was calculated, 20 hr^{-1} was used for the elemental iodine spray removal coefficient. In addition, the SRP and Reg. Guide 1.183 specify a maximum decontamination factor of 200 for spray removal of elemental iodine. The maximum decontamination factor (DF) for the elemental iodine spray removal coefficient is based on the maximum airborne elemental iodine concentration in the containment. The time for the containment sprays to reach an elemental iodine decontamination factor of 200 was determined by running a containment leakage case without environment leakage paths.

Radioactive decay and natural deposition of iodine were conservatively left on as removal mechanisms contributing to the decontamination factor. Due to mixing between the sprayed and unsprayed regions of containment, the iodine activity in both containment regions was included in the determination of the time required to reach a decontamination factor of 200. The decontamination factor for elemental iodine reaches 200 at just over 2.92 hours.

The particulate iodine removal rate is reduced by a factor of 10 when a DF of 50 is reached. Based upon the calculated iodine aerosol removal rate of 5.75 hr^{-1}, the time of a DF of 50 is computed with the same model used to determine the elemental iodine DF of 200. The time for containment spray to produce an aerosol decontamination factor of 50 with respect to the containment atmosphere is just over 3.56 hours.

6. Regulatory Position 3.4 – Reduction in airborne radioactivity in the containment by filter recirculation systems is not assumed in this analysis.
7. Regulatory Position 3.5 – Not applicable to Seabrook.
8. Regulatory Position 3.6 – Not applicable to Seabrook.
9. Regulatory Position 3.7 – A containment leak rate of 0.15% per day of the containment air is assumed for the first 24 hours. After 24 hours, the containment leak rate is reduced to 0.075% per day of the containment air.
10. Regulatory Position 3.8 – Routine containment purge is considered in this analysis. The purge release evaluation assumes that 100% of the radionuclide inventory in reactor coolant system liquid (based on Technical Specification RCS equilibrium activity) is released to the containment at the initiation of the LOCA. The purge system is isolated before the onset of the gap release phase.

11. Regulatory Position 4.1 – Leakage from containment collected by the secondary containment is processed by ESF filters prior to an assumed ground level release.

12. Regulatory Position 4.2 – Leakage into the secondary containment is assumed to be released directly to the environment as a ground level release prior to drawdown of the secondary containment at 8 minutes.

13. Regulatory Position 4.3 – The containment enclosure emergency air cleaning system is credited as being capable of maintaining a negative pressure with respect to the outside environment considering the effect of high windspeeds and LOCA heat effects on the annulus as described in Sections 6.5.1.1 and 6.5.1.3.

14. Regulatory Position 4.4 – No credit is taken for dilution in the secondary containment volume.

15. Regulatory Position 4.5 – 60% of the primary containment leakage is assumed to bypass the secondary containment. This bypass leakage is released from containment without filtration.

16. Regulatory Position 4.6 – The containment enclosure emergency air cleaning system is credited as meeting the requirements of RG 1.52 and Generic Letter 99-02 per Section 6.5.1.3 and Table 6.5-1.

17. Regulatory Position 5.1 – Engineered Safety Feature (ESF) systems that recirculate water outside the primary containment are assumed to leak during their intended operation. With the exception of noble gases, all fission products released from the fuel to the containment are assumed to instantaneously and homogeneously mix in the containment sump water at the time of release from the core.

18. Regulatory Position 5.2 – Leakage from the ESF system is taken as two times 24 gallons per day for a total leakage rate of 48 gallons per day. The leakage is assumed to start at the earliest time the recirculation flow occurs in these systems and continues for the 30-day duration. Backleakage to the Refueling Water Storage Tank is also considered separately as two times the measured leakage value of 0.47975 gpm for a total leakage rate of 0.9595 gpm.

19. Regulatory Position 5.3 – With the exception of the iodine, all radioactive materials in the recirculating liquid are assumed to be retained in the liquid phase.

20. Regulatory Position 5.4 – A flashing fraction of 4.7% was determined based on the temperature of the containment sump liquid at the time recirculation begins. The iodine available for release at the time recirculation begins is based on expected sump pH history and temperature (see the Release Inputs in the Methodology section below). All of the iodine available for release is assumed to become airborne and leak directly to the environment from the initiation of recirculation through 30 days. For ECCS leakage back to the RWST, the analysis demonstrates that the temperature of the leaked fluid will cool below 212°F prior to release from the tank.

21. Regulatory Position 5.5 – The iodine available for release at the time recirculation begins is based on expected sump pH history and temperature (see the Release Inputs in the Methodology section below). The amount of iodine that becomes airborne is assumed to be 10% of the total iodine available and leak directly to the environment from the initiation of recirculation through 30 days. For the ECCS leakage back to the RWST, the sump and RWST pH history and temperature are used to evaluate the amount of iodine that enters the RWST air space.

22. Regulatory Position 5.6 – The temperature and pH history of the sump and RWST are considered in determining the radioiodine available for release and the chemical form. Credit is taken for hold-up and dilution of activity in the RWST as allowed by Regulatory Position 5.6. No credit for ESF filtration of the RWST leakage is taken. Filtration of non-RWST ECCS leakage is credited.

23. Regulatory Position 6 – Not applicable to Seabrook.

24. Regulatory Position 7 – Containment purge is not considered as a means of combustible gas or pressure control in this analysis; however, the effect of routine containment purge before isolation is considered.

c. <u>Methodology</u>

For this event, the Control Room ventilation system cycles through two modes of operation. Inputs and assumptions fall into three main categories: Radionuclide Release Inputs, Radionuclide Transport Inputs, and Radionuclide Removal Inputs.

For the purposes of the LOCA analyses, a major LOCA is defined as a rupture of the RCS piping, including the double-ended rupture of the largest piping in the RCS, or of any line connected to that system up to the first closed valve. Should a major break occur, depressurization of the RCS results in a pressure decrease in the pressurizer. A reactor trip signal occurs when the pressurizer low-pressure trip setpoint is reached. A safety injection system signal is actuated when the appropriate setpoint (high containment pressure or low pressurizer pressure) is reached. The following measures will limit the consequences of the accident in two ways:

1. Reactor trip and borated water injection complement void formation in causing rapid reduction of power to a residual level corresponding to fission product decay heat, and

2. Injection of borated water provides heat transfer from the core and prevents excessive cladding temperatures.

d. Release Inputs

The core inventory of the radionuclide groups utilized for this event is based on RG 1.183, Regulatory Position 3.1, at 102% of core thermal power and is provided as Table 15C-1. The source term represents end of cycle conditions assuming enveloping initial fuel enrichment and an average core burnup of 45,000 MWD/MTU.

For the first 24 hours, the containment is assumed to leak at a rate of 0.15% of the containment air per day. Per RG 1.183, Regulatory Position 3.7, the primary containment leakage rate is reduced by 50% at 24 hours into the LOCA to 0.075%/day based on the post-LOCA primary containment pressure history.

The ESF leakage to the auxiliary building is assumed to be 48 gpd based upon two times the current value of 24 gpd. The temperature of the leakage is based on the sump temperature at and after the time recirculation begins (255°F in the sump at the time recirculation begins). The leakage is assumed to start at 26 minutes into the event and continue throughout the 30-day period. This portion of the analysis assumes that all of the non-particulate iodine available for release is released from the leaked liquid. Based on sump pH history and pH control (pH is greater than 7 at the time recirculation begins), the iodine in the sump solution is assumed to all be in nonvolatile iodide or iodate form during the time of interest for this analysis. This analysis conservatively assumes a 10% iodine release with a chemical form of 97% elemental and 3% organic.

The ECCS backleakage to the RWST is assumed to be 0.9595 gpm. The leakage is assumed to start at 26 minutes into the event when recirculation starts and continue throughout the 30-day period. Note that based on the leakage rate and the size of the piping, the leakage would not reach the RWST for an extended period of time after recirculation begins. This time period is conservatively not credited for determining when the leakage reaches the RWST (i.e., the leakage is assumed to reach the RWST instantaneously). Based on sump pH history and pH control (pH is greater than 7 at the time recirculation begins), the iodine in the sump solution is assumed to all be in nonvolatile iodide or iodate form during the time of interest for this analysis.

Based upon the initial RWST pH of 7.1 at the start of recirculation, and based on information provided in NUREG-5950, it is expected that no elemental iodine will be regenerated in the RWST. However, for this analysis it was conservatively assumed that 1% of the particulate iodine would be converted to elemental iodine in the RWST. This conversion fraction is conservatively assumed to exist throughout the event even though the pH of the RWST would increase during the course of the event.

The elemental iodine generated in the RWST is assumed to become volatile and partition between the liquid and vapor space in the RWST based upon the temperature dependent partition coefficient for elemental iodine as presented in NUREG-5950. The particulate portion of the leakage is assumed to be retained in the liquid phase of the RWST since no boiling occurs in the RWST. The release of the activity from the vapor space within the RWST is calculated based upon the displacement of air by the incoming leakage and the expansion due to the daily heating and cooling cycle of the contents (both air and liquid) of the RWST. The average daily temperature swing of 18.2°F is applied for every 24-hour period for 30 days and no credit is taken for daily cooling. The final iodine release rate from the RWST is implemented via an adjustment to the leakage flow rate from the containment sump, which is applied to the entire iodine inventory in the containment sump, then released directly to the environment. The adjusted release rate is determined as follows:

$$Adjusted\,release\,rate = \left[\frac{(Leaked\,Volume\,x\,Iodine\,Fraction)\,/\,RWST\,Liquid\,Volume}{Partition\,Coefficient\,(I_2)}\right] x\,Air\,Flow\,Rate$$

where:

Iodine Fraction = 0.01 (Elemental Iodine fraction available for release from the leaked water)

RWST Liquid Volume = Time dependent RWST liquid volume

Partition Coefficient (I_2) = Temperature dependent elemental iodine partition coefficient

Air Flow Rate = Time dependent air flow from RWST based on expansion and displacement

The adjusted release rate presented in Table 15.6-8 is then applied to the entire iodine inventory in the containment sump.

Containment purge is also assumed coincident with the beginning of the LOCA. Since the purge is isolated prior to the initial release of fission products from the core at 30 seconds, only the initial RCS activity (at an assumed 1.0 microcuries per gram DE I-131 and 100/E-bar gross activity) is available for release via this pathway. The release is modeled as an unfiltered release for 5 seconds until isolation occurs.

e. Transport Inputs

During the LOCA event, the activity collected by the secondary containment is assumed to be a filtered ground level release from the plant vent. The activity that bypasses the secondary containment is identified as being leaked via a ground level release from the containment without filtration. The activity from the ECCS leakage enters the secondary containment and is released to the environment via the plant vent after filtration. The activity from the RWST is modeled as an unfiltered ground level release from the RWST.

For this event, the Control Room ventilation system cycles through two modes of operation:

- Initially the ventilation system is assumed to be operating in normal mode. The air intake during this mode is 1000 cfm of unfiltered fresh air.
- After the start of the event, the Control Room normal air intake is isolated due to a high containment pressure signal. A 30-second delay is conservatively applied to account for the time to reach the signal, the diesel generator start time and damper actuation and positioning time. After isolation of the Control Room normal air intake, the air flow distribution consists of 600 cfm of filtered makeup flow through the worst of the two emergency intakes, 150 cfm of unfiltered inleakage and 390 cfm of filtered recirculation flow.
- The Control Room ventilation filter efficiencies that are applied to the filtered makeup and recirculation flows are 99% for particulates/aerosols, and 95% for elemental and organic iodine.

f. LOCA Removal Inputs

Reduction of the airborne radioactivity in the containment by natural deposition is credited. A natural deposition removal coefficient for elemental iodine is calculated per SRP 6.5.2 as 2.23 hr^{-1}. This removal is credited in the sprayed and unsprayed regions. A natural deposition removal coefficient of 0.1 hr^{-1} is assumed for all aerosols in the unsprayed region. No natural deposition removal of aerosols is credited in the sprayed regions. No removal of organic iodine by natural deposition is assumed.

Containment spray provides coverage to 85.4% of the containment. Therefore, the Seabrook containment building atmosphere is not considered to be a single, well-mixed volume. A mixing rate of two turnovers of the unsprayed region per hour is assumed.

The maximum decontamination factor (DF) for the elemental iodine spray removal coefficient is 200 based on the maximum airborne elemental iodine concentration in the containment. The time for the containment sprays to reach an elemental iodine decontamination factor of 200 was determined by running a containment leakage case without environment leakage paths. Radioactive decay and natural deposition of iodine were conservatively left on as removal mechanisms contributing to the decontamination factor. Due to mixing between the sprayed and unsprayed regions of containment, the iodine activity in both containment regions was included in the determination of the time required to reach a decontamination factor of 200. The decontamination factor for elemental iodine reaches 200 at just over 2.92 hours.

The particulate iodine removal rate is reduced by a factor of 10 when a DF of 50 is reached. Based upon the calculated iodine aerosol removal rate of 5.75 hr^{-1}, the time of a DF of 50 is computed with the same model used to determine the elemental iodine DF of 200. The time for containment spray to produce an aerosol decontamination factor of 50 with respect to the containment atmosphere is just over 3.56 hours.

Filter removal in the Control Room Emergency Mode is simulated using conservative assumptions based on plant design data as listed in Table 15.6-16.

g. Radiological Consequences

The atmospheric dispersion factors (*X/Qs*) used for this event for the Control Room dose are based on the postulated release locations and the operational mode of the control room ventilation system. These *X/Qs* are summarized in Table 2R-2 and Table 2R-3.

Three pathways for unfiltered inleakage to the control room were considered; inleakage via the diesel building, inleakage via the primary control room entrance (double air lock configuration), and inleakage via the emergency fire exit (two doors in series). A value of 10 cfm is typically assumed for door leakage for normal ingress/egress. However, this flow would be reduced or eliminated by a two-door vestibule. It was conservatively assumed that 20 cfm of total door leakage occurs via the most limiting door. The *X/Qs* for the fire exit are always more limiting than those for the primary control room entrance; therefore, all of the unfiltered inleakage via the doors was assumed to occur at the fire exit. For most release locations, the *X/Qs* for the fire exit are more limiting than the *X/Qs* for the diesel building inleakage. For these cases, the fire exit was considered as a separate path for unfiltered inleakage. In cases where the diesel building is more limiting than the fire exit, all of the unfiltered inleakage was assumed to enter via the diesel building.

For the EAB and LPZ dose analysis, the *X/Q* factors for the appropriate time intervals are used. These *X/Q* factors are provided in Appendix 2Q.

The radiological consequences of the design basis LOCA are analyzed using the RADTRAD-NAI code and the inputs/assumptions previously discussed. In addition, the MicroShield code, Version 5.05, Grove Engineering, is used to develop direct shine doses to the Control Room. MicroShield is a point kernel integration code used for general-purpose gamma shielding analysis.

The post accident doses are the result of four distinct activity releases:

- Containment leakage
- ESF system leakage into the Primary Auxiliary Building
- ESF leakage into the RWST
- Containment Purge at event initiation

The dose to the Control Room occupants includes terms for:

1. Contamination of the Control Room atmosphere by intake and infiltration of radioactive material from the containment and ESF.
2. External radioactive plume shine contribution from the containment and ESF leakage releases. This term takes credit for Control Room structural shielding.
3. A direct shine dose contribution from the Containment's contained accident activity. This term takes credit for both Containment and Control Room structural shielding.

4. A direct shine dose contribution from the activity collected on the Control Room ventilation filters.

As shown in Table 15.6-20, the sum of the results of all four activity releases for EAB dose, LPZ dose, and Control Room dose are all within the appropriate regulatory acceptance criteria.

15.6.6 BWR Transients

Not applicable to Seabrook.

15.6.7 References

1. Federal Register, "Emergency Core Cooling Systems: Revisions to Acceptance Criteria," V53, N180, pp. 35996-36005, September 16, 1988.
2. U. S. Nuclear Regulatory Commission 1975, "Reactor Safety Study - An Assessment of Accident Risks in U. A. Commercial Nuclear Power Plants" WASH-1400, NUREG-75/014.
3. Bordelon, F. M.; Massie, H. W.; and Zordan, T. A. "Westinghouse ECCS Evaluation Model - Summary," WCAP-8339, July 1974.
4. Foster, J.P., et al., Westinghouse Improved Performance Analysis and Design Model (PAD 4.0), WCAP-15063-P-A, Revision 1, with Errata, 2000.
5. USNRC Regulatory Guide 1.157, "Best-Estimate Calculations of Emergency Core Cooling System Performances," May 1989.
6. Boyack, B., et al., "Qualifying Reactor Safety Margins: Application of Code Scaling Applicability and Uncertainty (CSAU) Evaluation Methodology to a Large Break Loss-of-Coolant-Accident," NUREG/CR-5249.
7. Letter, R.C. Jones (USNRC) to N.J. Liparulo (W), "Acceptance for Referencing of the Topical Report WCAP-12945 (P), Westinghouse Code Qualification Document for Best Estimate Loss-of-Coolant Analysis," June 28, 1996.
8. Bajorek, S.M., et al., 1998, "Westinghouse Code Qualification Document for Best Estimate Loss of Coolant Accident Analysis," WCAP-12945-P-A (Proprietary), Volume 1, Revision 2, and Volumes II-V, Revision 1, and WCAP-14747 (Non-Proprietary).
9. Branch Technical Position CSB 6-1, "Minimum Containment Pressure Model for PWR ECCS Performance Evaluation," July 1981.
10. Deleted

11. Bordelon, F. M., and Murphy, E. T., "Containment Pressure Analysis Code (COCO)," WCAP-8327 (Proprietary) and WCAP-8326 (Non-Proprietary), June 1974.

12. Deleted

13. "Westinghouse ECCS - Evaluation Model Sensitivity Studies," WCAP-8341 (Proprietary) and WCAP-8342 (Non-proprietary), July 1974.

14. Salvatori, R., "Westinghouse ECCS - Plant Sensitivity Studies," WCAP-8340 (Proprietary) and WCAP-8356 (Non-proprietary), July 1974.

15. Johnson, W. J.; Massie, H. W.; and Thompson, C. M. "Westinghouse ECCS Four Loop Plant (17x17) Sensitivity Studies", 5 WCAP-8586-P-A (Proprietary) and WCAP-8566-A (Non-Proprietary), July 1975.

16. Meyer, P. E., "NOTRUMP - A Nodal Transient Small Break and General Network Code", 5 WCAP-10079-P-A (Proprietary) and WCAP-10080-A (Nonproprietary), August 1985.

17. Lee, N. et al., "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code," WCAP-10054-P-A (Proprietary) and 10081-A (Nonproprietary), August 1985.

18. "Generic Evaluation of Feedwater Transients and Small Break Loss-of-Coolant Accidents in Westinghouse - Designed Operating Plants," NUREG-0611, January 1980.

19. Bordelon, F. M. et al., "LOCTA-IV Program: Loss-of-Coolant Transient Analysis," WCAP-8305, June 1974, WCAP-8301 (Proprietary), June 1974.

20. "Clarification of TMI Action Plan Requirements", NUREG-0737, November 1980.

21. NRC Generic Letter 83-35 from D. G. Eisenhut, "Clarification of TMI Action Plan Item II.K.3.31", November 2, 1983.

22. Rupprecht, S. D. et al., "Westinghouse Small Break LOCA ECCS Evaluation Model Generic Study with the NOTRUMP Code," WCAP-11145-P-A (Proprietary), and WCAP-11372-A (Non-Proprietary), October 1986.

23. U.S. Nuclear Regulatory Commission (USNRC) Standard Review Plan (SRP) 15.6.3, "Radiological Consequences of Steam Generator Tube Failure (PWR)," NUREG-0800, Revision 2, July 1981.

24. EPRI NP-1850, Volume 1, Rev. 4, "RETRAN-02 - A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems, Volume 1: Theory and Numerics (Rev. 4)", Electric Power Research Institute, November 1988.

25. NUREG-0409, "Iodine Behavior in a PWR Cooling System Following a Postulated Steam Generator Tube Rupture Accident," A. K. Postma and S. Tam, January 1978.

26. EPRI-4787, "Tests of Steam Generator Transient Response to Scenarios Involving Steam Generator Tube Ruptures and Stuck-Open Safety Relief Valves," K. Garbett, O. J. Mendler, G. C. Gardner, R. Garnsey, and M. Y. Young, May 1987.

27. Deleted

28. Deleted

29. WCAP-16215-P, "Seabrook Station, Stretch Power Uprate Project NSSS Licensing Report," February 2004.

30. WCAP-10054-P-A, Addendum 2, Revision 1, "Addendum to the Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code: Safety Injection into the Broken Loop and COSI Condensation Model," July 1997.

31. Deleted

32. Deleted

33. Deleted

34. Deleted

35. Deleted

36. WCAP-11397-P-A, "Revised Thermal Design Procedure," A. J. Friedland and S. Ray, April 1984.

37. WCAP-7907-P-A, "LOFTRAN Code Description," T.W.T. Burnett, et al., April 1984.

38. "Seabrook Station Stretch Power Uprate License Amendment Request 04-03 (LAR 04-03)," March 17, 2004.

39. NAESCO letter to USNRC, "Steam Generator Tube Rupture Operator Response Times", NYN-94056, May 12, 1994.

40. WCAP-13247, "Report on the Methodology for the Resolution of the Steam Generator Tube Uncovery Issue", dated March 1992.

41. WCAP-14882-P-A, "RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analyses," Hoegel, D.S., et al., April 1999.

15.7 RADIOACTIVE RELEASE FROM A SYSTEM OR COMPONENT

15.7.1 Radioactive Gaseous Waste System Leak or Failure

15.7.1.1 Identification of Causes and Accident Description

The most limiting waste gas accident is defined as an unexpected and uncontrolled release to the atmosphere of the radioactive xenon and krypton fission gases that are stored in the five carbon delay beds. Although there is no credible mechanism by which this could occur, it is considered a limiting fault since it includes the potential for significant amounts of radioactive releases. The Radioactive Gaseous Waste System (RGWS) is discussed in Section 11.3.

Each low pressure (0-2 psig) carbon delay bed in the RGWS is designed to provide 17 hours of krypton (Kr) delay and 12 days of xenon (Xe) delay, which results in a total system delay of 85 hours for krypton and 60 days for xenon. It is assumed that there is no delay time of the noble gases before they reach the carbon delay beds. The gas volume of each carbon delay bed is approximately 20.4 ft^3. The decontamination factor of the beds is discussed in Subsection 12.2.1. The maximum expected inventory in the five carbon delay beds is shown on Table 12.2-27, and is based on 1 percent failed fuel. The design basis is further discussed in Subsection 12.2.1.

The RGWS is designated NNS, nonseismic Category I, in accordance with ANSI/ANS 51.1-1983 and NRC Regulatory Guide 1.143. Also, the system is designed to withstand a hydrogen explosion. The portion of the Waste Processing Building (WPB) which houses the RGWS is seismic Category I. It should be noted that the gas chillers, iodine guard beds, dryers, carbon delay beds and some valves were designed and fabricated as safety Class 3 seismic Category I, prior to the declassification of the RGWS to NNS.

15.7.1.2 Analysis of Effects and Consequences

In the event of a pipe or carbon delay bed failure, noble gases would be released from the carbon delay beds, since the RGWS operates at a slight positive pressure. The quantity of radioactivity released would depend on the failure location, but in all cases would be a small fraction of the total system inventory.

The sequence of events following this failure is shown in Table 15.7-1. The RGWS process stream is monitored continuously for radioactivity and oxygen upstream of the carbon delay beds. Outleakage of hydrogen would be detected and alarmed by the hydrogen monitors in the Waste Processing Building.

The ventilation system for the areas housing the carbon delay beds operates continuously. The WPB ventilating system would remove the radioactive waste gases and exhaust them to the atmosphere via the plant unit vent (see Subsection 9.4.4).

For the conservative analysis, it is assumed that 100 percent of the carbon delay bed's inventory is released to the environment in a two-hour period.

For the realistic case, it is assumed that a failure occurs upstream of the first carbon delay bed resulting in depressurization of the carbon delay beds and release of their inventory to the WPB atmosphere. Should this occur, the operator would take the actions described in Table 15.7-1, and the WPB ventilating system would operate as described in Subsection 9.4.4.

A leak in the hydrogen surge tank in the RGWS is considered an infrequent incident, since it could occur during the lifetime of the plant. The hydrogen surge tank has a volume of 44 ft^3, operates at 150 psig, and is nonnuclear safety class and nonseismic Category I. The WPB ventilating system also includes a separate exhaust system to purge the area housing the hydrogen surge tank, if the hydrogen gas level in the area approaches the low flammable limit. The Purge Exhaust System would dilute the abnormal hydrogen gas release in the hydrogen surge tank cubicle.

15.7.1.3 Radiological Consequences using Alternate Source Term

a. Background

This event involves a major rupture of one of the Radioactive Gaseous Waste System (RGWS) components. This analysis assumes that the ruptured RGWS component contains an inventory equivalent to the activity limit specified in Table 15.7-3. The entire source term is applied to this RGWS component at the beginning of the event. The leak rate from the RGWS to the environment is conservatively modeled to be very high to simulate a major tank rupture, which releases essentially all of the activity to the environment within 2 hours. No hold-up, dilution or filtration of the tank release is assumed. The impact of the release is then computed as it disperses to the offsite receptors. The dose to Control Room operators is computed as the release is modeled to be treated by the Control Room Air Conditioning and Emergency Cleanup system during the 30-day period following the accident.

b. Compliance with RG 1.183 Regulatory Positions

RG 1.183 does not provide direct guidance relative to the Waste Gas system failures. Therefore, this analysis will rely primarily upon the current UFSAR licensing basis for guidance on performance of this event.

c. Methodology

The dose assessment model releases the above-prescribed inventory from the RGWS at a high rate of release. Per the existing analysis assumptions, the contents are released to the environment without any hold up, dilution or filtration over a 2 hour period.

For this event, the Control Room ventilation system cycles through three modes of operation:

- Initially, the ventilation system is assumed to be operating in normal mode. The air flow distribution during this mode is 1000 cfm of unfiltered fresh air make up and an assumed value of 300 cfm of unfiltered inleakage.
- After the start of the event, the Control Room is isolated due to a high radiation reading in the Control Room ventilation system. A 30-second delay is applied to account for diesel generator start time, damper actuation time, instrument delay, and detector response time. After isolation, the air flow distribution consists of 600 cfm of unfiltered makeup flow from the outside, 300 cfm of unfiltered inleakage, and 390 cfm of filtered recirculation.
- The Control Room ventilation filter efficiencies that are applied to the filtered makeup and recirculation flows are 99% for particulate, 95% for elemental iodine, and 95% for organic iodine.

d. Radiological Consequences

The release-receptor point locations are chosen to model the distance from the release point to the Control Room intake. The X/Q values for the various combinations of release points and receptor locations are presented in Tables 2R-2 and 2R-3.

For the EAB and LPZ dose analyses, the X/Q factors are provided in Appendix 2Q.

Reg. Guide 1.183 does not provide any requirement or dose limits for a RGWS failure; therefore, the acceptance criteria are set by the current Seabrook Licensing basis. Therefore, the off-site dose acceptance criteria are established as 10% of the 10 CFR 50.67 limits. The control room dose limits are specified in 10 CFR 50.67. Therefore, the dose limits are:

Area	Dose Criteria	
EAB	2.5 rem TEDE	(for the worst two hour period)
LPZ	2.5 rem TEDE	(for 30 days)
Control Room	5 rem TEDE	(for 30 days)

The radiological consequences of the RGWS failure event are analyzed using the RADTRAD-NAI code and the inputs and assumptions previously discussed. As shown in Table 15.7-4, the radiological consequences of the Radioactive Gaseous Waste System failure are all within the appropriate acceptance criteria.

15.7.1.4 Conclusions

The doses which have been calculated for the radioactive gaseous waste system accident are below regulatory limits

15.7.2 Radioactive Liquid Waste System Leak or Failure (Release to Atmosphere)

This analysis evaluates the radiological consequences of the release to the atmosphere of radioactive fission gases, resulting from an unexpected and uncontrolled release of radioactive liquids that are stored or transferred in waste systems, to determine that they are small fractions of the 10 CFR 100 guideline values. The primary sources evaluated consisted of the Liquid Waste System that normally contains and processes the waste liquid before final disposal, as discussed in Section 11.2, and the below-listed systems that may store or handle a radioactive liquid:

- Boron Recovery System (Subsection 9.3.5)
- Steam Generator Blowdown System (Subsection 10.4.8)
- Equipment and Floor Drain System (Subsection 9.3.3)
- Chemical and Volume Control System (Subsection 9.3.4).

A leak or failure of a component in one of the above systems can release some fission gases and/or iodine-contaminated liquids into the building housing the particular component. The impact of the releases is evaluated in this section. For the purpose of these analyses, it is assumed that at the time of the leak or failure, there exist excessive fuel cladding defects on one (1.0) percent with no decay, as discussed in Subsection 11.1.1 and Section 12.2.

15.7.2.1 Identification of Causes and Accident Description

The above-mentioned systems are located in the Primary Auxiliary Building (PAB) and the Waste Processing Building (WPB), both seismic Category I structures.

For the purpose of release to atmosphere, the components that contain undergasified liquid are of major significance. The liquids in the Chemical and Volume Control System and the Boron Recovery System before degasification contain significant fission-gas inventories (see Section 12.2). There are also significant fission gas concentrations in the boron waste storage tank and spent resin sluice tank. In all these systems, components such as valve-stems, pump-seals, etc., are designed with double seals to minimize leakage. At strategic locations, manual bypass valves have been provided in case the main control valve fails and goes under repair. Moreover, the construction materials of the components (as discussed in specific sections) are of high allowable stresses and proper corrosion resistance, or allowance is made.

In spite of the safety features mentioned above, some design basis failures are postulated due to operator error, instrumentation/controls failure, or seismic loads beyond design limits. The leakage of a valve or pump-seal can release only minimal amounts of radioactivity which can be cleaned up and terminated quickly. The rupture of a pipe or tank can release considerable amounts of fission gases into the buildings where they are located. All the liquid-containing tanks (outside Containment) are listed in Table 15.7-15. The tanks which may contain significant quantities of fission gases, along with the appropriate table in Subsection 12.2.1, which gives their radionuclide inventory, are listed below.

- Letdown Degasifier (Subsection 9.3.3) in PAB - Table 12.2-5
- Primary Drain Tank Degasifier (Subsection 9.3.5) in WPB - Table 12.2-13
- Boron Waste Storage Tank (Subsection 9.3.5) in WPB - Table 12.2-11
- Spent Resin Sluice Tank (Section 11.4) in WPB - Table 12.2-15

For the purpose of this analysis, the rupture of only one tank (irrespective of seismic design category) is considered at a time. The backup tank is assumed to be available for the process. All the systems considered in this section are provided with Overpressure relief protection. Therefore, the rupture of a tank or adjacent pipe due to Overpressure is possible only if the relief capability becomes inoperative. Some of the system components are nonseismic Category I and could, therefore, rupture in the event of a major earthquake. Unexpected corrosion beyond the design allowance could also cause failure. However, due to the safety features, plant inspection and maintenance provided, it is highly improbable that failure of a tank will occur more than once during the expected plant life. Therefore, the frequency of occurrence of such a leak or failure is not of any significance.

15.7.2.2 Analysis of Effects and Consequences

All the tanks of the Liquid Waste System are contained within shielded enclosures, the floors of which slope toward drains. Some of the piping containing radioactive liquid waste is routed outside the shielded enclosures, but inside the buildings. Upon failure of a liquid waste system component, the majority of the liquid will be contained within the enclosure or will drain to the building sump. The liquid will flow into the local drains for eventual processing by one of the evaporators. There is a backup capacity available in the evaporators from the Boron Recovery System.

The building vents and area radiation (airborne) levels are monitored and high activity or radiation level is alarmed (see Subsection 12.3.4).

Upon receiving an alarm, the operator will verify the cause of the alarm. The following sequence of events will ensue:

a. The affected area will be evacuated of unnecessary personnel

b. Accessibility to the affected area will be limited under proper protective measures

c. Proper respirators will be made available to the operating personnel

d. The radioactive fission gases released in the buildings will be removed by the ventilation system, filtered and cleaned up to the capacity of the system, as described in Subsections 9.4.3 and 9.4.4.

The consequences of final releases to the atmosphere are discussed in Subsection 15.7.2.3. The operation of the failed system/component will not be continued without cleaning the area and repairing the affected component. The system is under administrative control during this period.

15.7.2.3 Radiological Consequences using Alternate Source Term Methodology

a. Background

This event involves a major rupture of one of the Radioactive Liquid Waste System (RLWS) components. This analysis considers the two separate cases of the rupture of either the Boron Waste Storage Tank or the Letdown Degasifier. This analysis assumes that the ruptured RLWS component contains an inventory equivalent to the activity limit specified in Table 15.7-8. The entire release source term is applied to this RLWS component at the beginning of the event. The leak rate from the RLWS to the environment is conservatively modeled to be very high to simulate a major tank rupture, which releases essentially all of the activity to the environment within 2 hours. No hold-up, dilution or filtration of the tank release is assumed. The impact of the release is then computed as it disperses to the offsite receptors. The dose to Control Room operators is computed as the release is modeled to be treated by the Control Room Air Conditioning and Emergency Cleanup system during the 30-day period following the accident.

b. Compliance with RG 1.183 Regulatory Positions

RG 1.183 does not provide direct guidance relative to the Liquid Waste system failures. Therefore, this analysis will rely primarily upon the current licensing basis for guidance on performance of this event.

c. Methodology

The dose assessment model releases the above-prescribed inventory from the RLWS at a high rate of release. Per the existing analysis assumptions, the contents are released to the environment without any hold up, dilution or filtration over a 2 hour period.

For this event, the Control Room ventilation system cycles through three modes off operation:

- Initially the ventilation system is assumed to be operating in normal mode. The air flow distribution during this mode is 1000 cfm of unfiltered fresh air make up and an assumed value of 300 cfm of unfiltered inleakage.
- After the start of the event, the Control Room is isolated due to a high radiation reading in the Control Room ventilation system. A 30-second delay is applied to account for diesel generator start time, damper actuation time, instrument delay, and detector response time. After isolation, the air flow distribution consists of 600 cfm of filtered makeup flow from the outside, 300 cfm of unfiltered inleakage, and 390 cfm of filtered recirculation flow.
- The Control Room ventilation filter efficiencies that are applied to the filtered makeup and recirculation flows are 99% for particulate, 95% for elemental iodine, and 95% for organic iodine.

d. <u>Radiological Consequences</u>

The release-receptor point locations are chosen to model the distance from the release point to the Control Room intake. The X/Q values for the various combinations of release points and receptor locations are presented in Tables 2R-2 and 2R-3.

For the EAB and LPZ dose analyses, the X/Q factors are provided in Appendix 2Q.

Reg. Guide 1.183 does not provide any requirement or dose limits for a RLWS failure. The dose limits for this event are based on 10 CFR Part 20. The acceptance criteria for this event is ≤ 100 mrem for offsite dose and ≤ 5 Rem for the control room.

The radiological consequences of the RLWS failure event are analyzed using the RADTRAD-NAI code and the inputs and assumptions previously discussed. As shown in Table 15.7-9, the radiological consequences of the Radioactive Liquid Waste System failure are all within the appropriate acceptance criteria.

15.7.2.4 <u>Conclusions</u>

The doses which have been calculated for the failure of either the boron waste storage tank or the letdown degasifier are within regulatory limits.

15.7.3 Postulated Radioactive Releases Due to Liquid Containing Tank Failures

15.7.3.1 Identification of Causes and Accident Description

A rupture of a liquid-containing tank or associated component inside the Containment would release radioactive liquid to the containment sump. The collected liquid would be processed in the Radioactive Liquid Waste Disposal Systems; i.e., the Boron Recovery System (Subsection 9.3.5) and the Liquid Waste Processing System (Section 11.2). The Containment has a steel-lined interior structure; therefore, there is no credible pathway for spilled fluid due to a ruptured tank in Containment to affect water in unrestricted areas. Thus, tanks in the Containment are not considered here.

All liquid containing tanks outside containment were evaluated for postulated radioactive releases due to liquid containing tank failures. The Waste Concentrate Tank is the bounding source term and is based on a condition of 1% failed fuel, as shown in Table 15.7-14. An accident involving the rupture of a tank (irrespective of seismic design category) resulting in leakage of liquid outside the tank is considered.

All the systems considered are provided with proper instrumentation, controls and over-pressure relief protection. If the pressure inside a tank rises above the design pressure of the tank due to controls not functioning properly or due to an operator error, pressure relief devices will relieve the overpressure. Therefore, the rupture of a tank due to overpressure is possible only if the relief devices are inoperative.

A seismic event beyond the design capacity of the tank, or unanticipated corrosion beyond the design corrosion allowance, could also result in failure. The frequency of a possible rupture of a specific tank is not anticipated to be more than once during the expected plant life. As such, the frequency of occurrence is not significant.

The failure of a component associated with a tank could, at the worst, evacuate the tank within the diked/cubicle area. Therefore, rupture of a tank resulting in complete evacuation of the tank is conservatively considered as the design basis.

Significant loading of the liquid waste management systems can only be caused by the failure of those tanks considered which have a large volume or a high concentration of radioactivity.

The refueling water storage tank, the waste concentrates tank, and the floor drain tank are considered for purposes of the analysis. The waste concentrates tank is the bounding source term. The wastes from the failed tanks will be disposed under controlled monitored conditions.

15.7.3.2 Analysis of Effects and Consequences

All the tanks considered are located in areas protected by concrete walls and floors (cubicles). The tank farm adjacent to the Waste Processing Building has some tanks. The tank farm is divided into two sections. These sections have concrete floors and surrounding dikes as high as the rupture level of the largest tank in a section. In one section, the largest tank is the refueling water storage tank and in the other, the reactor makeup water tank. The floors of the tank farm slope toward local drains. On rupture of a tank, the liquid gets drained and routed into the Liquid Waste Processing System. There is no flooding outside the diked areas. Control panel alarms alert the operator to liquid inside diked areas so that the liquid can be drained out under proper controlled conditions to suit the capacity of the drainage and processing equipment. Supply lines to the ruptured tank are then isolated by the operator, and all the leaked liquid is processed by the liquid waste system evaporators before final disposal (Section 11.2). Normal methods of disposal follow concentration and solidification. Effluents within the limits of 10 CFR 20 are discharged offshore via the circulating water tunnels. This is the only discharge into the surface waters. The failed tank is then under administrative control and the unfailed tank is under close supervision.

Because of the potential for cracks in concrete, no credit is taken for liquid retention by unlined building foundations. This means that there is infiltration into the ground until the spilled liquid has drained into the Liquid Waste Processing System. Therefore, for purposes of analysis, a conservative case of all spilled liquid of a tank seeping instantaneously into the ground is considered.

As described in Subsection 2.4.13.3, accidental liquid discharge seeping into the ground will not contaminate any well supplies in the area.

15.7.3.3 Radiological Consequences

The following analysis has been performed to determine the radiological consequences of the rupture of the worst case liquid radwaste tank.

The worst case tank failure was developed by considering the tank with the highest expected radionuclide level and individual radionuclide dose conversion factors. The spent resin sluice tank had the highest curie inventory but was not considered in the analysis. Due to the physical properties of the spent resin material, negligible quantities of the sludge would be able to diffuse through a hypothesized crack in the concrete wall and into the water table aquifer.

The waste concentrates tank had the second highest curie content, based on the conservative 1 percent failed fuel level, with the total activity equal to 1.5×10^3 μCi and liquid volume equal to 6.9×10^3 gallons. For conservatism, it is assumed that the waste concentrates tank has a tritium concentration of 1 μCi/ml.

Based on information presented in Subsection 2.4.13, the maximum rate of groundwater movement along a flow path moving southward from the southern portion of the site is 0.7 ft/day. Radionuclides released from the tank rupture are assumed to instantaneously enter the groundwater system and travel along the groundwater path to the adjoining marsh area. All radionuclides are assumed to travel at the groundwater flow rate with the exception of cesium and strontium isotopes. Since the shortest distance from the southern boundary of the site to the marsh is about 200 feet, it will require at least 290 days for a contaminant released at the site to reach the marsh. With this minimum decay time, all radionuclides with half-lives less than 22 days need not be considered in the dose calculations since they will have decayed to at least .01 percent of their original release concentrations enroute to the marsh. The cesium and strontium isotopes will travel considerably slower than the groundwater due to adsorption and ion exchange with the soil particles. The retardation factor associated with cesium and strontium isotopes traveling through soil columns is well established in literature (References 2, 3) and may be approximated by the following equation (Reference 4):

$$V_i = V_w/(1 + rK_d),$$

where:

V_i = The average velocity of the ionic species

V_w = The average velocity of the groundwater

r = The ratio of the weight of mineral to volume of water per unit volume of aquifer material

K_d = The distribution coefficient of the given ionic species for the prevailing conditions

The quantity $(1 + K_d)$ is referred to as the retardation factor.

The distribution coefficient (K_d) assumed in this evaluation (70 ml/g) is that value used for ^{90}Sr in Reference 5. This assumption is conservative in that cesium isotopes are more tightly bound by soil than strontium isotopes and will exhibit a larger distribution coefficient. Seabrook Station and the standard site used in Reference 5 are both coastal sites with similar soil parameters and groundwater flow rates.

With the above assumptions and parameters listed in Table 15.7-16, the average velocity of Sr and Cs isotopes is calculated to be 1.2×10^{-3} ft/day. The time required to travel 200 feet to the marsh area is 457 years. Based on 457 years decay time for cesium and strontium isotopes, and 290 days decay time for all other radionuclides released, specific nuclide concentrations at the marsh are calculated and listed in Table 15.7-17. These values are based on the assumption that 80 percent of the maximum liquid volume of the affected tank is released. No credit is taken for dilution in the groundwater or liquid retention by unlined building foundations or leakage barriers.

The marsh concentration of radioisotopes is subject to tidal flushing as well as wind and wave action into Hampton Harbor. The discharge into the marsh will be quickly diluted and mixed in the intertidal zone or tidal prism of Hampton Harbor. A value of 2.24×10^{8} ft^3 has been conservatively used in determining the extent of radionuclide dilution, since no credit is taken for dilution within the tidal prism of the marsh. Within the entire Hampton Harbor and estuary, the volume of the tidal prism is approximately 4.70×10^{8} ft^3.

Water is lost from the entire Hampton estuary at an average rate of 9850 ft^3/sec. Expressed on a percentage basis, about 88 percent of the estuary volume leaves and returns on each ebb and flood tide cycle. At ebb slack tide, the estuarine residual is approximately 12 percent of the total volume of the basin. These figures indicate that the Hampton Harbor estuary exhibits substantial tidal exchange rate under natural conditions.

Radionuclide concentrations in Hampton Harbor can be found in Table 15.7-17, and have been used to calculate doses to individuals by the ingestion of finfish and invertebrates. Doses have been calculated based on methodology and dose conversion factors, bioaccumulation factors and maximum usage factors delineated in Regulatory Guide 1.109, Revision 1. The highest organ dose was to the lower large intestine of an adult and was calculated to be 18.2 mrem. The adult total body dose is 1.4 mrem.

15.7.3.4 <u>Conclusions</u>

The radioactive liquid release from a tank rupture will not result in any uncontrolled surface release. The liquid will be processed in the Liquid Waste and/or Boron Recovery System and the final effluent will be controlled and monitored before discharging into the circulating water tunnels or disposed offsite. The seepage of tank contents through cracks in the concrete cubicles will not significantly impact any potable water supply or possible ingestion pathways.

15.7.4 Fuel Handling Accident

15.7.4.1 Identification of Causes and Accident Description

Subsequent to plant start-up, a Licensing Amendment Request (LAR), LAR 94-06, "Revision to Technical Specification 3.9.4," was submitted and accepted by the staff. LAR 94-06 proposed two changes to the Seabrook Station Technical Specifications that address containment building penetrations. The first change is to allow the use of alternate closure methodologies for containment building penetrations during core alterations or movement of irradiated fuel within containment. The second change would allow both personnel airlocks to be open during core alterations or movement of irradiated fuel within containment. Consequently, the most limiting fuel handling accident is defined as the dropping of a spent fuel assembly within an open containment, resulting in the rupture of the cladding of all the fuel rods in the assembly, despite administrative controls and physical limitations imposed on fuel handling operations (see UFSAR Subsection 9.1.4). This potential fuel handling accident is considered an ANS Condition IV event, a limiting fault, since it includes the potential for significant amounts of radioactive releases. All refueling operations are conducted in accordance with prescribed procedures.

Dropping or damaging an assembly within the Fuel Storage Building (FSB) is another postulated accident addressed in this analysis. Dropping an assembly within the FSB is evaluated assuming a minimum of 23 feet of water above the assembly is available for iodine scrubbing (effective iodine decontamination factor of 100) prior to release of fuel assembly gap activity to the FSB atmosphere. Damaging an assembly (i.e., during fuel assembly maintenance or inspection) assumes a minimum of 10 feet of water above the assembly release point (effective iodine DF of 37) prior to release of fuel assembly gap activity to the FSB atmosphere.

15.7.4.2 Analysis of Effects and Consequences

a. Method of Analysis

The following assumptions are postulated in the calculation of the radiological consequences of a fuel handling accident:

1. The accident occurs 80 hours following reactor shutdown, the earliest time when spent fuel would be first moved from the reactor vessel.
2. The accident results in the rupture of the cladding of all fuel rods in the assembly.
3. The damaged assembly was the one operating at the highest power level in the core region to be discharged. The power in this assembly, and the corresponding fuel temperatures, establish the total fission product inventory and the fraction of this inventory present in the fuel pellet-cladding gap at the time of reactor shutdown.

4. The fuel pellet-cladding gap inventory of fission products is released to the refueling cavity or spent fuel pool water at the time of the accident.

5. Refueling cavity or spent fuel pool water will retain a large fraction of the gap activity of halogens by virtue of their solubility and hydrolysis. Noble gases are not retained by the water as they are not subject to hydrolytic reactions.

Additional assumptions are given in Table 15.7-18.

b. Fission Product Inventories

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 parameters used for the calculation of the fission product inventory of the highest rated assembly to be discharged are summarized in Table 15.7-19. Table 15.7-20 shows the activity of the highest rated fuel assembly at the time of reactor shutdown and after 80 hours decay.

The conservative parameters are based on Regulatory Guide 1.25, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors," dated March 23, 1972.

c. Iodine Decontamination Factors

An experimental test program (Reference 6) was conducted by Westinghouse to evaluate the extent of removal of iodine released from a damaged irradiated fuel assembly. Iodine removal from the released gas takes place as the gas rises through the solution in the spent fuel pool to the pool surface. The extent of 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.

To obtain all the necessary information regarding this mass transfer process, a number of small-scale tests were conducted, using trace iodine and carbon dioxide in an inert carrier gas. Iodine testing was performed at the design basis solution conditions (temperature and chemistry), and data were collected for various bubble diameters and solution depths. This work resulted in the formulation of a mathematical expression for iodine decontamination factor in terms of bubble size and bubble rise time.

Similar tests were conducted with carbon dioxide in an inert carrier, except that the solution temperature and chemistry were patterned after that of a deep pool where large-scale tests were also performed with carbon dioxide. The small-scale carbon dioxide tests also resulted in a mathematical expression for decontamination factor in terms of bubble size and bubble rise time through the solution.

To complete the experimental program, a full-size fuel assembly simulator was fabricated and placed in a deep pool for testing, where gas released would be typical of that from the postulated damaged assembly. Tests were conducted with trace carbon dioxide in an inert carrier gas, and overall decontamination factors were measured as a function of the total gas volume released. These measurements, combined with the analytical expression derived from small-scale tests with carbon dioxide, permitted an in situ measurement of both the effective bubble diameter and rise time, both as a function of the volume of gas released. Having measured the characteristics of large-scale gas releases, the decontamination factor for iodine was obtained, using the analytical expression from small-scale iodine testing.

Decontamination factor = $73\ e^{0.313\ t/d}$

where:

t = rise time

d = effective bubble diameter

The overall test results clearly indicate that iodine will be readily removed from the gas rising through the spent fuel pool solution, and that the efficiency of removal will depend on the volume of gas released instantaneously from the full void space.

With consideration given to the total quantity of gas released from a full assembly, that is, 6.9 SCF for the 17x17 array, the pool decontamination factor for iodine is indicated to be a minimum of 589 for a 23 foot depth and 181 for a 10 foot depth. In the conservative analyses, a lower decontamination factor is selected to provide for reasonable deviation in the factors which control iodine adsorption by the pool water. For the dropped assembly analysis, an overall effective iodine decontamination factor of 100 is used, as discussed in Regulatory Guide 1.25. In both the realistic and conservative evaluations, a decontamination factor of one is used for noble gas isotopes. The activity released from the spent fuel pool surface or refueling canal by isotope is shown in Table 15.7-21.

Potential damaging of a fuel assembly in the FSB during maintenance or inspection could occur with a minimum of 10 feet of water above the assembly. As discussed above the calculated iodine DF for 10 feet of water is 181 using the analytical expression above. For the analysis described below for the potential damaging of an assembly while suspended with only 10 feet of water above the assembly, a reduced iodine DR of 37 is used to provide for reasonable deviation in the factors which control iodine adsorption by the pool water. This value was determined by appropriate normalization of the experimental test case values.

15.7.4.3 Radiological Consequences using Alternate Source Term Methodology

a. Background

This event consists of the drop of a single fuel assembly either in the Fuel Storage Building (FSB) or inside of Containment. All of the fuel rods in a single fuel assembly are damaged. In addition, a minimum water level of 23 feet is maintained above the damaged fuel assembly for both the containment and FSB release locations.

This analysis bounds dropping a fuel assembly either inside the containment (with the personnel hatch open) or inside the FSB. Although filtration can be credited for the accident in the FSB there is sufficient margin to allow the analysis to be performed without crediting FSB filters. The source term released from the overlying water pool is the same for both the FSB and the containment cases. RG 1.183 imposes the same 2-hour criteria for the direct unfiltered release of the activity to the environment for either location. With the containment personnel hatch assumed open and filtration of the Fuel Storage Building exhaust not credited, the analyses are essentially identical for either the containment or the FSB release point except that the dispersion factors from the containment are slightly greater than the dispersion factors from the FSB.

To ensure that this analysis bounds the FHA in Containment or in the Fuel Storage Building, the most limiting combination of release point dispersion factors (X/Q) from the containment personnel hatch or the Fuel Storage Building release points is used. Use of the most limiting dispersion factors with no credit for FSB filtration assures the event results bound a Fuel Handling accident in either the containment or the Fuel Storage Building.

b. Compliance with RG 1.183 Regulatory Positions

The FHA dose consequence analysis is consistent with the guidance provided in RG 1.183 Appendix B, "Assumptions for Evaluating the Radiological Consequences of a Fuel Handling Accident," as discussed below:

1. Regulatory Position 1.1 – The amount of fuel damage is assumed to be all of the fuel rods in a single fuel assembly.
2. Regulatory Position 1.2 – The fission product release from the breached fuel is based on Regulatory Positions 3.1 and 3.2 of RG 1.183. A listing of the FHA source term is provided in Table 15C-4. The gap activity available for release is specified by Table 3 of RG 1.183. This activity is assumed to be released from the fuel instantaneously.
3. Regulatory Position 1.3 – The chemical form of radioiodine released from the damaged fuel into the spent fuel pool is assumed to be 95% cesium iodide (CsI), 4.85% elemental iodine, and 0.15% organic iodide. The cesium iodide is assumed to completely dissociate in the spent fuel pool resulting in a final iodine distribution of 99.85% elemental iodine and 0.15% organic iodine.
4. Regulatory Position 2 – A minimum water depth of 23 feet is maintained above the damaged fuel assembly. Therefore, an overall decontamination factor of 200 is applied to the elemental and organic iodine based upon the composition specified in Regulatory Position 1.3
5. Regulatory Position 3 – All of the noble gas released is assumed to exit the pool without mitigation. All of the non-iodine particulate nuclides are assumed to be retained by the pool water.
6. Regulatory Position 4.1 – The analysis models the release from the FSB to the environment over a 2-hour period.
7. Regulatory Position 4.2 – No credit is taken for filtration of the release from the FSB.
8. Regulatory Position 4.3 – No credit is taken for dilution of the release in the FSB.
9. Regulatory Position 5.1 – The containment personnel hatch is assumed to be open at the time of the fuel handling accident.
10. Regulatory Position 5.2 – No automatic isolation of the containment is assumed for the FHA.
11. Regulatory Position 5.3 – The release from the containment fuel pool is assumed to leak to the environment over a two-hour period.
12. Regulatory Position 5.4 – No ESF filtration of the containment release is credited.

13. Regulatory Position 5.5 – No credit is taken for dilution or mixing in the containment atmosphere.

c. Methodology

The input assumptions used in the dose consequence analysis of the FHA are provided in Table 15.7-18. The limiting accident bounds a FHA inside of containment with the containment personnel hatch open or in the Fuel Storage Building without exhaust filtration. It is assumed that the fuel handling accident occurs at 80 hours after shutdown of the reactor per Licensing Submittal 02-06. 100% of the gap activity specified in Table 3 of RG 1.183 is assumed to be instantaneously released from a single fuel assembly into the fuel pool. A minimum water level of 23 feet is maintained above the damaged fuel for the duration of the event. 100% of the noble gas released from the damaged fuel assembly is assumed to escape from the pool. All of the non-iodine particulates released from the damaged fuel assembly are assumed to be retained by the pool. The iodine released from the damaged fuel assembly is assumed to be composed of 99.85% elemental and 0.15% organic. A DF of 285 for elemental iodine and 1 for organic iodine is applied to the pool to accomplish the overall DF of 200 for the iodine release. The activity released from the pool is then assumed to leak to the environment over a two-hour period.

The FHA source term meets the requirements of Regulatory Position 1 of Appendix B to RG 1.183. The source term is listed in Table 15C-4.

For this event, the Control Room ventilation system cycle through three modes of operation:

- Initially, the ventilation system is assumed to be operating in normal mode. The air flow distribution during this mode is 1000 cfm of unfiltered fresh air makeup and an assumed value of 300 cfm of unfiltered inleakage.
- After the start of the event, the Control Room is isolated due to a high radiation reading in the Control Room ventilation system. A 30-second delay is applied to account for diesel generator start time, damper actuation time, instrument delay, and detector response time. After isolation, the air flow distribution consists of 600 cfm of filtered makeup flow from the outside, 300 cfm of unfiltered inleakage, and 390 cfm of filtered recirculation flow.
- The Control Room ventilation filter efficiencies that are applied to the filtered makeup and recirculation flows are 99% for particulate, 95% for elemental iodine, and 95% for organic iodine.

d. Radiological Consequences

The atmospheric dispersion factors (X/Qs) used for this event for the Control Room dose are based on the location of the containment personnel hatch (bounds the FSB release) and the operational modes of the control room ventilation system. These X/Qs are summarized in Tables 2R-2 and 2R-3.

The EAB and LPZ dose is determined using the X/Q factors provided in Appendix 2Q.

The radiological consequences of the FHA are analyzed using the RADTRAD-NAI code and the inputs/assumptions previously discussed. As shown in Table 15.7-19, the results for EAB dose, LPZ dose, and Control Room dose are all within the appropriate regulatory acceptance criteria.

15.7.4.4 Conclusions

The doses calculated for both the fuel handling accident occurring within the containment and for the fuel handling accident occurring in the fuel storage building are well within the values specified in regulatory limits. The fuel handling accident occurring within the containment with an open release path to the environment is the bounding fuel handling event.

15.7.5 Spent Fuel Cask Drop Accident [Historical]

Note: The spent fuel cask drop accident represents assumptions used in the original plant design. This event contains historical information that is not relevant at this time.

15.7.5.1 Identification of Causes and Accident Description [Historical]

As discussed in Subsection 9.1.4, an isolation gate installed between the spent fuel pool and the transfer canal will prevent a loss of spent fuel pool water due to a postulated cask drop accident. This gate is closed during cask handling operations. The cask handling crane cannot be passed over the isolation gate or any part of the spent fuel storage area; hence, the spent fuel shipping cask cannot be transported over these areas. Consequently, in the event that a heavy cask were dropped, the spent fuel storage area integrity would not be compromised nor any stored fuel damaged. The limited travel of the cask handling crane prevents it from traveling over any safety-related equipment.

The cask is lifted in and out of the cask loading pool in two steps. The first step is from elevation (-) 23'-10½" to a shelf at elevation 4'-5¼," a lift of 28'-3¾". The second step is from elevation 4'-5¼" to clear the operating floor at the 25' elevation, a lift of 21'-6¾". The Engineered Safety Features Filter System (Subsection 6.5.1) is in operation during handling of a loaded cask. Operation of the Fuel Storage Building Ventilation System in the emergency mode is further discussed in Subsection 6.5.1.

15.7.5.2 Radiological Consequences [Historical]

The radiological consequences for the postulated spent fuel cask drop accident have been calculated based on no impact limiting devices in the designs of the Seabrook Station cask handling equipment. The cask handling crane cannot be passed over the fuel pool isolation gate or any part of the spent fuel storage area; hence, the only source of fission products available for release are those contained within the spent fuel cask and the contained fuel assemblies. For the purpose of this analysis, it is conservatively assumed that all of the fuel pins are breached, releasing all of the halogens and noble gases contained in the gap area of the fuel pins.

The following assumptions are postulated in the calculation of the radiological consequences of the spent fuel cask drop accident; additional parameters are given in Table 15.7-26.

a. Conservative Analysis

1. The maximum number of fuel assemblies contained within one shipping cask is seven assemblies, which have been stored and decayed for a minimum of 150 days. This is a conservative estimate based on methodology used in WASH 1238 "Environmental Survey of Transportation of Radioactive Materials To and From Nuclear Power Plants," USAEC, December 1972, and current practices, of storing spent fuel assemblies. Conservative case core (193 assemblies) and gap activities for iodines and noble gases are given in Table 15.0.6.

2. The postulated cask drop occurs within the Spent Fuel Storage Building. The Engineered Safety Features Filter System is in operation, providing an iodine DF of 20.

3. It is assumed, for the purpose of providing offsite dose consequences, that all of the fission products released within the cask are instantaneously released to the Fuel Storage Building environment and ultimately to the environment via the ESF filter system. This is very conservative in view of the stringent testing criteria (10 CFR 71, Appendix B) that spent fuel shipping casks must comply with. The activity released as a result of the postulated accident is given in Table 15.7-27.

   Offsite doses resulting from the spent fuel cask drop accident were evaluated based on the dose methodology given in Appendix 15A and the one hour atmospheric dispersion parameters presented in Appendix 15B. Results are given in Table 15.7-28.

b. Realistic Analysis

The realistic analysis assumes the realistic gap fractions (Table 15.7-22) and that the Fuel Storage Building Engineered Safety Features Filter System is 99 percent efficient for the removal of elemental iodine. Realistic doses are evaluated based on the dose methodology presented in Appendix 15A and the realistic one-hour atmospheric dispersion factor (X/Q) given in Appendix 15B. Results are given in Table 15.7-28.

15.7.6 References

1. Underhill, D.W., "Effect of Rupture in a Fission Gas Holdup Bed," Nuclear Safety, 13(6), November-December 1972.
2. S. Iwai, Y. Inove and K. Nishimaki, "Movement Through Soil of Radioactive Nuclides Contained in Chemical Processing Waste," Kyoto University, 1968.
3. "Comments on the Content of AEC's Proposed Environmental Impact Statement on the Hanford Site," US EPA, Office of Radiation Programs, 1973.
4. H. B. Levy, "On Evaluating the Hazards of Groundwater Contamination by Radioactivity from an Underground Nuclear Explosion," Lawrence Livermore Laboratory, Rept. UCRL-51278, 1972.
5. Y. Inove and S. Morisawa, "On the Selection of a Ground Disposal Site by Sensitivity Analysis," Health Phys. 26, 251-261 (1973).
6. D. D. Malinowski, et al., "Radiological Consequences of a Fuel Handling Accident," WCAP-7518-L (Proprietary), July 1971 and WCAP-7828 (Nonproprietary), December 1971.

15.8 ANTICIPATED TRANSIENTS WITHOUT SCRAM

An Anticipated Transient Without Scram (ATWS) is a postulated anticipated operational occurrence (such as loss of feedwater, loss of load, or loss of off-site power) that is accompanied by a failure of the Reactor Protection System (RPS) to shutdown the reactor.

The final ATWS rule, 10 CFR 50.62 (c) (Reference 1), requires that Westinghouse designed plants install NRC approved ATWS Mitigating System Actuation Circuitry (AMSAC) to initiate a turbine trip and actuate emergency feedwater flow independent of the Reactor Protection System. The basis for this rule and the AMSAC design is supported by Westinghouse analyses documented in Reference 2 and 3. The information presented in References 2 and 3 is applicable to Seabrook. The Seabrook ATWS mitigation system is described in Subsection 7.6.12.

The basis for the final ATWS rule and the AMSAC design are supported by Westinghouse analyses documented in NS-TMA-2182 (Reference 3). These analyses were performed based on the guidelines published in NUREG-0460 (1978) (Reference 4). Appendix A of WASH-1270 (Reference 5) states that in evaluating the reactor coolant system boundary for ATWS events, "the calculated reactor coolant system transient pressure should be limited such that the maximum primary stress anywhere in the system boundary is less than that of the "emergency conditions" as defined in the ASME Nuclear Power Plant Components Code, Section III" (recently termed Service Limit C). Based on a review of reactor vessels for 2-, 3- and 4-loop plants, the maximum allowable pressure for the reactor vessel is 3200 psig. This value corresponds to the maximum allowable pressure for the weakest component in the reactor pressure vessel (the nozzle safe end); thus, the Reference 3 analyses were performed to demonstrate that the RCS pressure did not exceed 3200 psig (3215 psia). Reference 3 describes the methods used in the analyses and provides reference analyses for 2-loop, 3-loop and 4-loop plant designs with several different steam generator models available in plants at that time.

The loss of normal feedwater (LONF) and loss of load (LOL) ATWS events are the two most limiting RCS overpressure transients reported in Reference 3. To address the Power Uprate for the Seabrook Station, these two events were reanalyzed at the Power Uprate conditions to ensure that the basis for the final ATWS rule continues to be met.

The primary input to the loss of normal feedwater and loss of load ATWS analyses performed in support of the Seabrook Station Power Uprate are the 4-loop reference LONF and LOL ATWS models with Model F steam generators supporting Reference 3. The nominal and initial conditions were updated to reflect an analyzed NSSS power of 3678 MWt corresponding to the Power Uprate.

Seabrook Station has two emergency feedwater pumps, 1 motor-driven and one turbine-driven pump. The design flow capacity of each pump is 710 gpm for a total emergency feedwater flow (EFW) of 1420 gpm. The emergency feedwater flow assumed in the Reference 3 analysis was a best-estimate value of 1760 gpm. In support of the Power Uprate, three cases were run for both the LONF and LOL ATWS events, one with Reference 3 best-estimate feedwater flow of 1760 gpm and two sensitivity cases, one with the Seabrook Station emergency feedwater design flow capacity of 1420 gpm and one with an assumed maximum emergency feedwater flow capacity of 1337 gpm.

To address the Seabrook Station Power Uprate, the two most limiting RCS overpressure transients reported in Reference 3, loss of normal feedwater and loss of load, were analyzed at the Power Uprate conditions to ensure that the analytical basis for the final ATWS rule (10 CFR 60.62) (c) continues to be met. These analyses were based on the Reference 3, 4-loop loss of normal feedwater and loss of load ATWS model with Model F steam generators. The nominal and initial conditions were appropriately revised to reflect a NSSS power level of 3678 MWt consistent with the Power Uprate.

The results of the loss of normal feedwater and loss of load ATWS analyses at 3678 MWt demonstrates that the analytical basis for the final ATWS rule continues to be met for operation of the Seabrook Station with the Power Uprate.

15.8.1 References

1. ATWS Final Rule - Code of Federal Regulations 10 CFR 50.62 and Supplementary Information Package, "Reduction of Risk from Anticipated Transients Without Scram (ATWS) Events for Light-Water-Cooled Nuclear Power Plants."
2. "Westinghouse Anticipated Transients Without Trip Analysis," WCAP-8330, August 1974.
3. Anderson, T. M., "ATWS Submittal," Westinghouse Letter NS-TMA-2182 to S. H. Hanauer of the NRC, December 1979.
4. NUREG-0460, Anticipated Transients Without Scram for Light Water Reactors, USNRC, December 1978.
5. WASH-1270, Technical Report on Anticipated Transients Without Scram for Water-Cooled Power Reactors, USNRC, September 1973.

15.9 STATION BLACKOUT

10 CFR 50.63 (Reference 1), Regulatory Guide 1.155 (Reference 2) and NUMARC 87-00 (Reference 3) require that each light-water-cooled nuclear power plant be able to withstand and recover from a loss of all alternating current power or Station Blackout (loss of both offsite power and onsite emergency power). The effects of Station Blackout are not considered as part of the design basis for the transients analyzed in Chapter 15. The capability for Seabrook to cope with Station Blackout is described in Section 8.4.

15.9.1 References

1. 10 CFR 50.63, "Loss of All Alternating Current Power (Station Blackout)"
2. Regulatory Guide 1.155, "Station Blackout"
3. NUMARC 87-00, "Guidelines and Technical Bases for NUMARC Initiatives Addressing Station Blackout at Light Water Reactors"

APPENDIX 15A SUMMARY OF PARAMETERS USED FOR DOSE CALCULATIONS [HISTORICAL]

Note: UFSAR Appendix 15A represents assumptions used in the original accident analysis. The information presented in this appendix is retained for historical purposes.

A. Method of Dose Calculations

1. Conservative Case

(a) Thyroid inhalation dose

$$D_{Thy} = (\chi/Q) \bullet B \bullet \sum_{i} Qi \bullet DCFi$$

(b) Whole body gamma dose (semi-infinite cloud model)

$$D_{\partial} = C(\chi/Q) \bullet \sum_{i} Qi \bullet DFBi$$

(c) Skin dose (beta plus gamma; semi-infinite cloud model)

$$D_s = C(\chi/Q) \bullet \sum_{i} Qi \bullet DFSi + 1.11\ C(\chi/Q) \bullet \Sigma Qi \bullet DF_i^{\delta}$$

2. Realistic Case

(a) Thyroid inhalation dose

$$D_{Thy} = (\chi/Q) \bullet B \bullet \sum_{i} Qi \bullet DCFi$$

(b) Whole body gamma dose (finite cloud model)

$$D_{\partial} = C(\chi/Q)_{\partial} \bullet \sum_{i} Qi \bullet DFBi$$

(c) Skin dose (semi-infinite cloud model for beta component and finite cloud model for gamma component)

$$D_s = C(\chi/Q) \bullet \sum_{i} Qi \bullet DFSi + 1.11\ C(\chi/Q)_{\partial} \sum_{i} Qi \bullet DF_i^{\delta}$$

Where:

D_{Thy}	=	Thyroid dose received during time interval of interest (rem)
D_s	=	Skin dose (equal to the sum of the beta dose and 1.11 times the gamma dose to air, per Regulatory Guide 1.109 (rem))
C	=	Conversion factor, equal to 1/3600 hours sec^{-1}
Qi	=	Quantity of isotope i released during the time interval of interest (curies)
(χ/Q)	=	Atmospheric dispersion parameter used for the determination of the ground-level concentration (sec $meter^{-3}$)
$(\chi/Q)_\partial$	=	The atmospheric dispersion parameter used for the determination of whole body gamma doses using the finite cloud model described in Meteorology and Atomic Energy, 1968 (sec $meter^{-3}$); see Updated FSAR Subsection 2.3.5.2(5)
B	=	Breathing rate ($meter^3$ - sec^{-1})
DCFi	=	Thyroid dose conversion factor for iodine isotope i (rem - $curie^{-3}$) inhaled
DFBi	=	Whole body gamma dose conversion factor for isotope i (rem - $hour^{-1}$/curie - $meter^{-3}$)
DF_i^δ	=	Gamma dose-to-air conversion factor (rad - $hour^{-1}$/curie - $meter^{-3}$) for isotope i
DFSi	=	Beta dose-to-skin conversion factor for isotope i (rem - $hour^{-1}$/curie - $meter^{-3}$)

The breathing rates assumed are as follows:

Breathing Rates *

Time Interval (hours)	Breathing Rate (m^3/seconds)
0-8	3.47×10^{-4}
8-24	1.75×10^{-4}
24-720	2.32×10^{-4}

B. Data for dose conversion factors are presented in Table 15A-1.

C. Doses discussed in individual accident analysis.

* Regulatory Guide 1.4

APPENDIX 15B SUMMARY OF PARAMETERS USED FOR EVALUATING RADIOLOGICAL EFFECTS OF ACCIDENTS [HISTORICAL]

Note: UFSAR Appendix 15B represents assumptions used in the original accident analysis. The information presented in this appendix is retained for historical purposes.

15B.1 INTRODUCTION

Table 15-3 of "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," Revision 3, dated November 1978, lists a number of parameters which should be identified, where applicable, for each accident which is evaluated. A number of these parameters are the same for all accidents while others must be specified for each event. It is the purpose of this appendix to tabulate the values and calculational models which are common to all accidents and to point out those which are specified for each accident individually.

15B.2 DATA AND ASSUMPTIONS

The following tabulation follows the format of Table 15-3 of the Safety Analysis Report Guide.

I. Data and assumptions used to estimate radioactive source from postulated accidents.

A. Power Level

Conservative and Realistic Cases

A power level of 3654 MWt is used based on the guaranteed core thermal output of 3411 MWt plus a 5 percent allowance for possible increased capability and a 2 percent uncertainty allowance for calorimetric measurements.

For the purpose of environmental qualification of safety equipment a power of 3479 MWt was used, which is 3411 MWt plus 2%.

B. Burnup

Conservative and Realistic Cases

Operation at 3654 MWt for a three region equilibrium cycle core at end of life is assumed for purposes of calculating fission product inventories. The three regions have operated at a specific power of 40.03 MW/Mtu for 300, 600 and 900 EFPD, respectively. This is a core burnup of 24,000 MWD/MTU. A re-evaluation of source terms and doses indicates that a burnup of up to 37,616 MWD/MTU is acceptable.

C. Percent of Fuel Perforated

Discussed in individual accident analyses.

D. Release of Activity by Nuclide

Discussed in individual accident analyses.

E. Iodine Fractions

Discussed in individual accident analyses.

F. Reactor Coolant Activity Before the Accident

Conservative Case

1. For analyses that consider reactor coolant activity levels which include a pre-existing iodine spike.

   Technical Specification limits for iodine reactor coolant activity levels of 60 μCi/gm dose equivalent I-131 and a gross activity limit of $100/\overline{E}$ are assumed for the pre-existing iodine spike. For conservatism, it is assumed that the gross activity limit of $100/\overline{E}$ is comprised solely of noble gas activity. The individual radionuclide activity concentrations corresponding to these limits are presented in Table 15B-1.

2. For analyses that consider reactor coolant activity levels which include a coincident iodine spike as a direct result of the accident.

   Initial reactor coolant iodine activity levels (prior to the accident) are assumed to be at the Technical Specification limit of 1 μCi/gm dose equivalent I-131 (individual radioiodine concentrations corresponding to this limit are given in Table 15B-2). The isotopic iodine escape rate coefficients corresponding to these values are calculated and multiplied by a factor of 500 to account for the coincident iodine spike. The revised iodine escape rate coefficient is applied for the initial 4 hours following the accident. Isotopic iodine activity concentrations as a function of time are given in Table 15B-3 for both the conservative and realistic coincident iodine spiking analyses.

Realistic Case

1. Reactor coolant activity concentrations corresponding to operation with fuel defects in 0.12 percent of the fuel rods (see Updated FSAR Table 11.1-1). No pre-existing iodine spike is assumed for the realistic reactor coolant activity levels.

2. A coincident iodine spike is assumed to occur for the realistic analyses and is determined using the same approach as presented above for the conservative analysis; however, the iodine spike and increase in the iodine release rate coefficients are based on starting at 0.12 percent failed fuel iodine activity levels given in Table 11.1-1. Isotopic iodine activity concentrations as a function of time are given in Table 15B-3.

G. Secondary Coolant Activity Before the Accident

Conservative Analysis

Secondary side activity concentrations at Technical Specification limit of 0.1 μCi/gm dose equivalent I-131. Secondary side iodine isotopic activity concentrations corresponding to 0.1 μCi/gm DE I-131 are given in Updated FSAR Table 15.1-3. Noble gases are continuously removed from the secondary side by the main Steam Condenser Evacuation System; therefore, the secondary side noble gas activity concentration is considered negligible.

Realistic Analysis

Iodine activity in each steam generator is based on 0.12 percent fuel defects and 100 lbs. per day primary-to-secondary leakage distributed evenly among the four steam generators. See Updated FSAR Table 11.1-4.

II. Data and assumptions used to estimate activity released

A. Primary Containment Volume and Leak Rate

Containment Free Air Volume is 2.704×10^6 ft^3

Conservative Case Leak Rate

0.15 percent of contained volume per day for first 24 hours and 0.075 percent of contained volume per day thereafter.

Realistic Case Leak Rate

0.1 percent of contained volume per day for first 24 hours and 0.05 percent of contained volume per day thereafter.

B. Secondary Containment Volume and Leak Rate

Total Free Volume Serviced By Emergency Exhaust Filters is 8.861×10^5 ft^3

Conservative Case Leak Rate

Ventilation exhaust rate of 2,000 cfm; no mixing.

Realistic Case Leak Rate

Ventilation exhaust rate of 2,000 cfm; 50 percent mixing.

C. Valve Movement Times

Discussed in applicable accident analyses.

D. Adsorption and Filtration Efficiencies

1. Containment Enclosure Emergency Exhaust Filter Efficiencies:

Conservative Analysis

Elemental Iodine	-	95%
Organic Iodine	-	85%
Particulate Iodine	-	95% (99% for Control Room)

Realistic Analysis

Elemental Iodine	-	99%
Organic Iodine	-	95%
Particulate Iodine	-	99%

Note: No credit for filters (Filter Efficiency = 0) for the first 8 minutes following the accident. No credit for mixing within the annulus region for the conservative analysis and 50 percent mixing credit for the realistic analysis.

Containment Enclosure Emergency Exhaust Filter By-Pass Fractions:

Conservative Analysis = 0.60 La

Realistic Analysis = 0.075 La

2. Fuel Storage Building Exhaust Filter Efficiencies:

   Same as given above for Containment Enclosure Filters except Organic Iodine - 90 percent

3. Control Room Makeup Air Intake Filter Efficiencies:

   Conservative Analysis

Elemental Iodine	-	95%
Organic Iodine	-	95%
Particulate Iodine	-	99%

E. Control Room Recirculation System Filter Efficiencies:

Serve as makeup air filter (above)

F. Containment Spray Parameters (Refer to Subsections 6.2.2 and 15.6.5.4 for details)

Conservative Case

λ(elemental) = 10.0 hr^{-1}

λ(organic) = 0.0 hr^{-1}

λ(particulate) = 0.45 hr^{-1}

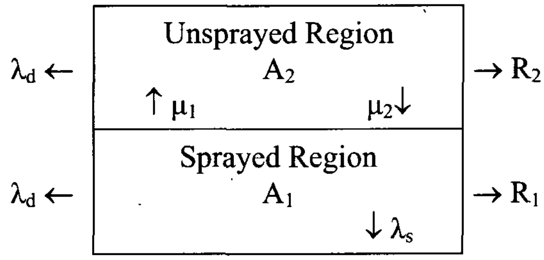
Realistic Case

λ(elemental) = 36.2 hr^{-1}

λ(organic) = 0.0 hr^{-1}

λ(particulate) = 0.84 hr^{-1}

G. Two Compartment Spray Model With Mixing



Where: A_2 = activity in unsprayed compartment

A_1 = activity in sprayed compartment

R_2 = removal rate from unsprayed compartment (leakage or venting)

R_1 = removal rate from sprayed compartment (leakage or venting)

μ_1 = transfer rate from sprayed to unsprayed compartment

μ_2 = transfer rate from unsprayed to sprayed compartment

λ_s = spray removal rate constant (hr^{-1})

λ_d = decay removal rate constant (hr^{-1})

$$\frac{dA_1}{dt} = \mu_2 A_2 - \lambda_1 A_1 \text{ where } \lambda_1 = \lambda\text{decay} + \lambda_s + \mu_1 + R_1$$

$$\frac{dA_2}{dt} = \mu_1 A_1 - \lambda_2 A_2 \text{ where } \lambda_1 = \lambda\text{decay} + \mu_2 + R_2$$

Assumed solutions:

$A_1 = Be^{-\alpha t} + Ce^{-\beta t}$

$A_2 = De^{-\alpha t} + Ee^{-\beta t}$

Where: $\alpha = ½ \left\{(\lambda_1 + \lambda_2) + \sqrt{(\lambda_1 - \lambda_2)^2 + 4\mu_1\mu_2}\right\}$

$\beta = ½ \left\{(\lambda_1 + \lambda_2) + \sqrt{(\lambda_1 - \lambda_2)^2 + 4\mu_1\mu_2}\right\}$

$$B = \frac{1}{\alpha - \beta}\left\{(\lambda_1 - \beta)A_1{}^\circ - \mu_2 A_2{}^\circ\right\}$$

$$C = \frac{1}{\alpha - \beta}\left\{-(\lambda_1 - \alpha)A_1{}^\circ + \mu_2 A_2{}^\circ\right\}$$

$$D = \frac{1}{\alpha - \beta}\left\{(\lambda_2 - \beta)A_2{}^\circ - \mu_1 A_1{}^\circ\right\}$$

$$E = \frac{1}{\alpha - \beta}\left\{-(\lambda_2 - \alpha)A_2{}^\circ + \mu_1 A_1{}^\circ\right\}$$

For limiting case where $\mu_2 \rightarrow 0$

$$\alpha = \frac{1}{2}\left\{(\lambda_1 + \lambda_2) + (\lambda_1 - \lambda_2)\right\} = \lambda_1$$

$$\beta = \lambda_2$$

$$B = \frac{1}{\lambda_1 - \lambda_2}\left\{(\lambda_1 - \lambda_2)A_1{}^\circ\right\} = A_1{}^\circ$$

$$C = 0$$

$$D = \frac{1}{\lambda_1 - \lambda_2}\left\{-\mu_1 A_1{}^\circ\right\}$$

$$B = \frac{1}{\lambda_1 - \lambda_2}\left\{-(\lambda_2 - \lambda_1)A_2{}^\circ + \mu_1 A_1{}^\circ\right\}$$

$$A_1 = A_1^{\circ} e^{-\lambda_1 t}$$

$$A_2 = \frac{-\mu_1 A_1^{\circ}}{\lambda_1 - \lambda_2} e^{-\lambda_1 t} + \left\{ A_2^{\circ} + \frac{\mu 1}{\lambda_1 - \lambda_2} A_1^{\circ} \right\} e^{-\lambda_2 t}$$

$$= \mu_1 A_1^{\circ} \left\{ \frac{e^{-\lambda_2 t} - e^{-\lambda_1 t}}{\lambda_1 - \lambda_2} \right\} + A_2^{\circ} e^{-\lambda_2 t}$$

$$= \mu_1 A_1^{\circ} E_{12} + A_2^{\circ} E_2 \quad \text{where}$$

$$E_1 = e^{-\lambda_1 t}, \; E_2 = e^{-\lambda_2 t}$$

$$E_{12} = \frac{E_1 - E_2}{\lambda_2 - \lambda_1}$$

III. Atmospheric Dispersion Factors (CHI/Q)

A. Exclusion Area Boundary and Low Population Zone

The Exclusion Area Boundary (EAB) and Low Population Zone (LPZ) accident atmospheric dispersion factors are calculated as described in Section 2.3 and presented in Table 15B-4 and Table 15B-5.

B. Control Room CHI/Q Values

The following relationships (which are approximations of Equation 6 from Murphy and Campe, Reference 1) were implemented to determine CHI/Q values for the Control Room Building, control room east air intake, and control room west air intake:

$$CHI/Q_{pc} = \frac{1}{\pi s \Sigma_y \Sigma_z}$$

$$CHI/Q_{sa} = \frac{2.032}{s u \Sigma_z}$$

where:

CHI/Q_{pc}	=	plume-centerline dispersion factor (sec/m^3)
CHI/Q_{sa}	=	sector-average dispersion factor (sec/m^3) assuming uniform horizontal dispersion across a 22.5-degree sector with the limiting condition $CHI/Q_{sa} \leq CHI/Q_{pc}$
u	=	lower level (43-foot) wind speed (m/sec)
Σ_y	=	$\left[\sigma_y^{\,2} + \dfrac{a}{\pi(k+2)}\right]^{0.5}$
Σ_z	=	$\left[\sigma_z^{\,2} + \dfrac{a}{\pi(k+2)}\right]^{0.5}$
Σ_y, σ_z	=	plume standard deviations in the horizontal and vertical directions (m) based on atmospheric stability as determined from the 150'-43' delta-temperature measurements (σ_y and σ_z values were derived by applying parabolic interpolation on a log-log basis to tabular data extrapolated from the σ_y and σ_z curves in References 2 and 3)
a	=	projected area of the Containment Building (2406 m^2)
k	=	$\dfrac{3}{(s/d)^{1.4}}$
s	=	distance between containment surface and receptor location (30 m for the Control Room Building, 81.4 m for the east air intake, and 128 m for the west air intake)
d	=	diameter of the containment (48.8 m)

Plume-centerline and sector-average dispersion factors CHI/Q_{pc} and CHI/Q_{sa} were computed for the above relationships for each sequential hour of measured onsite meteorological data for the period April 1979 through March 1980. These hourly dispersion factors and the corresponding wind direction for each hour were then stored sequentially for further processing.

Arrays of average plume-centerline dispersion factors for time windows of 1, 2, and 8 hours (for use in the 0-1 hour, 1-2 hour, and 2-8 hour accident time intervals) and arrays of average sector-average dispersion factors for time windows of 24, 96, and 720 hours (for use in the 8-24 hour, 24-96 hour, and 96-720 hour accident time intervals) were then generated for each receptor according to the expression:

$$\overline{CHI/Q_n} = \frac{1}{j_n} \sum_{j=1}^{j_m} \delta_j \, CHI/Q_{n+j-1}$$

where:

n	=	hourly sequence in the meteorological data stream at which initiation of the release was assumed to occur ($1 \leq n \leq N+1-j_m$)
N	=	total number of hours in the meteorological data base
j_m	=	length of the time window (hours)
j_n	=	number of valid hourly dispersion factors during the time window (equal to j_m if all the meteorological observations during the time window were valid)
δ_j	=	1 if the wind direction during the (n+j-1) hour affected the receptor of interest
	=	2 if the wind direction during the (n+j-1) hour did not affect the receptor of interest

The number of wind direction sectors which were assumed to result in receptor exposure for each receptor was as follows:

- Control Room Building: 7 (downwind sectors south-southwest clockwise to north-northwest)
- east air intake: 4 (downwind sectors north-northwest clockwise to northeast)
- west air intake: 3 (downwind sectors south clockwise to southwest)

For each window size, the processing began with the first hourly dispersion factor on record and then repeated for the same window size starting with each subsequent hour of dispersion data. If more than 50 percent of the hourly dispersion factors were missing in the averaging interval because of missing meteorological data, no average dispersion factor was calculated for that interval.

As an illustrative example, consider a 4-hour time window and the following sequence of hourly wind data information:

W N N W S M W M M W S S S S S

In this sequence, N represents wind blowing towards the north sector, W represents wind blowing towards the west sector, S represents wind blowing towards the south sector, and M represents missing hourly meteorological data. Assuming that: (1) winds blowing towards the north and west sectors affect the receptor-of-interest, and (2) each hourly dispersion factor is set equal to unity, the sequence of averaged dilution factors for the receptor-of-interest would be as follows:

4/4, 3/4, 2/3, 2/3, 1/2, blank, 2/2, 1/2, 1/3, 1/4, 0, 0

Frequency histograms and cumulative distributions were then generated for each resulting array of average dispersion factors. Since dispersion factors typically range over a number of decades in value, group boundaries were defined on a logarithmic scale as follows:

$$L_i = \overline{CHI/Q}_{min} * R^{i-2} \quad i \geq 2$$

$$L_{i+1} = \overline{CHI/Q}_{min} * R^{i-1}$$

where:

$$R = \left[\frac{\overline{CHI/Q}_{max}}{\overline{CHI/Q}_{min}}\right]^{1/(k-2)}$$

L_i = lower boundary of group i

L_{i+1} = upper boundary of group i (and the lower boundary of group i+1)

$\overline{CHI/Q}_{min}$ = minimum nonzero average dispersion factor

$\overline{CHI/Q}_{max}$ = maximum average dispersion factor

k = number of groups in the histogram

Group 1 was reserved for average dispersion factors with values of zero; such values occurred whenever the receptor was not downwind during any given time interval. The group index for a given nonzero dispersion factor was then determined from the largest whole number in the expression:

$$i = \frac{\log_e\left[\overline{CHI/Q}\right] - \log_e\left[\overline{CHI/Q}_{min}\right]}{\log_e\left[R\right]} + 2$$

where $\overline{CHI/Q}$ was the average dispersion factor to be classified.

A total of 541 groups were used to identify the expected probabilities that a given nonzero average dispersion factor will not be exceeded. The CHI/Q values which were exceeded five percent of the time were then chosen as the appropriate accident CHI/Q values. The resulting Control Room Building, east air intake, and west air intake CHI/Q values are presented in Table 15B-6.

References

1. K. G. Murphy, Dr. K. M. Campe, 13th AEC Air Cleaning Conference, "Nuclear Power Plant Control Room Ventilation System Design for Meeting General Design Criterion 19," August 1974.

2. NRC Regulatory Guide 1.111, "Methods For Evaluating Atmospheric Transport And Dispersion Of Gaseous Effluents In Routing Releases From Light-Water Cooled Reactors," Revision 1, July 1977.

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

APPENDIX 15C SUMMARY OF PARAMETERS USED FOR EVALUATING RADIOLOGICAL EFFECTS OF ACCIDENT USING ALTERNATIVE SOURCE TERM

15C.1 Introduction

Presented in this Appendix are various parameters employed in the performance of radiological calculations using alternative source term (AST). Regulatory Guide 1.183 provides guidance on the performance of AST analyses. The technical report for the licensing of the AST analyses is provided in Reference 15.C.7.6.

15C.2 Compliance with Regulatory Guidelines

The revised Seabrook Station accident analyses addressed in this report follow the guidance provided in RG 1.183. Assumptions and methods utilized in this analysis for which no specific guidance is provided in RG 1.183, but for which a regulatory precedent has been established, are as follows:

- Selection of Radioactive Gaseous and Liquid System Failures and Letdown Line Break dose consequences acceptance criteria for the EAB and LPZ are based on the current licensing basis of "a small fraction" of the guidelines, which is defined as 10%. 10% of the 10 CFR 50.67 limits for the EAB and LPZ equals 2.5 rem TEDE.
- Use of the MicroShield code to develop direct shine doses to the Control Room. MicroShield is a point kernel integration code used for general-purpose gamma shielding analysis. It is qualified for this application and has been used to support licensing submittals that have been accepted by the NRC.

15C.3 Computer Codes

The following computer codes are used in performing the Alternative Source Term analyses:

Computer Code	Version	Reference	Purpose
ARCON96	June 1997	15C.7.1	Atmospheric Dispersion Factors
MicroShield	5.05	15C.7.2	Direct Shine Dose Calculations
ORIGEN	2.1	15C.7.3	Core Fission Product Inventory
PAVAN	2.0	15C.7.4	Atmospheric Dispersion Factors
RADTRAD-NAI	1.1	15C.7.5	Radiological Dose Calculations

- ARCON96 – used to calculate relative concentrations (X/Q factors) in plumes from nuclear power plants at control room intakes in the vicinity of the release point using plant meteorological data.

15C.4 Radiological Evaluation Methodology

15C.4.1 Analysis Input Assumptions

Common analysis input assumptions include those for the control room ventilation system and dose calculation model (Sections 15C.4.3 and 15C.4.4), direct shine dose (Section 15C.4.6), radiation source terms (Section 15C.5), and atmospheric dispersion factors (Section 15C.6). Event-specific assumptions are discussed in the specific event analyses presented in the main body of Section 15.

15C.4.2 Acceptance Criteria

Offsite and Control Room doses must meet the guidelines of RG 1.183 and requirements of 10 CFR 50.67. The acceptance criteria for specific postulated accidents are provided in Table 6 of RG 1.183. For Seabrook Station, the events not specifically addressed in RG 1.183 are the Radioactive Gaseous and Liquid System Ruptures and Letdown Line Break.

15C.4.3 Control Room Ventilation System Description

The Normal Makeup Air Subsystem consists of two 100 percent capacity vane axial fans with a flow capacity of 1000 cfm each at the system static pressure, and the associated dampers. Air is drawn from two remote air intakes (east and west located more than 700 ft. apart). Location of the air intakes was selected considering the plant configuration and the site-specific meteorological conditions to preclude contamination of both intakes at the same time. Air flows through two 12" heavy wall carbon steel pipes provided with radiation and smoke detecting devices, as well as a normally open manual isolation valve on each path. Two 18" lines, each provided with a backdraft damper bypass the normal makeup air supply fans to supply makeup air to the filtration assemblies during the emergency mode of operation.

The Emergency Makeup Air and Filtration Subsystem consists of two filtration assemblies with a maximum capacity of 1210 cfm each (= 1100 cfm + 10% tolerance). Each assembly includes a prefilter, an electric heater, a HEPA-Carbon-HEPA filter configuration, a fan, manual inlet isolation damper, discharge isolation dampers and backdraft dampers.

In the event of an accident with a significant radiological release, high radiation is detached in either remote air supply piping, the Emergency Makeup Air and Filtration Subsystem fans are actuated and their associated dampers (1-CBA-DP-27A and DP-27B) are opened. The normal makeup air fan automatically trips off and its associated discharge damper is automatically closed. The isolation function of these dampers is safety-related.

When the normal makeup air flow path is isolated, air is drawn from the remote air intakes through the bypass lines provided with backdraft dampers. When the Emergency Makeup Air and Filtration Subsystem fans are actuated, they generate a Control Room Makeup Air Filter Recirculation Mode Signal.

Three main pathways for unfiltered inleakage to the control room were considered; inleakage via the diesel building, inleakage via the primary control room entrance (double air lock configuration), and inleakage via the emergency fire exit (two doors in series). A value of 10 cfm is typically assumed for door leakage for normal ingress/egress. However, this flow would be reduced or eliminated by a two-door vestibule. It was conservatively assumed that 20 cfm of total door leakage occurs via the most limiting door. The X/Qs for the fire exit are always more limiting than those for the primary control room entrance; therefore, all of the unfiltered inleakage via the doors was assumed to occur at the fire exit. For most release locations, the X/Qs for the fire exit are more limiting than the X/Qs for the diesel building inleakage. For these cases, the fire exit was considered as a separate path for unfiltered inleakage. In cases where the diesel building is more limiting than the fire exit, all of the unfiltered inleakage was assumed to enter via the diesel building.

15C.4.5 Control Room Inleakage Sensitivity Study

The results of the control room dose calculations were used to establish the sensitivity of the control room dose due the amount of "unfiltered inleakage" assumed to be introduced into the control room. Sensitivity studies were performed that varied allowances for unfiltered control room air inleakage. The results were then used to establish the maximum allowable unfiltered CR inleakage.

The event-specific modeling assumptions used to construct the RADTRAD-NAI files for performing the various aspects of the accident dose calculation are discussed in subsequent event analysis sections along with the input parameters used to model the Seabrook Station plant parameters. The cases presented represent the cases using the control room unfiltered inleakage rate that is determined by the sensitivity study to be limiting with respect to the CR dose acceptance criteria. The limiting unfiltered CR inleakage rates assumed in the analyses are provided in Table 15C-5.

15C.4.6 Direct Shine Dose

The total control room dose also requires the calculation of direct shine dose contributions from:

- the radioactive material on the control room filters,
- the radioactive plume in the environment, and
- the activity in the primary containment atmosphere.

The contribution to the total dose to the operators from direct radiation sources such as the control room filters, the containment atmosphere, and the released radioactive plume were calculated for the LOCA event. The LOCA shine dose contribution is assumed to be bounding for all other events. The 30-day direct shine dose to a person in the control room, considering occupancy, is provided in Table 15C-6.

The Seabrook Station reactor core consists of 193 fuel assemblies. The full core isotopic inventory is determined in accordance with RG 1.183, Regulatory Position 3.1, using the ORIGEN-2.1 isotope generation and depletion computer code (part of the SCALE-4.3 system of codes) to develop the isotopics for the specified burnup, enrichment, and burnup rates (power levels). The plant-specific isotopic source terms are developed using a bounding approach.

Sensitivity studies were performed to assess the bounding fuel enrichment and bounding burnup values. The assembly source term is based on uprated power with calorimetric uncertainty (3659 MW_{th}). For rod average burnups in excess of 54,000 MWD/MTU the heat generation rate is limited to 6.3 kw/ft in accordance with RG 1.183. For non-LOCA events with fuel failures, a bounding radial peaking factor of 1.65 is then applied to conservatively simulate the effect of power level differences across the core that might affect the localized fuel failures for assemblies containing the peak fission product inventory.

The core inventory release fractions for the gap release and early in-vessel damage phases for the design basis LOCAs utilized those release fractions provided in RG 1.183, Regulatory Position 3.2, Table 2, "PWR Core Inventory Fraction Released into Containment." For non-LOCA events, the fractions of the core inventory assumed to be in the gap are consistent with RG 1.183 Regulatory Position 3.2, Table 3, "Non-LOCA Fraction of Fission Product Inventory in Gap." In some cases, the gap fractions listed in Table 3 are modified as required by the event-specific source term requirements listed in the Appendices for RG 1.183.

The following assumptions are applied to the source term calculations:

1. A conservative maximum fuel assembly uranium loading (492 kilograms) is assumed to apply to all 193 fuel assemblies in the core.
2. Radioactive decay of fission products during refueling outages is ignored in the source term calculation.
3. When adjusting the primary coolant isotopic concentrations to achieve Technical Specification limits, the relative concentrations of fission products in the primary coolant system are assumed to remain constant.

15C.5.1 Primary Coolant Source Term

The primary coolant source term for Seabrook Station is derived from Table 15C-2-1. Table 15C-2-2 summarizes the parameters used in the calculation of the primary coolant source term.

The iodine activities from Table 15C-1 are adjusted to achieve the Technical Specification 3.4.8 limit of 1.0 μCi/gm dose equivalent I-131 using the proposed Technical Specification definition of Dose Equivalent I-131 (DE I-131) and dose conversion factors for individual isotopes from FGR 11. The non-iodine species are adjusted to achieve the Technical Specification limit of 100/E-bar for non-iodine activities.

Per Section 3.1 of Reg. Guide 1.183, the source term methodology for the Fuel Handling Accident is similar to that used for developing the LOCA containment leakage source term, except that for DBA events that do not involve the entire core, the fission product inventory of each of the damaged fuel rods is determined by dividing the total core inventory by the number of fuel rods in the core. To account for differences in power level across the core, a radial peaking factor of 1.65 is applied in determining the inventory of the damaged rods.

The LOCA containment leakage source term is based on the activity of 193 fuel assemblies and the radial peaking factor of 1.65. Thus, based on the methodology specified in Reg. Guide 1.183, the fuel handling accident source term is derived by applying a factor of 1.65/193 to the LOCA containment leakage source term. To ensure that the "bounding" assembly is identified, the activity of a peak burnup assembly (62,000 MWD/MTU), at 1.6 w/o, 3.8 w/o and 5.0 w/o, is determined and compared to the source term derived from the LOCA data. For each nuclide, the bounding activity for the allowable range of enrichments and discharge exposure is determined.

The FHA source term is presented in Table 15C-4.

15C.6 Atmospheric Dispersion (X/Q) Factors

The methodology and results for the determination of the offsite and control room atmospheric dispersion factors are discussed in Appendices 2Q and 2R.

15C.7 References

1. ARCON96 Computer Code ("Atmospheric Relative Concentrations in Building Wakes," NUREG/CR-6331, Rev. 1, May 1997, RSICC Computer Code Collection No. CCC-664 and July 1997 errata).

2. MicroShield Version 5 "User's Manual" and "Verification & Validation Report, Rev. 5," Grove Engineering, both dated October 1996.

3. Oak Ridge National Laboratory, CCC-371, "RSICC Computer Code Collection – ORIGEN 2.1," May 1999.

4. "PAVAN An Atmospheric Dispersion Program for Evaluating Design Bases Accident Releases of Radioactive Material from Nuclear Power Stations," NUREG/CR-2858, November 1982, (RSISS Computer Code Collection No. CCC-445).

5. Numerical Applications Inc., NAI-9912-04, Revision 3, "RADTRAD-NAI Version 1.1 (QA) Documentation," April 2003.

6. Numerical Applications Inc., NAI-1131-013, Revision 2, "AST Licensing Technical Report for Seabrook Station," September 8, 2003.

TABLE 15.0-1 NUCLEAR STEAM SUPPLY SYSTEM POWER RATINGS

Nuclear Steam Supply System Thermal power output	3678 MWt
Thermal power generated by the reactor coolant pumps	19 MWt
Core thermal power	3659 MWt

TABLE 15.0-3 SUMMARY OF INITIAL CONDITIONS AND COMPUTER CODES USED

	Faults	Computer Codes Utilized	Reactivity Coefficients Assumed: Moderator Temperature	Reactivity Coefficients Assumed: Doppler Power	Thermal Power Output Assumed %
15.1	Increase in Heat Removal by the Secondary System Feedwater System				
-	Feedwater System Malfunction Causing an Increase in Feed- water Flow	RETRAN	Most Negative	Least Negative	0 and 100
-	Excessive Increase in Secondary Steam Flow	N/A	N/A	N/A	100
-	Accidental Depressurization of the Main Steam System	Bounded by Steam System Piping Failure Analysis	---	---	---
-	Steam System Piping Failure	RETRAN	Most Negative	Least Negative	0

	Faults	Computer Codes Utilized	Reactivity Coefficients Assumed: Moderator Temperature	Reactivity Coefficients Assumed: Doppler Power	Thermal Power Output Assumed %
15.4	Reactivity and Power Distribution Anomalies				
-	Uncontrolled Rod Cluster Control Assembly Bank Withdrawal from a Subcritical or Low power Startup Condition	TWINKLE, FACTRAN, VIPRE	Most Positive	Least Negative	0
-	Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power	RETRAN	Most and Least Negative	Most and Least Negative	10, 60,100
-	Control Rod Misalignment	VIPRE, LOFTRAN	---	N/A	100
-	Chemical and Volume Control System Malfunction that Results in a Decrease in Boron Concentration in the Reactor Coolant	N/A	N/A	N/A	0 and 100
-	Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position	N/A	N/A	N/A	N/A
-	Spectrum of Rod Cluster Control Assembly Ejection Accidents	TWINKLE, FACTRAN	Predicted values plus uncertainty	Predicted values plus uncertainty	0 and 100

Table 15.0-4 Trip Points And Time Delays To Trip Assumed In Accident Analyses

Trip Function	Limiting Trip Point Assumed In Analysis	Time Delays (Seconds)
Power Range High Neutron Flux, High Setting	118%	0.5
Power Range High Neutron Flux, Low Setting	35%	0.5
Power Range Neutron Flux High Positive Rate	6.9%	0.65
Power Range Neutron Flux, P-8	50%	0.5
Power Range Neutron Flux, P-10	10%	N/A
Overtemperature ΔT	Variable	6.0*
Overpower ΔT	Variable	6.0*
High pressurizer pressure	2425 psia	2.0
Low pressurizer pressure	1935 psia	2.0
Low reactor coolant flow (from loop flow detectors)	87% loop flow	1.0
Undervoltage Trip	70% nominal (9660 volts)	1.5
Underfrequency Trip	55 HERTZ	0.6
Turbine Trip	Not applicable	1.0
Low-low steam generator level	0% of narrow range level span	2.0
High-high steam generator level, P-14	100% of narrow range level span	2.0

Note that P-4 is implicitly assumed in all reactor trip scenarios.

* Total time delay (including RTD time response and trip circuit channel electronics delay) from the time the temperature difference in the coolant loops exceeds the trip setpoint until the rods are free to fall.

	Incident	Reactor Trip Functions	ESF Actuation Functions	Other Equipment	ESF Equipment
--	Steam System Piping Failure	SIS, Low pressurizer pressure, Manual, OPΔT	Low pressurizer pressure, low compensated steam line pressure, Hi--1 containment pressure, Manual, high neg. steam line pressure rate	Feedwater isolation valves, Steam line stop valves, turbine stop and control valves	Emergency Feed System, Safety Injection System

	Incident	Reactor Trip Functions	ESF Actuation Functions	Other Equipment	ESF Equipment
15.3	Decrease in Reactor Coolant System Flow Rate				
--	Partial and Complete Loss of Forced Reactor Coolant Flow	Low flow, Undervoltage, Underfrequency, Manual	--	Steam generator safety valves, P-8, turbine stop and control valves	--
--	Reactor Coolant Pump Shaft Seizure (Locked Rotor)	Low flow, Manual	--	Pressurizer safety valves, steam generator safety valves, turbine stop and control valves	--

	Incident	Reactor Trip Functions	ESF Actuation Functions	Other Equipment	ESF Equipment
--	Spectrum of Rod Cluster Control Assembly Ejection Accidents	Power range high flux, High positive flux rate, Manual, low pressurizer pressure, P--10	--	Turbine stop and control valves	--
15.5	Increase in Reactor Coolant Inventory				
--	Inadvertent Operation of ECCS During Power Operation	Low pressurizer pressure, Manual, SI trip	--	Turbine stop and control valves	Safety Injection System

Table 15.0-6 [Historical] **Iodine And Noble Gas Inventory In Reactor Core And Fuel Rod Gaps**

Isotope	Core (Ci)	Percentage of Core Activity in Gap*	Assumed Fuel Rod Gap Activity (Ci)
I-127	0 (2.80 Kg)	30	0 (0.84 Kg)
I-129	2.0E+0	30	5.9E-0.1
I-130	1.8E+06	10	1.8E+05
I-131	1.0E+08	10	1.0E+07
I-132	1.4E+08	10	1.4E+07
I-133	2.1E+08	10	2.1E+07
I-134	2.3E+08	10	2.3E+07
I-135	2.0E+08	10	2.0E+07
Kr-83m	1.2E+07	10	1.2E+06
Kr-85m	2.8E+07	10	2.8E+06
Kr-85	6.8E+05	30	2.0E+05
Kr-87	5.0E+07	10	5.0E+06
Kr-88	7.2E+07	10	7.2E+06
Kr-89	8.9E+07	10	8.9E+06
Xe-131m	7.2E+05	10	7.2E+04
Xe-133m	3.0E+07	10	3.0E+06
Xe-133	2.0E+08	10	2.0E+07
Xe-135m	4.1E+07	10	4.1E+06
Xe-135	4.3E+07	10	4.3E+06
Xe-138	1.6E+08	10	1.6E+07

Note: The information presented in Table 15.0-6 represents assumptions used in the original accident analysis. The information presented in table is retained for historical purposes.

Power level 3654 MWt.

Three-region equilibrium cycle core at end of life. The three regions have operated at a specific power of 40.03 MW/Mtu for 300, 600, and 900 EFPD respectively.

* Based on Regulatory Guides 1.25 and 1.77.

COMPONENT	RESPONSE TIME	CAPACITY	TEST PROVISIONS
Emergency Feedwater[1]	2 second logic and delay, 75 second delay for pump start, and 23 second control valve closure time on high flow during an FLB	Feedline rupture -- EFW flow is based on 470 gpm minimum total flow at a steam generator back pressure of 1236 psia to two intact steam generators with two EFW pumps operational, or 470 gpm minimum total flow to three intact steam generators with one EFW pump operational Loss of feedwater with AC power: 650 gpm total minimum flow at a steam generator back pressure of 1236 psia to all steam generators with one EFW pump operational. Loss of feedwater without AC power: 650 gpm total minimum flow at a steam generator back pressure of 1236 psia to all steam generators with one EFW pump operational.	See Table 14.2-3, item 14

(1) For Steam line Rupture, see Subsection 15.1.5

Table 15.0-8 Steam Generator Tube Rupture

Component	Response Time	Capacity
Main Steam Isolation Valves	Manual operation on indicated level	--
Intact Steam Generator ASDVs		Nominal flow rate of 530,908 lb/hr/valve @ 1125 psig
Ruptured Steam Generator ASDV		Maximum flow rate of 583,385 lb/hr/valve @ 1125 psig

Table 15.1-2 MAIN STEAM LINE BREAK (MSLB) – INPUTS AND ASSUMPTIONS

Input/Assumption	Value
Core Power Level	3659 MW_{th} (includes uncertainty)
Initial RCS Equilibrium Activity (1.0 μCi/gm DE I-131 and 100/E-bar gross activity)	Table 15C-2
Initial Secondary Side Equilibrium Iodine Activity (0.1 μCi/gm DE I-131)	Table 15C-3
Maximum pre-accident spike iodine concentration	60 μCi/gm DE I-131
Maximum equilibrium iodine concentration	1.0 μCi/gm DE I-131
Duration of accident-initiated spike	8 hours
Steam Generator Tube Leakage Rate	Faulted SG – 500 gallons/day Intact SGs – 940 gallons/day
Time to establish shutdown cooling and terminate steam release	8 hours
Time for RCS to reach 212°F and terminate SG tube leakage	48 hours
RCS Mass	539,037 lb_m
SG Secondary Side Mass	Maximum (Hot Zero Power) – 166,000 lb_m (used for faulted SG to maximize release) Minimum (Hot Full Power) – 99, 304 lb_m per SG for a total of 297,912 lb_m (used for intact SGs to maximize concentration)
Release from Faulted SG	Instantaneous
Steam Release from Intact SGs	Table 15.1-3
Secondary Coolant Iodine Activity prior to accident	0.1 μCi/gm DE I-131
Steam Generator Secondary Side Partition Coefficients	Faulted SG – none Intact SGs – 100
Atmospheric Dispersion Factors Offsite Control Room	 Appendix 2Q Tables 2R-2 and 2R-3
Control Room Ventilation System Time of automatic control room normal intake isolation and switch to emergency mode	30 seconds

Table 15.1-3 Intact SGs Steam Release Rate

Time (Hours)	Intact SGs Steam Release Rate* (lb_m/min)
0-2	3383.3
2-8	2563.9
8-720.0	0

*Total release rate for all three (3) intact SGs.

Table 15.1-5 Iodine Equilibrium Appearance Assumptions

Input Assumption	**Value**
Maximum Letdown Flow	120 gpm
Assumed Letdown Flow *	132 gpm at 115°F, 2235 psia
Maximum Identified RCS Leakage	10 gpm
Maximum Unidentified RCS Leakage	1 gpm
RCS Mass	505,000 lb_m
I-131 Decay Constant	5.986968E-5 min^{-1}
I-132 Decay Constant	0.005023 min^{-1}
I-133 Decay Constant	0.000555 min^{-1}
I-134 Decay Constant	0.013178 min^{-1}
I-135 Decay Constant	0.001748 min^{-1}

* Maximum letdown flow plus 10% uncertainty

Table 15.1-7 MSLB Dose Consequences

Case	**EAB Dose[1] (rem TEDE)**	**LPZ Dose [2] (rem TEDE)**	**Control Room Dose[2] (rem TEDE)**
MSLB pre-accident iodine spike	0.08	0.15	1.00
Acceptance Criteria (pre-accident iodine spike)	25 [3]	25 [3]	5 [4]
MSLB concurrent iodine spike	0.39	0.82	3.23
Acceptance Criteria (concurrent iodine spike)	2.5 [3]	2.5 [3]	5 [4]

[1] Worst 2-hour dose

[2] Integrated 30-day dose based on an unfiltered Control Room inleakage rate of 150 cfm

[3] RG 1.183, Table 6

[4] 10 CFR 50.67

Accident	Event	Time (seconds)
	Minimum SG inventory occurs	640
	Long term peak pressurizer water level occurs	1300
c. Loss of Normal Feedwater		
	Main feedwater flow stops	20
	Low-low steam generator water level setpoint reached	59
	Rods begin to drop	61
	Four SGs begin to receive emergency feedwater flow from one emergency feedwater pump	136
	Long term peak pressurizer water level occurs	1055
	Minimum SG inventory occurs	1165
d. Feedwater System Pipe Break		
	Main feedwater line rupture occurs	0.0
	Low-low steam generator water level setpoint reached in broken loop	5.0
	Rods begin to drop	7.0
	Low Pressurizer Pressure reached for SIS injection	76.3
	Safety injection flow is started	103.3
	Emergency feedwater flow is started	105.0
	Low steamline pressure isolation setpoint is reached	114.5
	All main steam line isolation valves are closed	118.5
	Steam generator safety valves open in steam generators of intact loops	810.1
	Core decay heat plus pump heat decrease to emergency feedwater heat removal capacity	~ 4000

Table 15.2-3 Deleted

Table 15.3-1 Time Sequence Of Events For Incidents That Result In A Decrease In Reactor Coolant System Flow

Accident	Event	Time (seconds)
Partial Loss of Forced Reactor Coolant Flow	Two of four operating RCPs begin coasting down	0.0
	Low flow reactor trip setpoint reached	1.5
	Rods begin to drop	2.5
	Minimum DNBR occurs	3.0
Complete Loss of Forced Reactor Coolant Flow - Undervoltage	All operating RCPs lose power and coastdown begins	0.0
	RCP undervoltage setpoint reached	0.0
	Rods begin to drop	1.5
	Minimum DNBR occurs	2.3
Complete Loss of Forced Reactor Cooland Flow – Underfrequency	Frequency decay begins	0.0
	RCP underfrequency setpoint reached and all pumps begin to coast down	1.0
	Rods begin to drop	1.6
	Minimum DNBR occurs	3.4
Reactor Coolant Pump Shaft Seizure (Locked Rotor)	Rotor on one pump locks	0
	Low flow reactor trip setpoint reached in the affected loop	0.045
	Rods begin to fall into core; Turbine trip, loss of offsite power, unaffected reactor coolant pumps begin to coast down	1.045
	Maximum clad temperature occurs	3.50
	Maximum RCS pressure occurs	4.75

Table 15.3-3 REACTOR COOLANT PUMP SHAFT SEIZURE (LOCKED ROTOR) – INPUTS AND ASSUMPTIONS

Input/Assumption	Value
Core Power Level	3659 MW_{th} (including uncertainty)
Core Fission Product Inventory	Table 15C-1
RCS Equilibrium Activity (1.0 μCi/gm DE I-131)	Table 15C-2
Release Fraction from Breached Fuel	RG 1.183, Section 3.2, Table 3
Core Average Fuel Burnup	45,000 MWD/MTU
Fuel Enrichment	1.6 – 5.0 w/o
Maximum Radial Peaking Factor	1.65
Fuel Failure	10.0%
RCS Mass	minimum – 434,044 lb_m maximum – 539,037 lb_m Minimum mass used for fuel failure dose contribution to maximize SG tube leakage activity. Maximum mass used for RCS initial activity dose contribution.
Primary-to-Secondary Leakage Rate	1.0 gpm total (500 gpd maximum to any one SG)
Time to establish shutdown cooling and terminate release	8 hours
SG Minimum Mass (per SG)	99,304 lb_m
Secondary Side Iodine Activity prior to accident	Table 15C-3
Secondary Side Mass Releases to environment	Table 15.3-4
Steam Generator Secondary Side Partition Coefficient	100
Atmospheric Dispersion Factors Offsite Control Room	 Appendix 2Q Tables 2R-2 and 2R-3

Table 15.3-4 Locked Rotor Steam Release Rate

Time (hours)	**Intact SG Steam Release Rate (lb_m/min)**
0.0 – 2.0	3392
2.0 – 8.0	2675
8.0 – 720.0	0

Table 15.3-6 Deleted

Table 15.3-8 Deleted

Table 15.4-1 Time Sequence Of Events For Incidents That Cause Reactivity And Power Distribution Anomalies

Accident		Event	Time (seconds)
a.	Uncontrolled RCCA Bank withdrawal from a subcritical or Low-Low Power Startup Condition.	Initiation of uncontrolled rod Control withdrawal from 10^{-9} of nominal power	0.0
		Power range high neutron flux low setpoint reached	10.4
		Peak nuclear power occurs	10.6
		Rods begin to fall into core	10.9
		Minimum DNBR occurs	12.6
		Peak heat flux occurs	12.6
		Peak average clad temperature occurs	12.9
		Peak average fuel temperature occurs	13.1
b.	Uncontrolled RCCA Bank Withdrawal at Power (minimum feedback)		
	Case A	Initiation of uncontrolled RCCA withdrawal at a high reactivity insertion rate (75 pcm/sec)	0.0
		Power range high neutron flux high setpoint reached	1.9
		Rods begin to fall into core	2.4
		Minimum DNBR occurs	3.4
	Case B	Initiation of uncontrolled RCCA withdrawal at a small reactivity insertion rate (1 pcm/sec)	0
		Overtemperature ΔT reactor trip setpoint reached	78.9
		Rods begin to fall into core	80.9
		Minimum DNBR occurs	81.0

Accident			Event	Time (seconds)
7.	Dilution during power operation			
	(a)	Automatic reactor control	Dilution begins	0
			Rod insertion limit alarms occur at	T
			Operator isolates source of dilution prior to loss of all shutdown margin at time ≥	T + 900
	(b)	Manual reactor control	Dilution begins	0
			Rod insertion limit alarms, or OTΔT or other trip alarm at	T
			Operator isolates source of dilution prior to loss of all shutdown margin at time ≥	T + 900
d.	Rod Cluster Control Assembly Ejection			
1.	Beginning-of-Life, Full Power		Initiation of rod ejection	0.0
			Power range high neutron flux setpoint reached	0.04
			Peak nuclear power occurs	0.135
			Rods begin to fall into core	0.54
			Peak fuel average temperature occurs	2.10
			Peak clad temperature occurs	2.15
			Peak heat flux occurs	2.16
2.	End-of-Life, Zero Power		Initiation of rod ejection	0.0
			Power range high neutron flux low setpoint reached	0.19
			Peak nuclear power occurs	0.22
			Rods begin to fall into core	0.69
			Peak heat flux occurs	1.51
			Peak clad temperature occurs	1.51
			Peak fuel average temperature occurs	1.72

Table 15.4-3 ROD CLUSTER CONTROL ASSEMBLY (RCCA) EJECTION – INPUTS AND ASSUMPTIONS

Input/Assumption	Value
Core Power Level	3659 MW_{th} (including uncertainty)
Core Average Fuel Burnup	45,000 MWD/MTU
Fuel Enrichment	1.6 – 5.0 w/o
Maximum Radial Peaking Factor	1.65
% DNB Fuel	15%
% Fuel Centerline Melt	0.375%
LOCA Containment Leakage Source Term	Table 15C-1
Initial RCS Equilibrium Activity (1.0 μCi/gm DE I-131 and 100/E-bar gross activity)	Table 15C-2
Initial Secondary Side Equilibrium Activity (0.1 μCi/gm DE I-131)	Table 15C-3
Release from DNB Fuel	Section 1 of Appendix H to RG 1.183
Release from Fuel Centerline Melt Fuel	Section 1 of Appendix H to RG 1.183
Steam Generator Secondary Side Partition Coefficient	100
Steam Generator Tube Leakage	1.0 gpm total
Time to establish shutdown cooling and terminate steam release	8 hours
RCS Mass	minimum – 434,044 lb_m maximum – 539,037 lb_m Minimum mass used for fuel failure dose contribution to maximum SG tube leakage activity. Maximum mass used for RCS initial activity dose contribution.

Input/Assumption	Value
Secondary Containment Bypass Fraction	60%
Containment Natural Deposition Coefficients	Aerosols – 0.1 hr^{-1} Elemental Iodine – 2.2 hr^{-1} Organic Iodine - None

Table 15.4-5 RCCA Ejection Dose Consequences

Case	**EAB Dose (1) (rem TEDE)**	**LPZ Dose (2) (rem TEDE)**	**Control Room Dose (2) (rem TEDE)**
RCCA Ejection – Containment Release (3)	1.73	1.97	4.96
RCCA Ejection – Secondary Release (4)	2.57	1.99	3.22
Acceptance Criteria	6.3	6.3	5

(1) Worst 2-hour dose
(2) Integrated 30-day dose
(3) Based on an unfiltered control room inleakage rate of 190 cfm
(4) Based on an unfiltered control room inleakage rate of 150 cfm

Table 15.4-7 Deleted

Table 15.4-9 Deleted

Table 15.4-11 Deleted

Table 15.5-1 Time Sequence Of Events For Increase In Reactor Coolant Inventory Events

Accident	Event	Time (seconds)
Inadvertent Actuation of ECCS During Power Operation	Spurious SI signal generated; two charging pumps begin injecting borated water	0.0
	Terminate flow from all but one centrifugal charging pump	540.0
	Terminate all charging flow	780.0
	Peak pressurizer water volume occurs (6.0 cu. ft. below pressurizer fill)	1712.1

Table 15.5-3 Deleted

Table 15.5-5 Deleted

Table 15.6-2 LETDOWN LINE RUPTURE – INPUTS AND ASSUMPTIONS

Input/Assumption	Value
Core Power Level	3659 MW_{th} (including uncertainty)
Initial RCS Equilibrium Activity (1.0 μCi/gm DE I-131 and 100/E-bar gross activity)	Table 15C-2
Initial Secondary Side Equilibrium Iodine Activity (0.1 μCi/gm DE I-131)	Table 15C-3
Iodine spike appearance rate	500 times (see Table 15.6-4 for values)
Duration of accident initiated spike	8 hrs
Condenser Decontamination Factor	100
Steam Generator Tube Leakage	1 gpm (total for all SGs)
RCS Mass	minimum – 434,044 lb_m maximum – 539,037 lb_m Minimum mass used for iodine spike dose contribution to maximum SG tube leakage activity. Maximum mass used for RCS initial activity dose contribution.
Letdown Line Rupture flow rate	140 gpm (1160 lb/min) for 30 minutes
Letdown Line Flashing Fraction	0.1815 at 380°F and 2235 psia
Letdown Line Rupture Release Point	Worst release point (to the CR) from the PAB
Secondary Side Release Point	Worst release point (to the CR) from the Turbine Building (condenser)
Control Room Ventilation System Time of automatic control room isolation	 30 seconds
Atmospheric Dispersion Factors Offsite Control Room	 Appendix 2Q Tables 2R-2 and 2R-3
Breathing rates Offsite Control Room	 RG 1.183, Section 4.1.3 RG 1.183, Section 4.2.6
CR Occupancy Factors	RG 1.183, Section 4.2.6

Table 15.6-4 Concurrent Iodine Spike (500X) Activity Appearance Rate

Isotope	**Appearance Rate (Ci/min)**
Iodine-131	209.029344
Iodine-132	235.958907
Iodine-133	404.536383
Iodine-134	317.823719
Iodine-135	315.402448

Table 15.6-6 STEAM GENERATOR TUBE RUPTURE (SGTR) – INPUTS AND ASSUMPTIONS

Input/Assumption	Value
Core Power Level	3659 MW_{th} (include uncertainty)
Initial RCS Equilibrium Activity (1.0 µCi/gm DE I-131 and 100 E-bar gross activity)	Table 15C-2
Initial Secondary Side Equilibrium iodine Activity (0.1 µCi/gm DE I-131)	Table 15C-3
Maximum pre-accident spike iodine concentration	60 µCi/gm DE I-131
Maximum equilibrium iodine concentration	1.0 µCi/gm DE I-131
Duration of accident-initiated spike	8 hours
Steam Generator Tube Leakage Rate (the selected split between SGs maximizes dose)	Faulted SG – 313.33 gallons/day Intact SGs – 1126.67 gallons/day
Time to establish shutdown cooling and terminate steam release	8 hours
Time for RCS to reach 212°F and terminate SG tube leakage	48 hours
RCS Mass	539,037 lb_m
SG Secondary Side Mass	99,304 lb_m per SG (minimum mass used to maximize concentration)
Release Rates	Table 15.6-8
Secondary Coolant Iodine Activity prior to accident	0.1 µCi/gm DE I-131
Steam Generator Secondary Side Partition Coefficients	Faulted SG (flashed tube flow) – none Faulted SG (non-flashed tube flow) – 100 Intact SGs – 100
Break Flow Flash Fraction	Table 15.6-8
Atmospheric Dispersion Factors Offsite Control Room	 Appendix 2Q Tables 2R-2 and 2R-3
Control Room Ventilation System Time of automatic control room normal intake Isolation and switch to emergency mode	 30 seconds
Breathing Rates Offsite Control Room	 RG 1.183, Section 4.1.3 RG 1.183, Section 4.2.6
Control Room Occupancy Factor	RG 1.183, Section 4.2.6

Table 15.6-8 SGTR Release Information

Tube Break Flow - ASDV Failure Case 1

Time (hours)	Break Flow (lb_m/sec)
0.000000	12.5
0.002778	46.2
0.274167	34.9
0.500000	36.7
0.753611	42.7
1.000000	43.8
1.253611	41.4
1.461944	37.2
1.712778	37.3
1.762778	34.1
1.778333	26.2
1.793611	3.9
1.825278	4.6
1.901944	12.7
2.000000	12.6
2.777778	0

Tube Break Flow Flashing Fraction – ASDV Failure Case 1

Time (hours)	Flashing Fraction
0.000000	0.17688
0.002778	0.17864
0.274167	0.07193
0.500000	0.06080
0.753611	0.12432
1.000000	0.11501
1.253611	0.03959
1.461944	0.00229
1.712778	0.00000

Intact Steam Generator Steam Release – ASDV
Failure Case 1

Time (hours)	**Steam Release from Unaffected SGs (lb_m/min)**
0.000000	217,542
0.002778	216,967
0.274167	3,630
1.461944	9,959
1.778333	1,934
2.0	3,056
8.0	0.0
720.0	0.0

Faulted Steam Generator Steam Release – ASDV Failure Case 1

Time (hours)	**Steam Release from Faulted SG (lb_m/min)**
0.000000	72,393
0.002778	73,140
0.274167	2,743
0.500000	11,860
0.753611	7,032
1.000000	4,843
1.253611	13.9
1.461944	0
2.0	42.6
8.0	0.0
720.0	0.0

TABLE 15.6-9 60 µCi/gm DE I-131 ACTIVITIES

Isotope	Activity (µCi/gm)
Iodine-131	46.36
Iodine-132	16.88
Iodine-133	74.18
Iodine-134	10.76
Iodine-135	40.80

Table 15.6-11 Concurrent Iodine Spike (335 x) Activity Appearance Rate

Isotope	Appearance Rate (CI/min)
Iodine-131	140.04966
Iodine-132	158.092468
Iodine-133	271.039377
Iodine-134	212.941892
Iodine-135	211.31964

Table 15.6-13 Input Parameters Used In The SBLOCA Eccs Analysis

Analyzed core power (MWt)	3659
Peak linear power (kW/ft.)	13.94
Power shape	See Figure 15.6-4 (+20% axial offset)
Fuel assembly array	17x17 ZIRLO
Accumulator water volume, nominal (ft^3/ accumulator)	850
Accumulator gas pressure, minimum (psia)	600
Safety injection pump flow	See Figure 15.6-49
Nominal vessel average temperature range	571.0 – 589.1°F
Reactor coolant pressure (psia)	2250
Steam generator tube plugging level (%)	10

Table 15.6-15 Small Break Loca Results - Fuel Cladding Data

Results	3 Inch	4 Inch	6 Inch
Peak clad temperature (°F)	1114	1373	1156
Peak clad temperature location (ft)	11.25	11.25	11.0
Local Zr/H_20 reaction, maximum (%)	0.06	0.20	0.02
Local Zr/H_20 reaction location (ft)	11.25	11.25	11.0
Total Zr/H_20 reaction (%)	<1.0	<1.0	<1.0
Hot rod burst	None	None	None

Input/Assumption	Value
Spray Removal Rates: Elemental Iodine Time to reach DF of 200 Particulate Iodine Time to reach DF of 50	 20/hour 2.92 hours 5.75/hour 3.56 hours
Spray Initiation Time	65 seconds (0.018 hours)
Control Room Ventilation System Time to automatic control room normal intake isolation and switch to emergency mode	30 seconds
Containment enclosure emergency air cleaning system filter efficiency	Particulate – 95% Elemental – 95% Organic – 85%
Containment enclosure emergency air cleaning system drawdown time	8 minutes
Containment enclosure emergency air cleaning system bypass fraction	60%
Containment Purge Filtration	0%
Transport Inputs:	
Containment Leakage Release	Plant vent (filtered by CEVA) and closest containment point (CEVA bypass)
ECCS Leakage	Plant vent
RWST Backleakage	RWST tank
Containment Purge	Plant vent
Personal Dose Conversion Inputs:	
Atmospheric Dispersion Factors Offsite Control Room	 Appendix 2Q Tables 2R-2 and 2R-3
Breathing Rates Offsite Control Room	 RG 1.183, Section 4.1.3 RG 1.183, Section 4.2.6
Control Room Occupancy Factor	RG 1.183, Section 4.2.6

Table 15.6-18 Adjusted Release Rate from RWST

Time (hours)	Release Rate (cfm)
0	1.0403E-05
22	2.7185E-05
24	6.4831E-05
100	1.0186E-04
200	1.3059E-04
300	1.5290E-04
400	1.7027E-04
500	1.8411E-04
600	1.8538E-04
700	1.8044E-04

Table 15.6-20 LOCA Dose Consequences

Dose Component	**EAB Dose [1] (rem TEDE)**	**LPZ Dose [2] (rem TEDE)**	**Control Room Dose [2] (rem TEDE)**
Containment Purge	4.2391E-04	2.0602E-04	3.8398E-04
Containment Leakage	4.6199	3.2319	3.8024
ECCS Leakage	9.5305E-03	4.7662E-02	2.7304E-02
RWST Backleakage	1.0728E-02	0.14140	0.47062
Radiation Shine			0.45
Total	4.64	3.42	4.75
Acceptance Criteria	25	25	5

(1) Worst 2-hour dose
(2) Integrated 30-day dose based on an unfiltered Control Room inleakage rate of 150 cfm

Table 15.6-22 DELETED

Table 15.6-24 DELETED

Table 15.6-26 Seabrook Station large Break LOCA Containment Data Used for Calculation of Containment Pressure

Net Free Volume	2.974×10^6 ft^3
Initial conditions	
Pressure	14.6 psia
Temperature	90°F
RWST temperature (Spilling SI and Spray)	50°F
Temperature outside containment	50°F
Spray System	
Post-accident spray system initiation delay	
with LOOP	65 sec
without LOOP	39 sec
Maximum spray system delivered flow (both pumps operating)	7000 gal/min
Containment Fan Coolers	N/A

MATERIAL PROPERTIES

Material	Thermal Conductivity (BTU/hr-°F-ft)	Volumetric Heat Capacity (BTU/ft.^{3}-°F)
Concrete	0.92	22.62
Carbon Steel	27.0	58.80
Stainless Steel	10.0	58.80
Paint	10.54	24.12

Time (sec)	M&E from Loop Side BCL Mass Flow (lbm/s)	M&E from Loop Side BCL Energy Flow (Btu/s)	M&E from Vessel Side BCL Mass Flow (lbm/s)	M&E from Vessel Side BCL Energy Flow (Btu/s)
170	65	81052	574	196391
180	64	80468	584	192434
190	66	82776	240	154844
200	68	84175	232	162589

Parameter	Reference Split Break Transient	Uncertainty or Bias
2.0 Plant Initial Operating Conditions		
2.1 Reactor Power		
a. Core average linear heat rate (AFLUX)	Nominal – Based on 100% of power (3659 MWt)	ΔPCT_{PD}[2]
b. Hot Rod Peak linear heat rate (PLHR)	FQ = 2.2; Derived from desired Tech Spec (TS) limit FQ = 2.5 and maximum baseload FQ = 2.0	ΔPCT_{PD}[2]
c. Hot rod average linear heat rate (HRFLUX)	$F_{\Delta H}$ = 1.683; Derived from TS $F_{\Delta H}$ = 1.65	ΔPCT_{PD}[2]
d. Hot assembly average heat rate (HAFLUX)	HRFLUX/1.04	ΔPCT_{PD}[2]
e. Hot assembly peak heat rate (HAPHR)	PLHR/1.04	ΔPCT_{PD}[2]
f. Axial power distribution (PBOT, PMID)	Figure 15.6-5	ΔPCT_{PD}[2]
g. Low power region relative power (PLOW)	0.2	Bounded*
h. Hot assembly burnup	BOL	Bounded
i. Prior operating history	Equilibrium decay heat	Bounded
j. Moderator Temperature Coefficient (MTC)	Tech Spec Maximum (0)	Bounded
k. HFP boron	800 ppm	Generic

	Parameter	Reference Split Break Transient	Uncertainty or Bias
3.0	Accident Boundary Conditions		
	a. Break location	Cold leg	Bounded
	b. Break type	Split	ΔPCT_{MOD}[1]
	c. Break Size	2.0 times cold leg area	ΔPCT_{MOD}[1]
	d. Offsite power	Loss-Of-Offsite-Power (LOOP)	Bounded*
	e. Safety injection flow	Minimum	Bounded
	f. Safety injection temperature	Nominal (75.0°F)	ΔPCT_{IC}
	g. Safety injection delay	Max delay with LOOP (30.0 sec)	Bounded
	h. Containment pressure	Bounded – Slightly lower than pressure curve shown in Figure 15.6-21	Bounded
	i. Single failure	ECCS: Loss of 1 SI train Containment press: all trains operations	Bounded
	j. Control rod drop time	No control rods	Bounded

Table 15.6-30 Seabrook station Large Break LOCA Confirmatory Cases PCT results Summary

Case	PCT (°F)		
	Blowdown	1st Reflood	2nd Reflood
Initial Transient	1337	1489	1462
Reduced SGTP (0%)	1322	1424	1376
Offsite Power Available	1296	1327	1364
Low Nominal RCS T_{avg} (571.0°F)	1310	1349	1323
Increased PLOW (0.6)	1337	1462	1450

Table 15.6-32 Seabrook Station Large Break LOCA Results

Component	First Reflood	Second Reflood	Criteria
50^{th} Percentile PCT (°F)	< 1520	< 1412	N/A
95^{th} Percentile PCT (°F)	< 1789	< 1724	< 2200
Maximum Local Oxidation (%)	< 3.53		< 17.0
Maximum Total Hydrogen Generation (%)	< 0.30		< 1.0

Parameter	**Operating Range**
e) Hot assembly peak linear heat rate	$F_{QHA} \leq 2.5/1.04$
f) Axial power dist (PBOT, PMID)	Figure 15.6-20
g) 44 assembly peripheral region relative power (PLOW)	$0.2 \leq PLOW \leq 0.6$
h) Hot assembly burnup	≤ 75000 MWD/MTU, lead rod
i) Prior operating history	All normal operating histories
j) MTC	≤ 0 at HFP
k) HFP boron (minimum)	800 ppm (at BOL)
l) Rod power census	See Table 15.6-34
2.2 Fluid Conditions	
a) T_{avg}	$571.0 - 2.9 \leq T_{avg} \leq 589.1 + 5.7$°F (4)
b) Pressurizer pressure	$P_{RCS} = 2250 \pm 50$ psia
c) Loop flow	≥ 93600 gpm/loop
d) T_{UH}	Current upper internals, T_{cold} UH
e) Pressurizer level	Normal level, automatic control
f) Accumulator temperature	$70 \leq T_{ACC} \leq 100$°F

(4) 571°F and 589.1°F are nominal values. The +/- values reflect bias and uncertainty.

Table 15.6-34 **SEABROOK STATION ROD CENSUS USED IN BEST-ESTIMATE LARGE BREAK LOCA ANALYSIS**

Rod Group	Power Ratio (Relative to HA Rod Power)	% of Core
1	1.0	10
2	0.912	10
3	0.853	10
4	0.794	10
5	0.735	10
6	0.676	10
7	<0.65	40

Table 15.7-1 Sequence Of Events For Rgws Failure

Approximate Elapsed Time	Events
0 second	Event begins - one carbon delay bed ruptures
0 second	Noble gases are released (Ref. Table 12.2-27)
<1 minute	Radiation and hydrogen alarms alert plant personnel
>1 minute	Operator actions begin with: 1. Purge RGWS with nitrogen 2. Isolate system 3. Evacuate unnecessary personnel from the area

Table 15.7-3 RGWS Source Term (Ci)

Isotope	1st Carbon Delay Bed	2nd Carbon Delay Bed	3rd Carbon Delay Bed	4th Carbon Delay Bed	5th Carbon Delay Bed	Total RGWS Inventory (Curies)
Kr-83m	8.2E+01	1.5E-01	-	-	-	8.2E+01
Kr-85m	7.5E+02	5.2E+01	3.5E+00	2.4E-01	1.7E-02	8.1E+02
Kr-85	1.8E+02	1.8E+02	1.8E+02	1.8E+02	1.8E+02	9.0E+02
Kr-87	1.5E+02	1.4E-02	<1.0E-02	-	-	1.5E+02
Kr-88	1.0E+03	1.5E+01	2.2E-01	<1.0E-02	-	1.0E+03
Xe131m	1.1E+03	5.7E+02	2.8E+02	1.4E+02	7.0E+01	2.2E+03
Xe-133m	3.5E+03	8.8E+01	2.2E+00	5.6E-02	<1.0E-02	3.6E+03
Xe-133	3.0E+05	6.1E+04	1.3E+04	2.6E+03	5.4E+02	3.8E+05
Xe-135m	1.2E+01	-	-	-	-	1.2E+01
Xe-135	3.2E+03	-	-	-	-	3.2E+03
Xe-137**	2.1E-01	-	-	-	-	2.1E-01
Xe-138	8.3E+00	-	-	-	-	8.3E+00

** Not included in analysis (insignificant)

Table 15.7-5 DELETED

Table 15.7-7 Radioactive Liquid Waste System Failure – Inputs and Assumptions

Input/Assumption	**Value**
RLWS release inventory	Table 15.7-8
RLWS component volume (arbitrary)	10,000 ft^3
RLWS leak rate (arbitrarily high)	Entire inventory released within 2 hours
Control Room Ventilation System Time of automatic control room isolation	30 seconds
Atmospheric Dispersion Factors Offsite Control Room	 Appendix 2Q Tables 2R-2 and 2R-3
Breathing Rates Offsite Control Room	 RG 1.183, Section 4.1.3 RG 1.183, Section 4.2.6
CR Occupancy Factors	RG 1.183, Section 4.2.6

Table 15.7-9 RLWS Failure Dose Consequences

Case	EAB Dose (1) (rem TEDE)	LPZ Dose (2) (rem TEDE)	Control Room Dose (2) (rem TEDE)
Boron Waste Tank	0.04	0.02	0.92
Letdown Degasifier	0.04	0.02	0.46
Acceptance Criteria	0.1	0.1	5

(1) Worst 2-hour dose
(2) Integrated 30-day dose based on an unfiltered control room inleakage of 300 cfm

Table 15.7-11 DELETED

Table 15.7-13 DELETED

Table 15.7-15 DELETED

Table 15.7-17 Radionuclide Concentration

Isotope	Concentration in Marsh (μCi/ml)	Concentration in Hampton Harbor (μCi/ml)
Sr-90	1.7E-08*	6.9E-14
Y-91	1.1E-03	4.6E-09
Zr-95	2.0E-04	8.2E-10
Nb-95	1.7E-05	6.9E-11
Cs-137	6.9E-04	2.8E-09
Ce-144	1.8E-03	7.4E-09
Mn-54	3.4E-03	1.4E-08
Co-60	6.2E-03	2.5E-08
Fe-59	6.6E-05	2.7E-10
Cr-51	3.0E-06	1.2E-11
H-3	8.0E-01	3.3E-06

* 1.7E-08 = 1.7×10^{-8}

Table 15.7-19 Fuel Handling Accident Dose Consequences

Case	EAB Dose (1) (rem TEDE)	LPZ Dose (2) (rem TEDE)	Control Room Dose (2) (rem TEDE)
FHA	1.41	0.69	2.39
Acceptance Criteria	6.3	6.3	5

(1) Worst 2-hour dose
(2) Integrated 30-day dose based on an unfiltered Control Room inleakage rate of 300 cfm

Radionuclide	Total Activity At Shutdown (Ci)	Fraction In Cladding Gap	Activity in Cladding Gap (Ci) At Reactor Shutdown	Activity in Cladding Gap (Ci) 80 Hours After Shutdown
Xe-133	1.7E+06	0.05	8.5E+04	6.5E+04
Xe-135m	3.6E+05	0.05	1.8E+04	3.0E+00
Xe-135	4.3E+05	0.05	2.2E+04	4.8E+02
Xe-138	1.4E+06	0.05	7.0E+04	Negl.

Table 15.7-22 DELETED

Table 15.7-24 Deleted

Table 15.7-25 Deleted

Table 15.7-26 Summary Of Parameters And Assumptions Used For The Spent Fuel Cask Drop Accident [Historical]

			Conservative Analysis	Realistic Analysis
I.	Data and assumptions used to estimate radioactive source from postulated accident			
	A.	Power level	Appendix 15B	Appendix 15B
	B.	Burnup	Appendix 15B	Appendix 15B
	C.	Percent of fuel perforated	100 (7 assemblies)	100 (7 assemblies)
	D.	Release of activity by nuclide	Table 15.7-27	Table 15.7-27
	E.	Iodine fractions elemental, organic and particulate)	All elemental	All elemental
	F.	Reactor coolant and secondary coolant activity before the accident	NA	NA
II.	Data and assumptions used to estimate activity released			
	A.	Primary containment leak rate	NA	NA
	B.	Secondary containment leak rate	NA	NA
	C.	Valve movement times	NA	NA
	D.	Absorption and filtration efficiency (%)		
	E.	Recirculation system parameters (flow rates versus time, missing factor, etc.)	95	99
	F.	Containment spray parameters	NA	NA
	G.	Containment volumes	NA	NA
	H.	All other pertinent data and assumptions	Subsection 15.7.5.2	Subsection 15.7.5.2

Note: The spent fuel cask drop accident represents assumptions used in the original plant design. This event contains historical information that is not relevent at this time.

Table 15.7-27 Activity Released To Environment From Spent Fuel Cask Drop Accident[*] [Historical]

Radionuclide	Activity Released (curies) Conservative	Realistic
I-131	8.7E-02[**]	1.6E-03
Kr-85	7.2E+03	6.6E+03
Xe-131M	4.2E-01	9.5E-02
Xe-133	1.8E-03	2.8E-04

Note: The spent fuel cask drop accident represents assumptions used in the original plant design. This event contains historical information that is not relevant at this time.

[*] Based on gap activity of seven fuel assemblies and 150 days decay.

[**] 8.7E-02 = 8.7×10^{-2} = 0.087

Thyroid Dose Conversion Factors
(Rem/Ci Inhaled)

Iodine Isotope	R. G. 1.109 DCF Rem/Ci	ICRP-30 DCF Rem/Ci
I-131	1.490E+06	1.080E+06
I-132	1.430E+04	6.438E+03
I-133	2.690E+05	1.798E+05
I-134	3.730E+03	1.066E+03
I-135	5.600E+04	3.130E+04

The Dose Conversion Factors (DCFs) used to determine the radiological consequences of UFSAR Chapter 15 Design Bases Accidents (DBAs) have evolved considerably since the plant's inception in the 1970's when TID14844 values were considered. In general, the thyroid and skin DCFs used for the Seabrook Station DBAs are based on Regulatory Guide 1.109, and the whole body DCFs are from an internally generated document "Gamma Dose Correction Factors for Finite Hemispherical Clouds," model by J. Hamawi, May 31, 1977.

Recent reanalysis of the radiological consequences of certain DBAs (for instance, *the Steam Generator Tube Rupture Event, Containment Fuel Handling Accident and the LOCA Control Room Habitability)* have used the more recent ICRP-30 DCFs. The ICRP-30 dose conversion values (as taken from Federal Guidance Report 11) are used for thyroid, skin, and whole body dose evaluation.

The table above illustrates the differences between the more conservative Regulatory Guide 1.109 DCFs and the more recent ICRP-30 thyroid DCFs. The ICRP-30 values are used industry wide and by the Regulatory staff as referenced in NRC Regulatory Issue Summary 2001-19; "Deficiencies in the Documentation of Design Basis Radiological Analyses Submitted in Conjunction with License Amendment Requests," October 18, 2001.

Note: This table represents information used in the original accident analysis. The information presented in this table is retained for historical purposes.

Reactor Coolant Isotopic Iodine Activity Concentrations For 1 μCi/Gm Dose Equivalent I-131

Radionuclide	Reactor Coolant Concentration (μCi/gm) Equivalent to 1 Dose Equivalent I-131
I-131	7.6E-01*
I-132	2.7E-01
I-133	1.2E+00
I-134	1.7E-01
I-135	6.6E-01

Note: This table represents information used in the original accident analysis. The information presented in this Table is retained for historical purposes.

* 7.6E-01 = $7.6x10^1$

Summary Of Dilution Factors At The Exclusion Radius (Sec/M^3) 914 Meters, Apr 79 - Mar 80 Onsite Meteorology

		Time Interval	Maximum (ESE) Sector Values[(a)]	Overall-Site Values[(b)]
I. Concentration CHI/Q Values				
A.	Conservative Estimates	0-1 Hour	2.67×10^{-4}	2.32×10^{-4}
		1-2 Hours	1.88×10^{-4}	1.72×10^{-4}
		2-8 Hours	1.02×10^{-4}	9.35×10^{-5}
		8-24 Hours	2.58×10^{-5}	2.64×10^{-5}
		1-4 Days	1.43×10^{-5}	1.49×10^{-5}
		4-30 Days	7.78×10^{-6}	7.57×10^{-6}
B.	Realistic Estimates	0-1 Hour	3.53×10^{-5}	3.78×10^{-5}
		1-2 Hours	2.66×10^{-5}	2.83×10^{-5}
		2-8 Hours	1.44×10^{-5}	2.26×10^{-5}
		8-24 Hours	5.97×10^{-6}	1.06×10^{-5}
		1-4 Days	5.21×10^{-6}	7.45×10^{-6}
		4-30 Days	5.74×10^{-6}	5.81×10^{-6}
II. Effective Gamma CHI/Q Values				
A.	Conservative Estimates	0-1 Hour	2.98×10^{-5}	3.00×10^{-5}
		1-2 Hours	2.05×10^{-5}	2.13×10^{-5}
		2-8 Hours	1.14×10^{-5}	1.12×10^{-5}
		8-24 Hours	6.02×10^{-6}	6.21×10^{-6}
		1-4 Days	3.71×10^{-6}	3.74×10^{-5}
		4-30 Days	2.37×10^{-6}	2.31×10^{-6}
B.	Realistic Estimates	0-1 Hour	6.19×10^{-6}	7.23×10^{-6}
		1-2 Hours	4.75×10^{-6}	5.73×10^{-6}
		2-8 Hours	2.66×10^{-6}	4.30×10^{-6}
		8-24 Hours	1.91×10^{-6}	3.39×10^{-6}
		1-4 Days	1.48×10^{-6}	2.21×10^{-6}
		4-30 Days	1.61×10^{-6}	1.63×10^{-6}

Note: This table represents information used in the original accident analysis. The information presented in this table is retained for historical purposes.

(a) The maximum sector conservative CHI/Q values represent the ESE sector's values which are exceeded 0.5 percent of the total time; the maximum sector realistic CHI/Q values represent the ESE sector's median values.

(b) The overall-site conservative CHI/Q values represent the overall-site values which are exceeded 5 percent of the total time; the overall-site realistic CHI/Q values represent the overall-site median values.

Control Room Dilution Factors

Time Interval	Accident Control Room CHI/Q Values (sec/m³) Concentration[(a)]	Gamma
1 hr	4.08×10^{-3}	2.49×10^{-4}
2 hrs	3.18×10^{-3}	1.92×10^{-4}
8 hrs	2.04×10^{-3}	1.24×10^{-4}
24 hrs	1.44×10^{-3}	8.85×10^{-5}
96 hrs	9.78×10^{-4}	6.02×10^{-5}
720 hrs	7.51×10^{-4}	4.63×10^{-5}

Time Interval	5% Concentration CHI/Q Values (sec/m³) East CR Intake[(b)]	West CR Intake
1 hr	1.42×10^{-3}	1.57×10^{-3}
2 hrs	1.14×10^{-3}	9.81×10^{-4}
8 hrs	6.95×10^{-4}	4.59×10^{-4}
24 hrs	4.67×10^{-4}	2.53×10^{-4}
96 hrs	3.05×10^{-4}	1.49×10^{-4}
720 hrs	2.00×10^{-4}	7.77×10^{-5}

Note: This table represents information used in the original accident analysis. The information presented in this table is retained for historical purposes.

(a) These values are appropriate for infiltration or leakage of air into the control room from either containment leakage or a primary vent stack

(b) These values are appropriate for a containment leakage pathway only.

NUCLIDE	CURIES	NUCLIDE	CURIES
I-131	1.051E+08	EU-154	2.003E+06
I-132	1.491E+08	EU-155	1.387E+06
I-133	1.988E+08	EU-156	4.767E+07
I-134	2.152E+08	LA-143	1.466E+08
I-135	1.872E+08	NB-97	1.628E+08
XE-133	1.994E+08	NB-95M	1.177E+06
XE-135	5.012E+07	PM-147	1.379E+07
CS-134	3.258E+07	PM-148	2.968E+07
CS-136	8.347E+06	PM-149	6.826E+07
CS-137	1.365E+07	PM-151	2.409E+07
BA-139	1.747E+08	PM-148M	3.426E+06
BA-140	1.684E+08	PR-144	1.350E+08
LA-140	1.750E+08	PR-144M	1.609E+06
LA-141	1.593E+08	SM-153	7.818E+07
LA-142	1.538E+08	Y-94	1.448E+08
CE-141	1.623E+08	Y-95	1.560E+08
CE-143	1.476E+08	Y-91M	6.581E+07
CE-144	1.340E+08	BR-82	8.712E+05
PR-143	1.464E+08	BR-83	1.183E+07
ND-147	6.392E+07	BR-84	2.044E+07
NP-239	2.922E+09	AM-242	1.285E+07
PU-238	5.151E+05	NP-238	7.062E+07
		PU-243	1.266E+08

Table 15C-2-1 Reactor Coolant Fission and Corrosion Product Activities – 1% Clad Defects

NUCLIDE	MCI/GM	NUCLIDE	MCI/GM
I-131	2.5	SR-91	0.031
I-132	0.91	Y-90	0.00022
I-133	4.0	Y-91	0.0058
I-134	0.58	Y-92	0.001
I-135	2.2	ZR-95	0.00067
H-3	5.0	NB-95	0.00068
KR-83M	0.43	MO-99	3.3
KR-85M	1.7	CS-134	0.44
KR-85	0.13	CS-136	0.22
KR-87	1.3	CS-137	2.2
KR-88	3.4	BA-140	0.0045
XE-131M	0.067	LA-140	0.0014
XE-133M	0.57	CE-144	0.00044
XE-133	25	MN-54	0.00031
XE-135M	0.82	CO-58	0.016
XE-135	3.1	CO-60	0.002
XE-138	0.71	FE-59	0.001
SR-89	.0041	CR-51	0.0019
SR-90	.00018	FE-55	0.0016

Table 15C-3 Secondary Side Source Activities *

Isotope	µCi/gm
I-131	0.07727
I-132	0.02813
I-133	0.12363
I-134	0.01793
I-135	0.06800

* 0.1 µCi/gm Dose Equivalent I-131

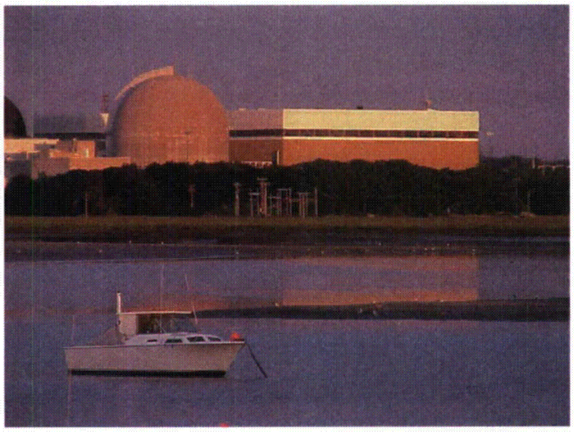
Table 15C-5 Control Room Ventilation System Parameters

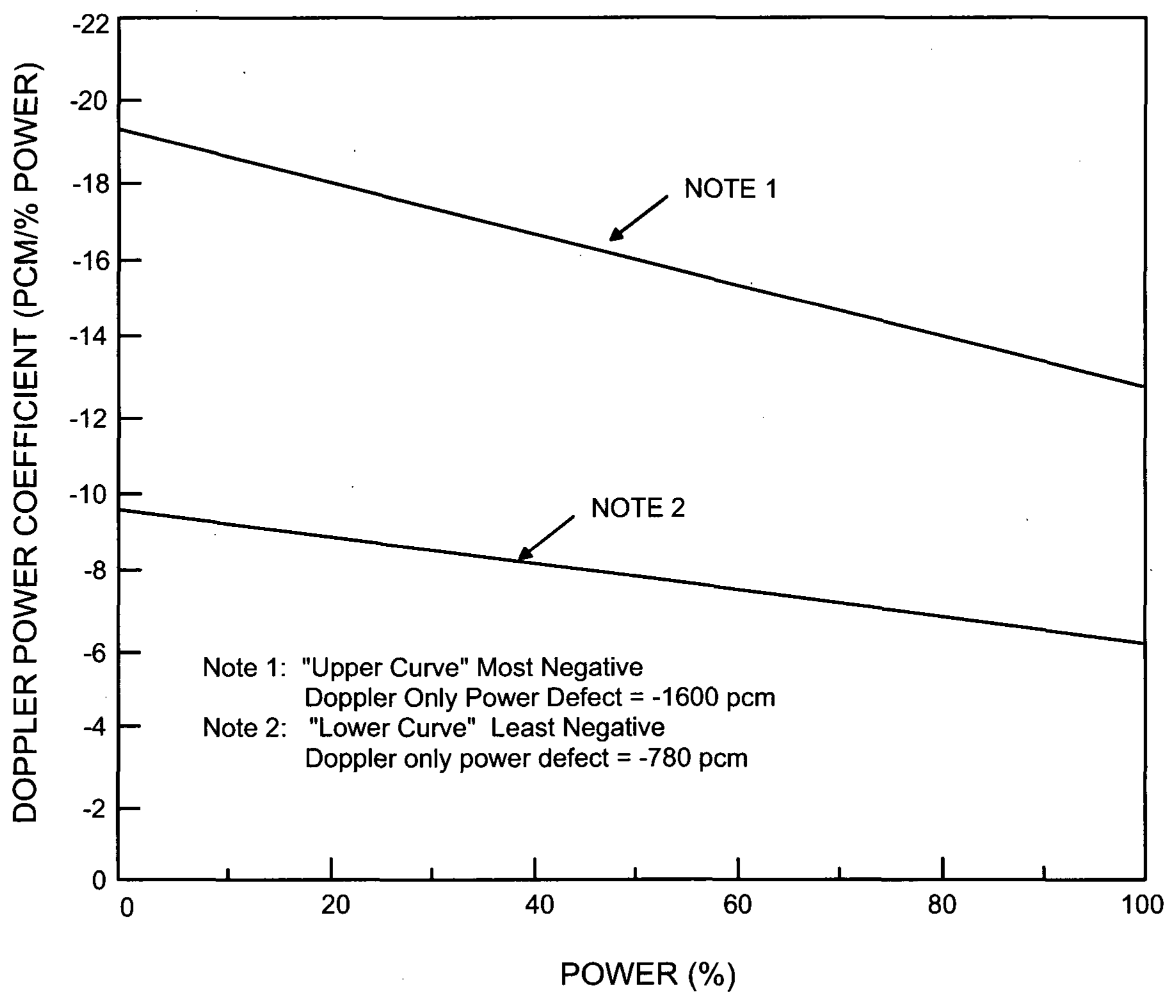
Parameter	**Value**
Control Room Volume	246,000 ft^3
Normal Operation	
Filtered Make-up Flow Rate	0 cfm
Filtered Recirculation Flow Rate	0 cfm
Unfiltered Make-up Flow Rate	1000 cfm
Unfiltered Inleakage (Total)	
LOCA, MSLB, Locked Rotor, RCCA Ejection – Secondary Release	150 cfm
SGTR, Small Line Break Outside Containment (Letdown Line), Radioactive Gaseous Waste System Failure, Radioactive Liquid Waste System Failure and Fuel Handling Accident	300 cfm
RCCA Ejection – containment release	190 cfm
Emergency Operation	
Filtered Make-up Flow Rate	600 cfm
Filtered Recirculation Flow Rate	390 cfm (entire 10% tolerance on total CR filter flow conservatively applied to reduce recirculation flow)
Unfiltered Make-up Flow Rate	0 cfm
Unfiltered Inleakage (Total)	
LOCA, MSLB, Locked Rotor, RCCA Ejection – Secondary Release	150 cfm
SGTR, Small Line Break Outside Containment (Letdown Line), Radioactive Gaseous Waste System Failure, Radioactive Liquid Waste System Failure and Fuel Handling Accident	300 cfm
RCCA Ejection – containment release	190 cfm
Filter Efficiencies	
Elemental	95%
Organic	95%
Particulate	99%

SEABROOK STATION
UPDATED FINAL SAFETY ANALYSIS REPORT

CHAPTER 15
ACCIDENT ANALYSES

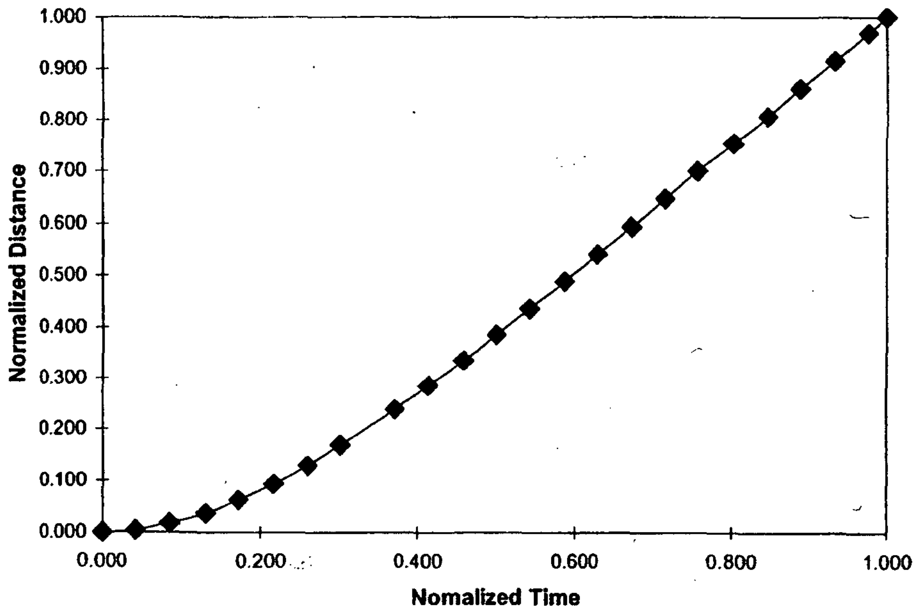
FIGURES



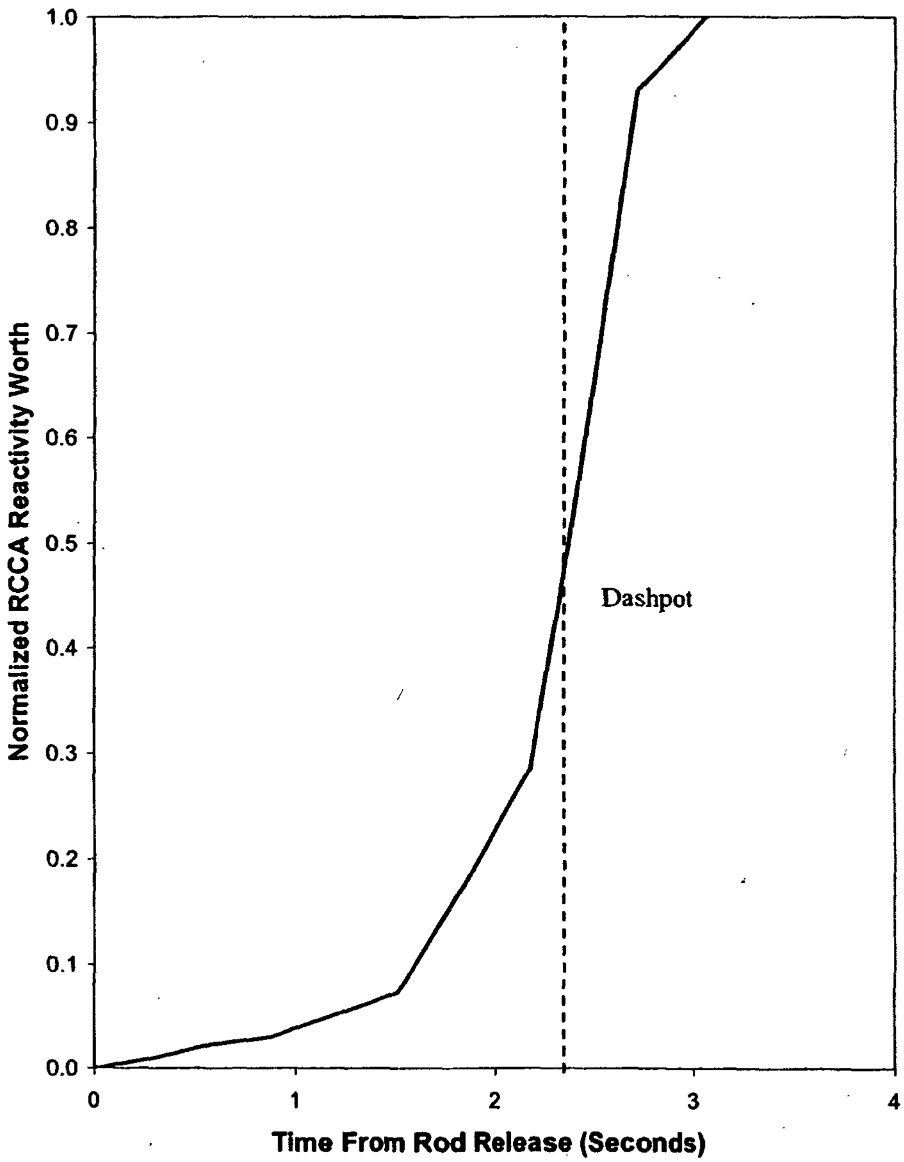


G:\Word\Images_P\Images_P\15002.DS4

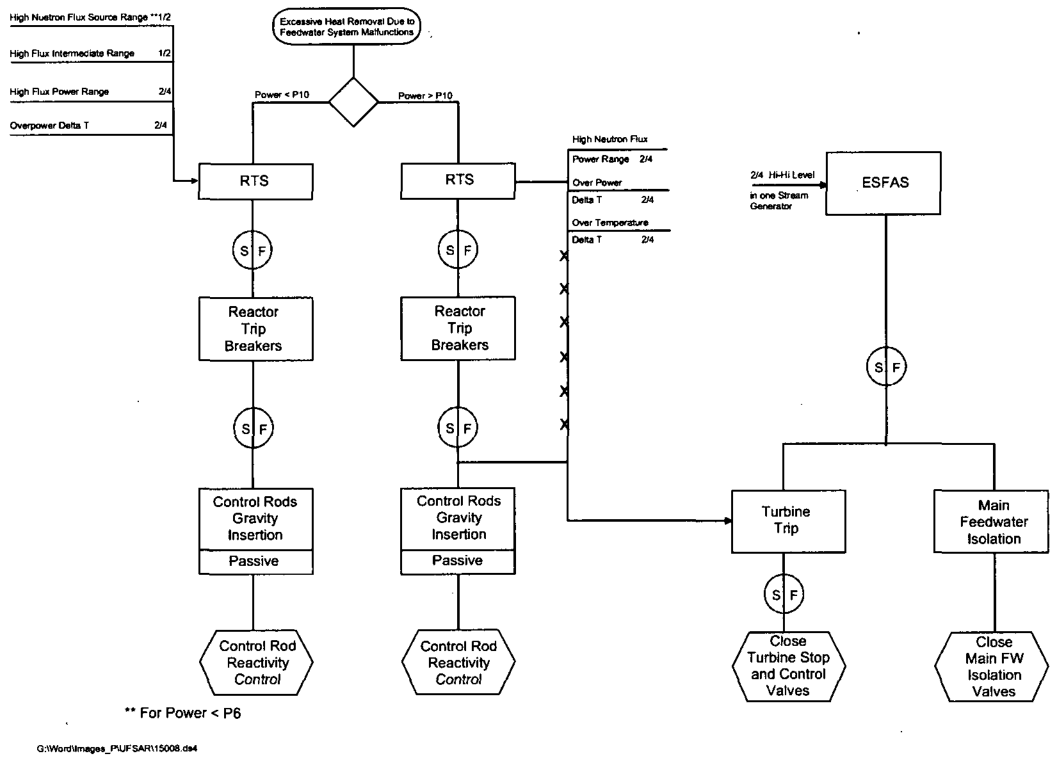
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Doppler Power Coefficient Assumed in Analyses	
		Figure 15.0-02



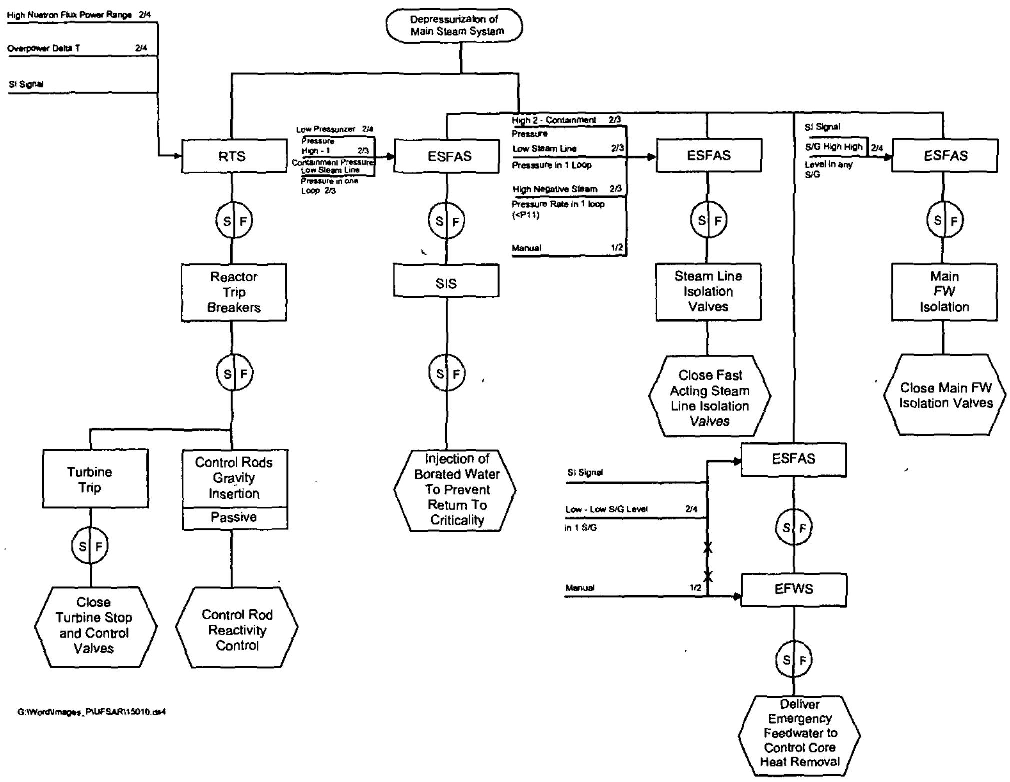
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Normalized RCCA Position (Fraction insertion) vs. Normalized RCCA Drop Time After Release	
		Figure 15.0-04



SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Normalized RCCA Reactivity Worth versus Time from Start of Rod Motion	
		Figure 15.0-06

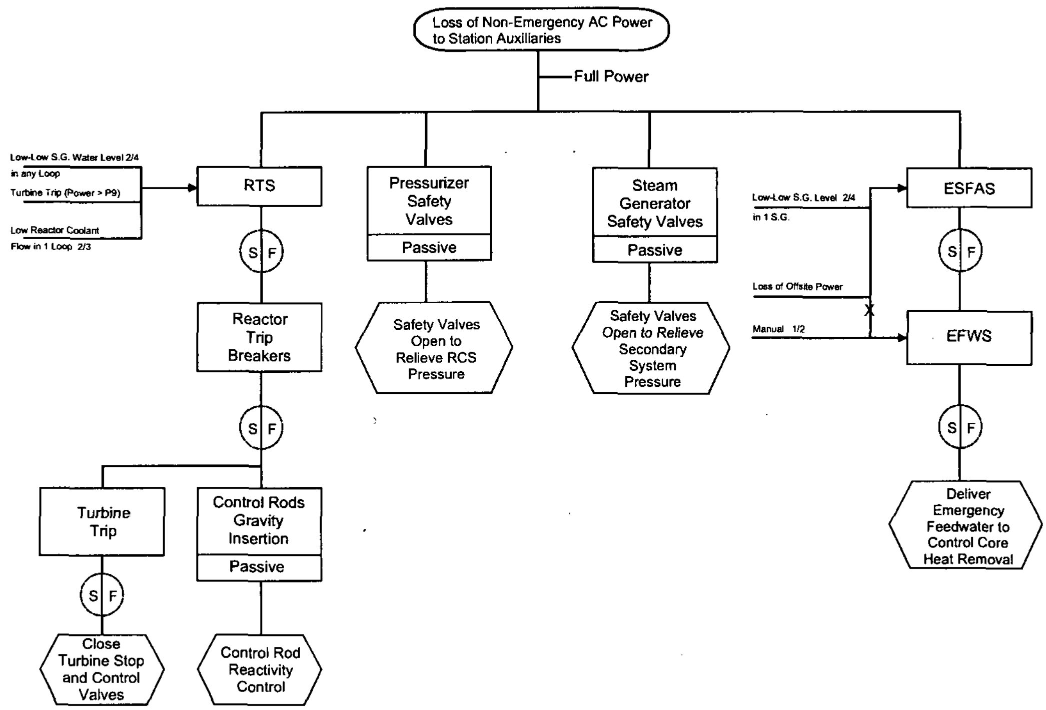


SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Feedwater System Malfunction	
		Figure 15-0-08



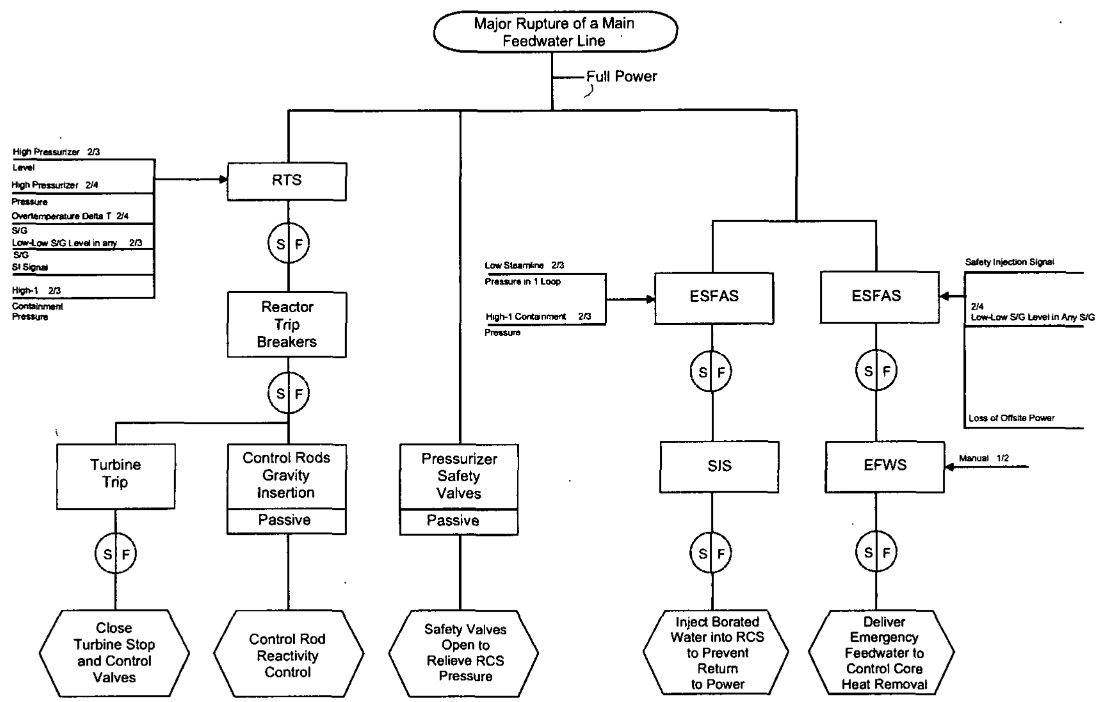
G:\WordImages_P\UFSAR\15010.ds4

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Depressurization of Main Steam System	
		Figure 15-0-10



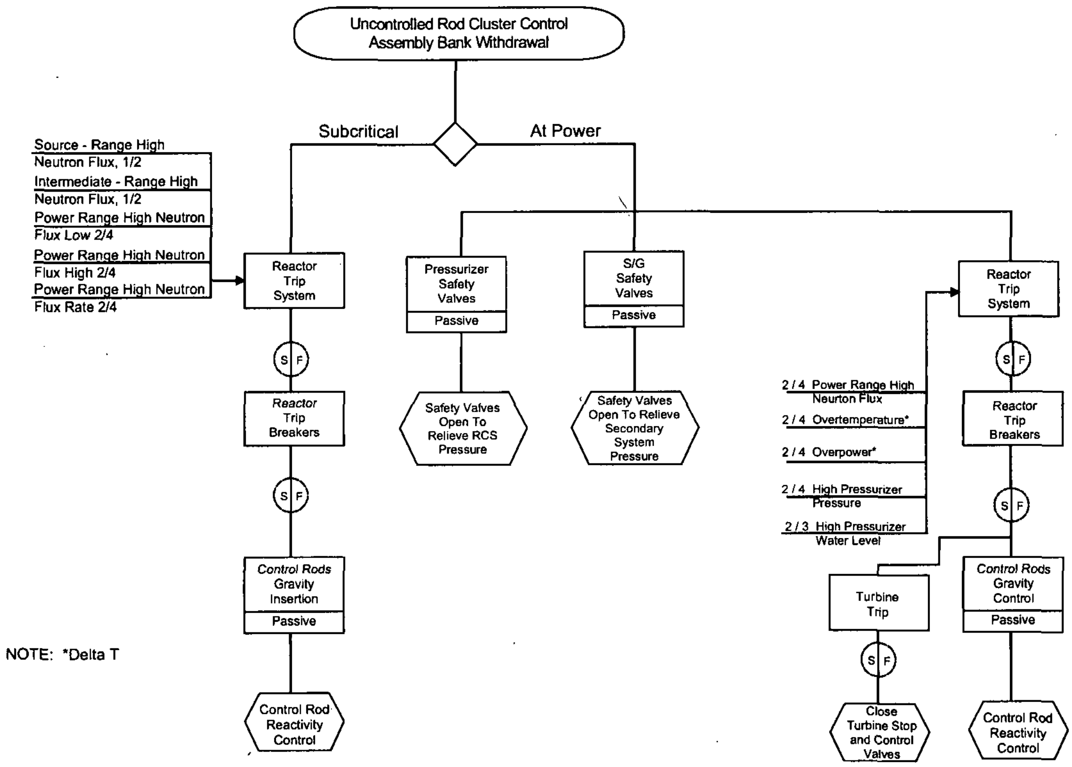
G:\Word\Images_P\UFSAR\15012.ds4

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Loss of Offsite Power to Station Auxiliaries	
		Figure 15-0-12



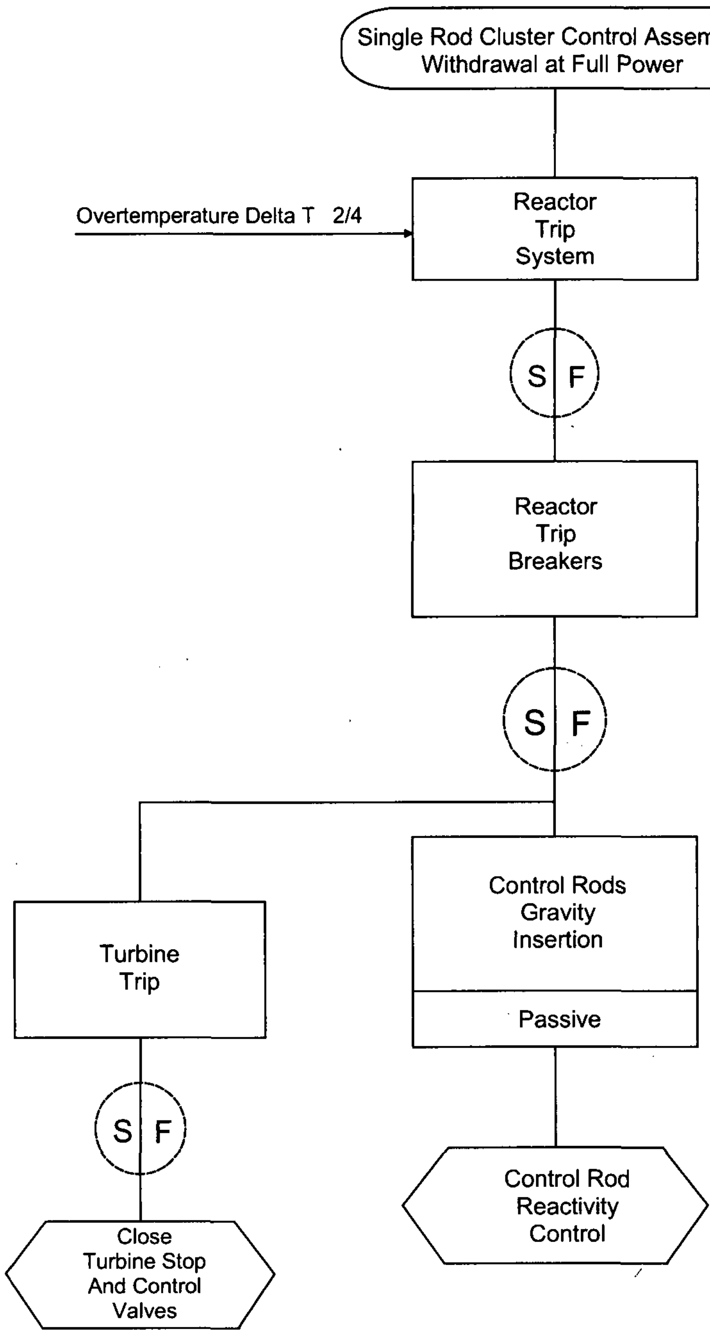
G:\Word\Images_P\UFSAR\15014.ds4

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Major Rupture of a Main Feedwater Line	
		Figure 15-0-14



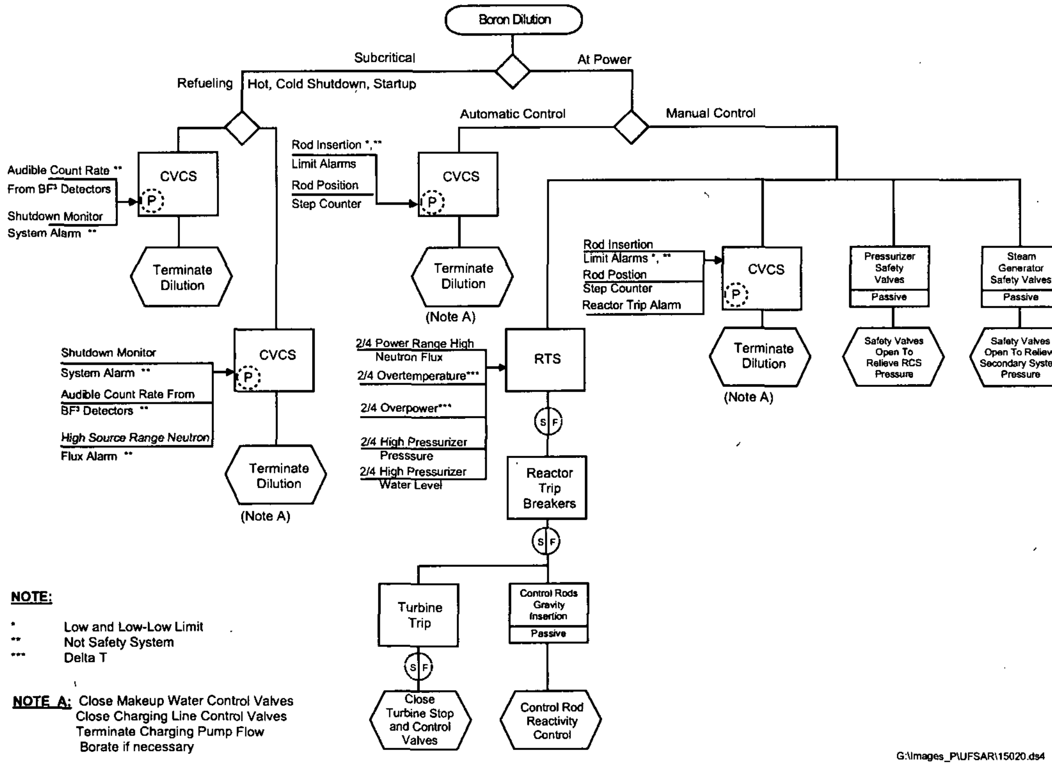
G:\Images_P\UFSAR\15016.ds4

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Uncontrolled Rod Cluster Control Assembly Bank Withdrawal	
		Figure 15-0-16

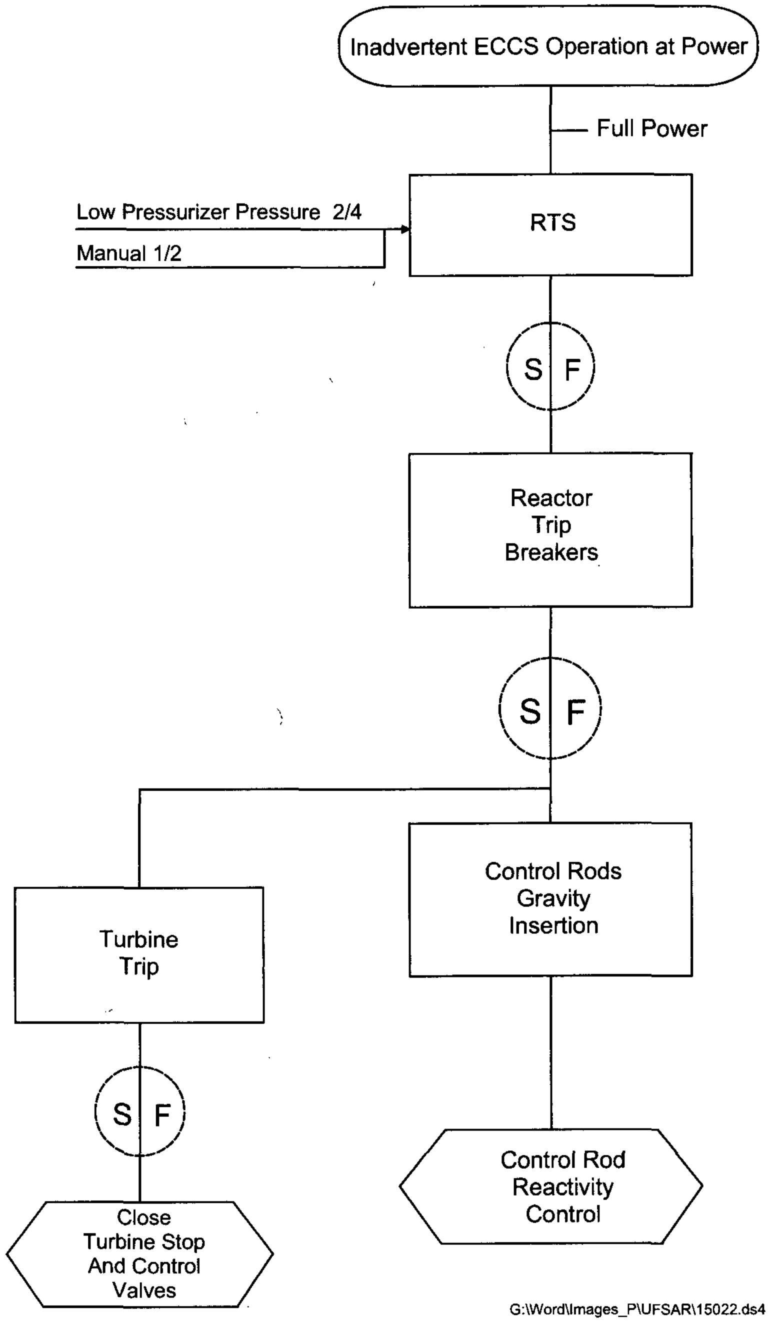


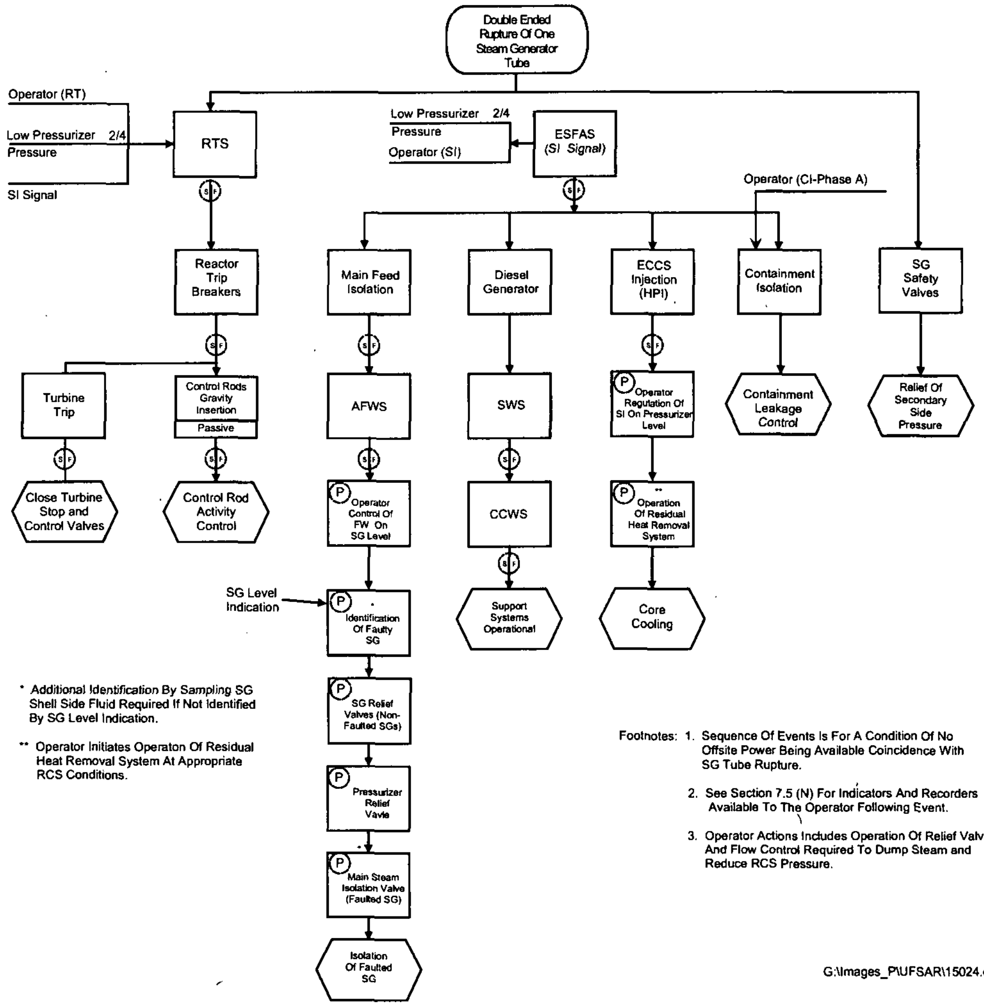
G:\Word\Images_P\UFSAR\15018.ds4

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Single Rod Cluster Control Assembly Withdrawal at Full Power	
		Figure 15-0-18



SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Boron Dilution	
		Figure 15-0-20


Inadvertent ECCS Operation at Power
Full Power
Low Pressurizer Pressure 2/4
Manual 1/2
RTS
S F
Reactor Trip Breakers
S F
Turbine Trip
Control Rods Gravity Insertion
S F
Close Turbine Stop And Control Valves
Control Rod Reactivity Control
G:\Word\Images_P\UFSAR\15022.ds4



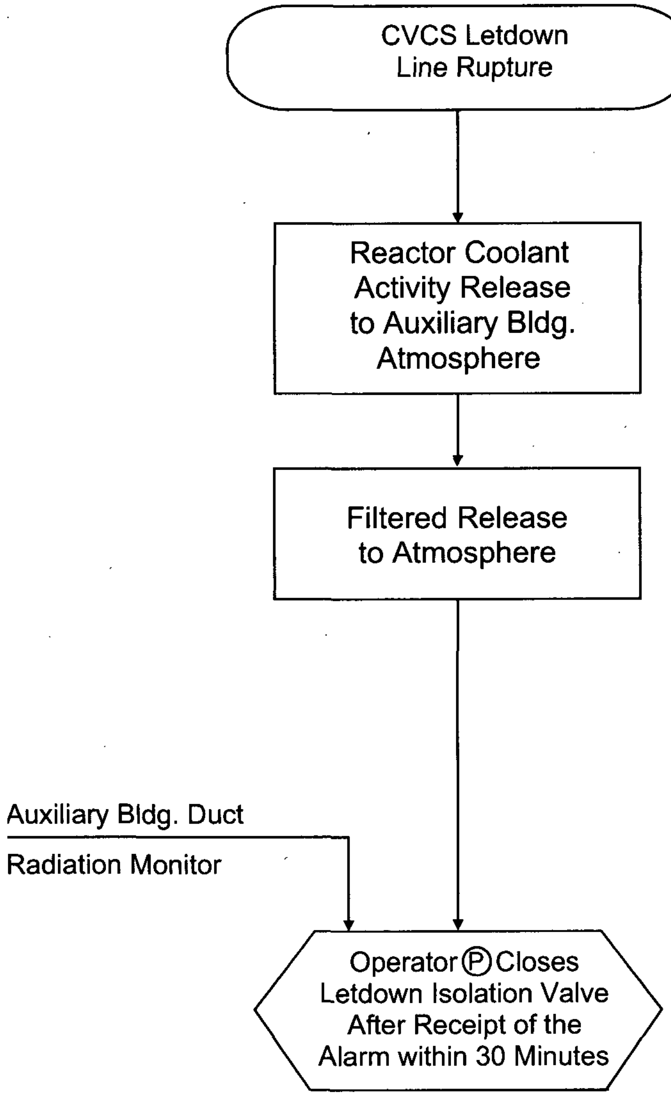
* Additional Identification By Sampling SG Shell Side Fluid Required If Not Identified By SG Level Indication.

** Operator Initiates Operaton Of Residual Heat Removal System At Appropriate RCS Conditions.

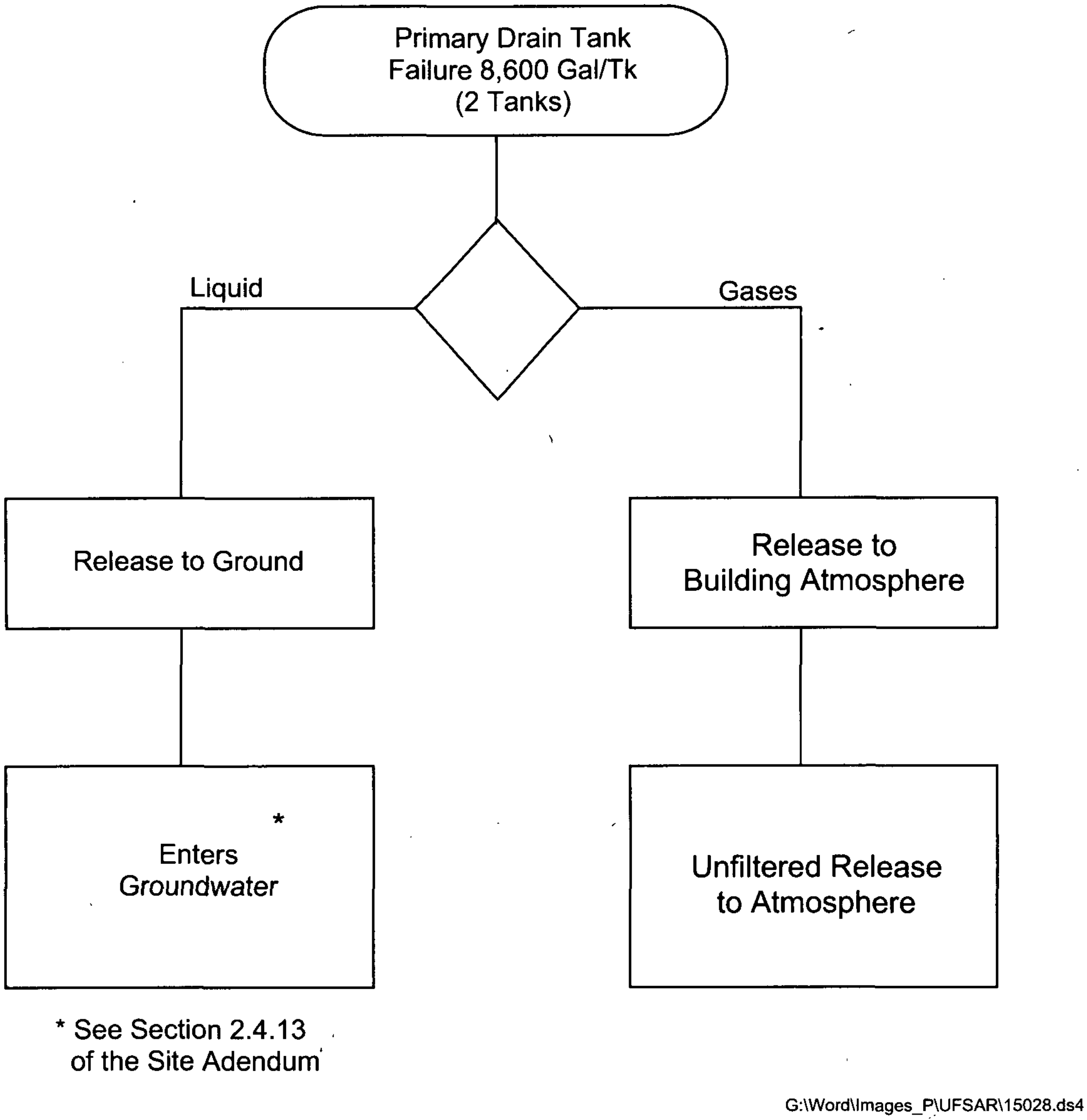
Footnotes:
1. Sequence Of Events Is For A Condition Of No Offsite Power Being Available Coincidence With SG Tube Rupture.
2. See Section 7.5 (N) For Indicators And Recorders Available To The Operator Following Event.
3. Operator Actions Includes Operation Of Relief Valves And Flow Control Required To Dump Steam and Reduce RCS Pressure.

G:\Images_P\UFSAR\15024.ds4

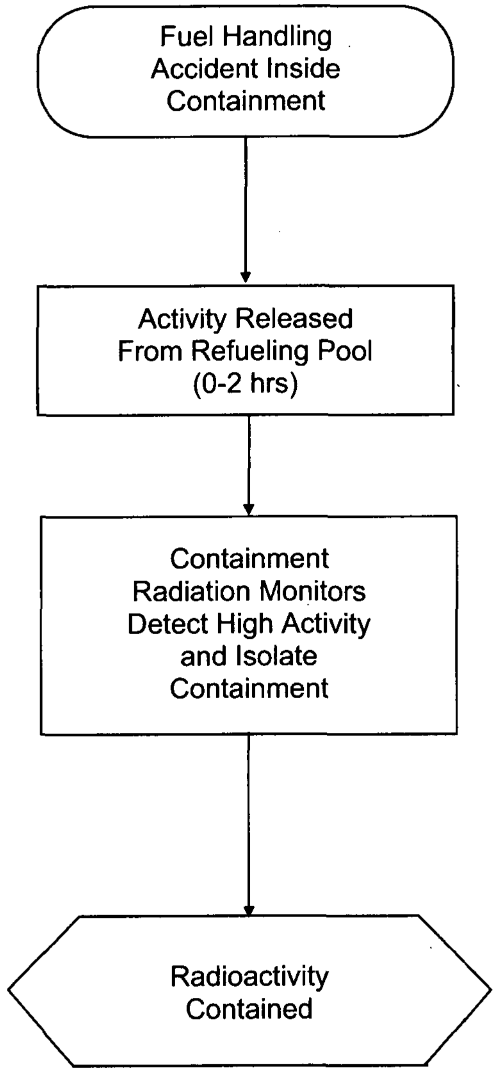
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Steam Generator Tube Rupture	
		Figure 15-0-24



G:\Word\Images_P\UFSAR\15026.ds4

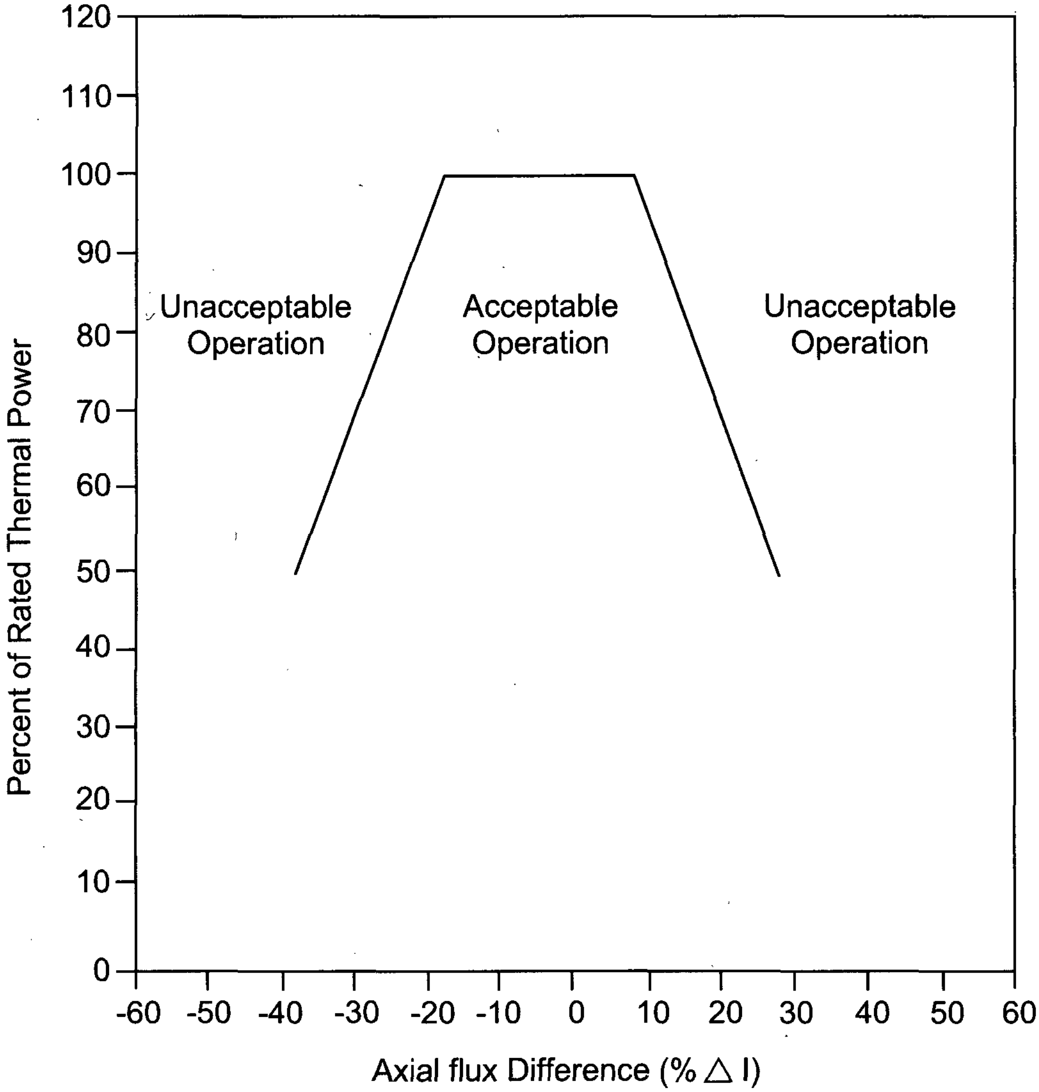


SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Floor Drain Tank Failure	
		Figure 15-0-28



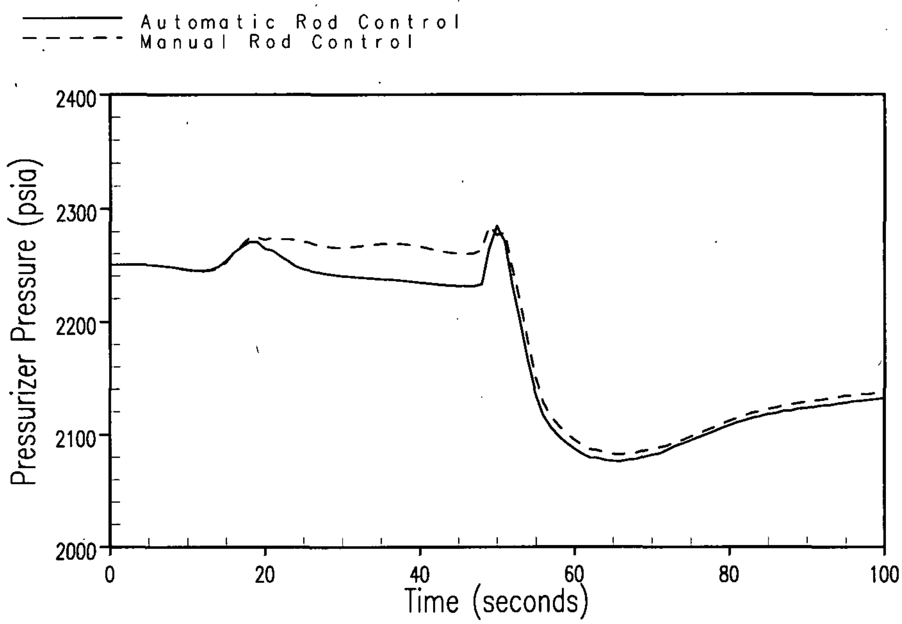
G:\Word\mages_P\UFSAR\15030.ds4

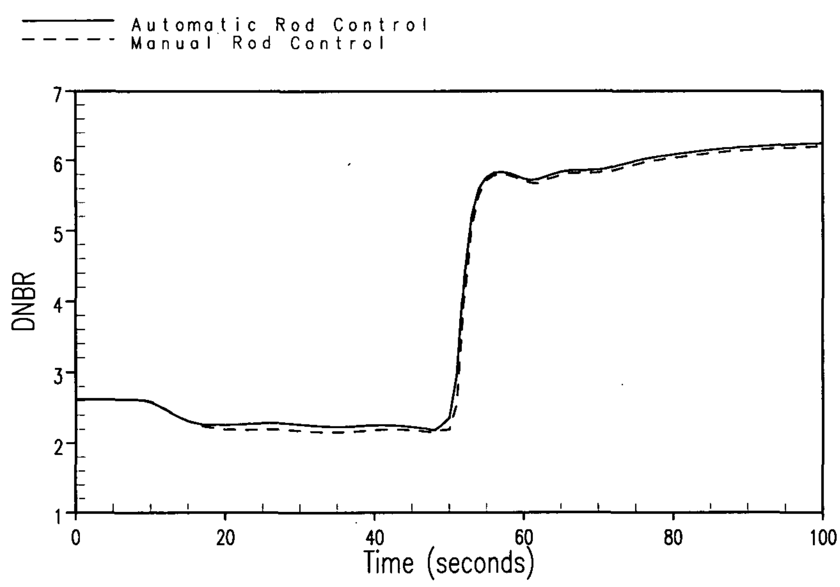
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Fuel Handling Accident Inside Containment	
		Figure 15-0-30



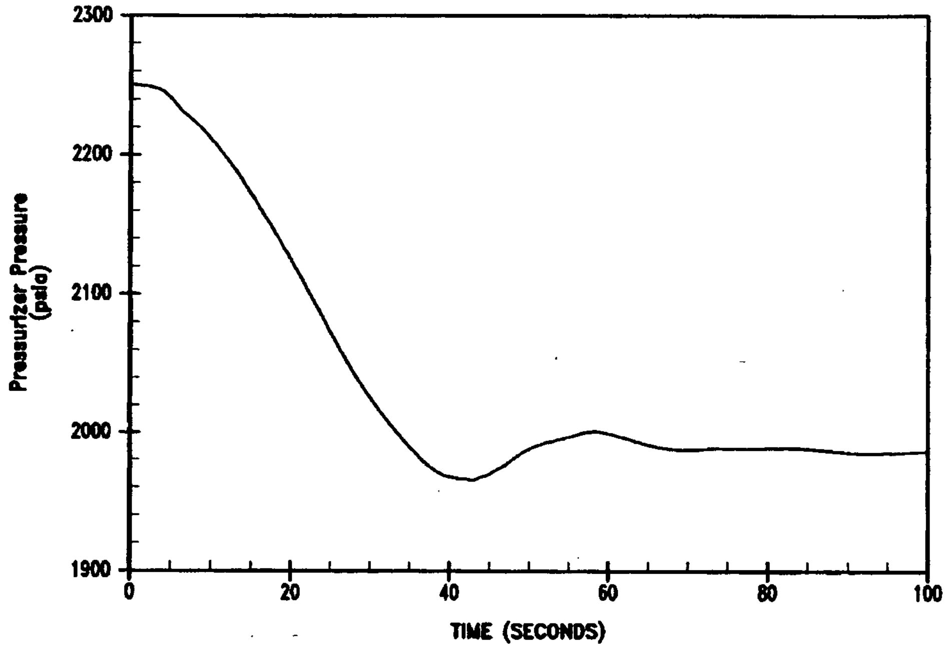
G:\Word\Images_P\UFSAR\15032.doc

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Illustration of Allowable Axial Flux Difference Limits as a Function of Rated Thermal Power
	Figure 15-0-32





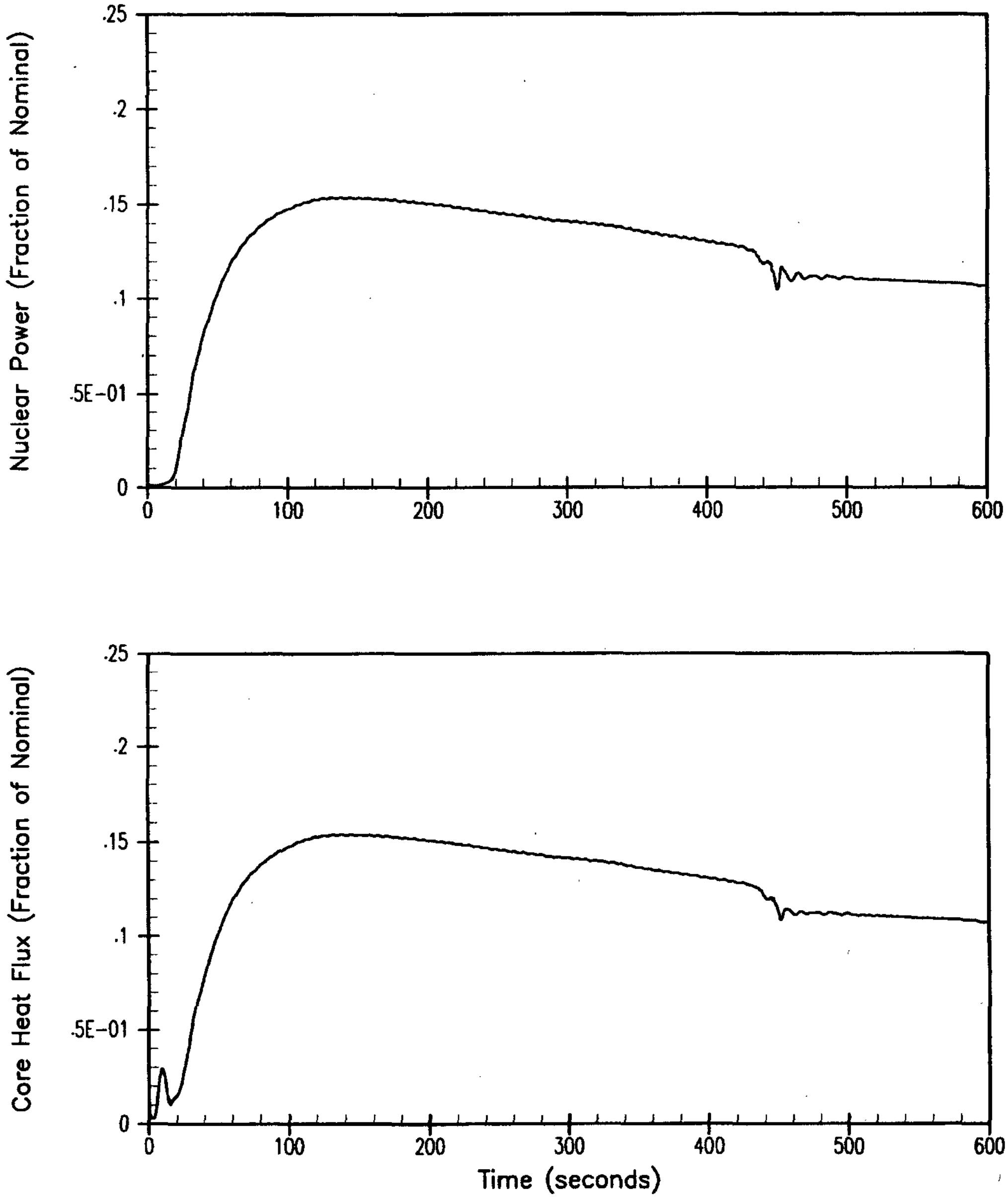
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Pressurizer Pressure and DNBR Transients for a Feedwater Control Valve Malfunction at Full Power	
	Rev. 12	Figure 15.1-1 Sh. 2 of 4


2300
2200
2100
2000
1900
Pressurizer Pressure (psia)
0
20
40
60
80
100
TIME (SECONDS)

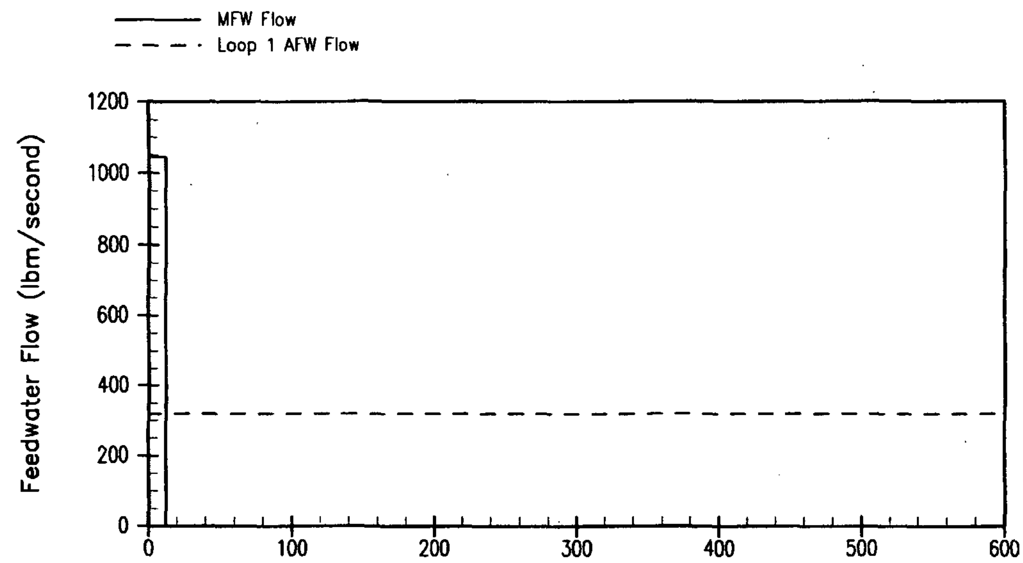
Deleted

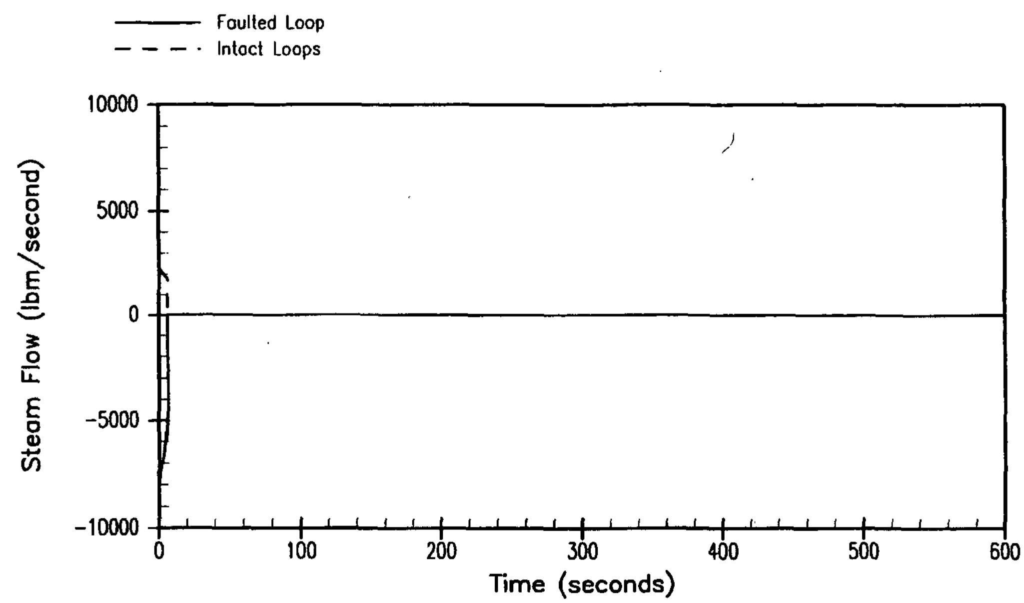
Deleted

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Deleted	
		Figure 15.1-5

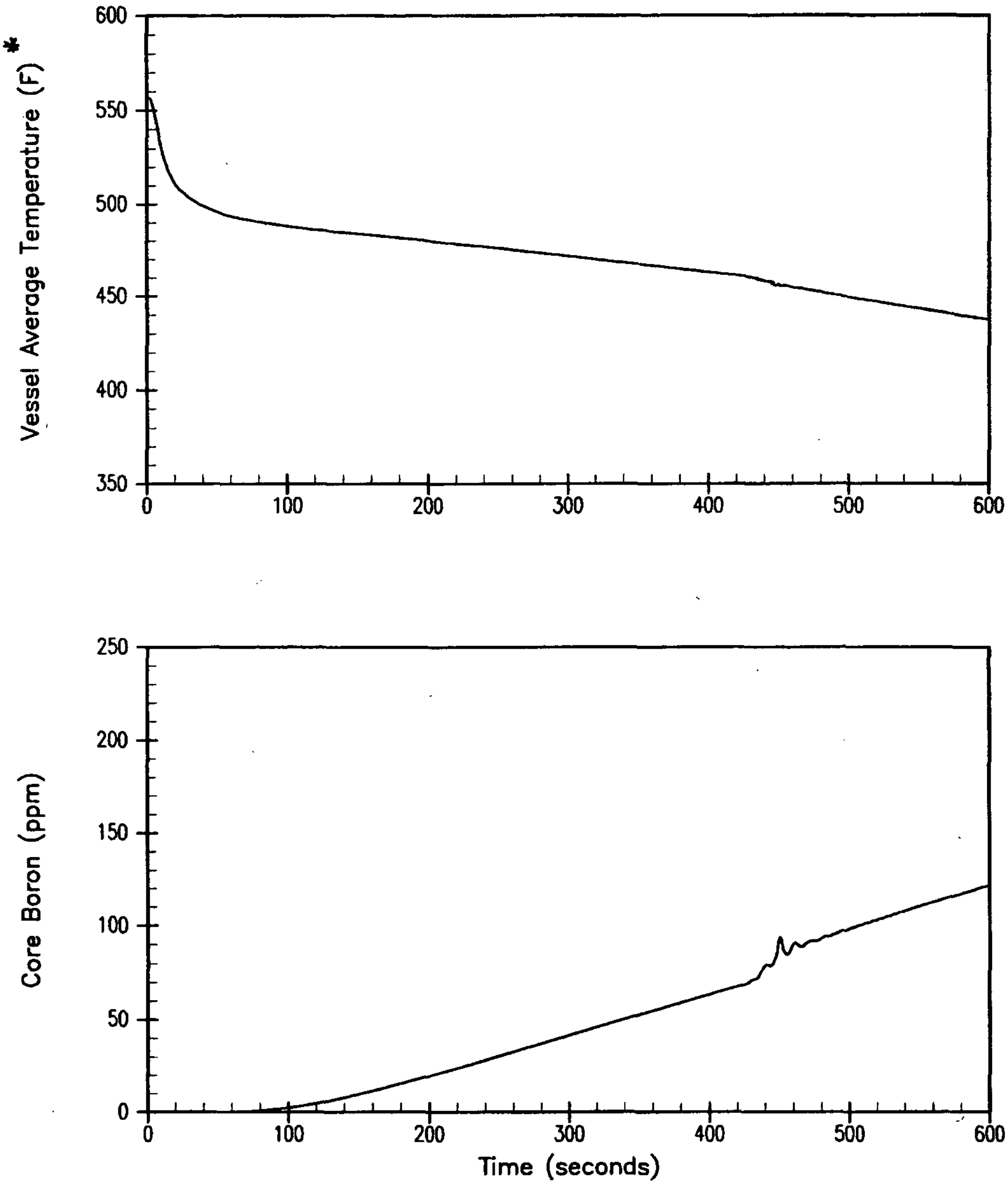


Nuclear Power (Fraction of Nominal)
.25
.2
.15
.1
.5E-01
0
0
100
200
300
400
500
600
Core Heat Flux (Fraction of Nominal)
.25
.2
.15
.1
.5E-01
0
0
100
200
300
400
500
600
Time (seconds)





SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Feedwater Flow and Steam Flow Transients for a Steam Line Rupture at Zero Power with Offsite Power Available	
		Figure 15.1-6 Sh. 4 of 6



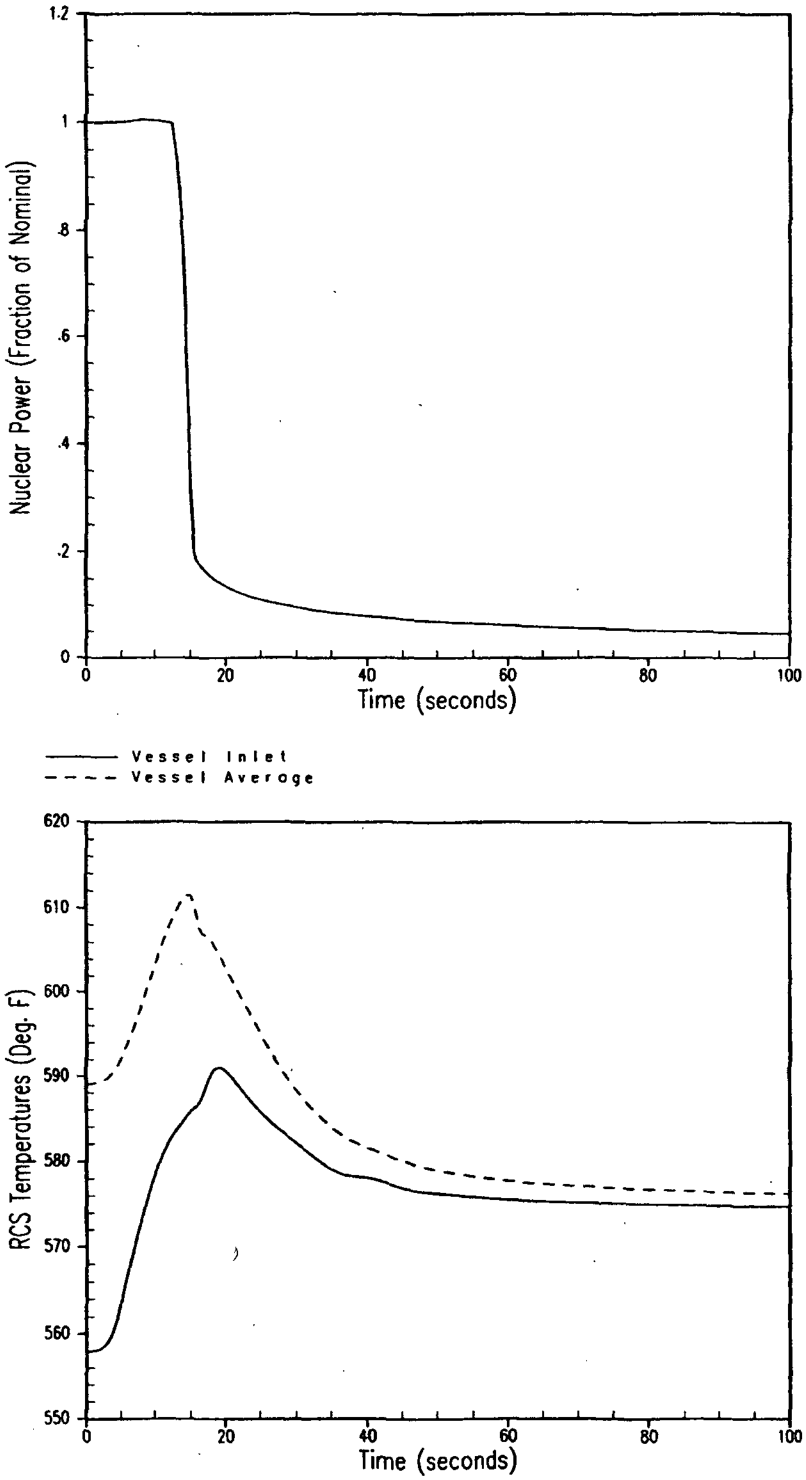
*Average of Loop Average Temperatures

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Core Average Temperature and Core Boron Transients for a Steam Line Rupture at Zero Power with Offsite Power Available	
		Figure 15.1-6 Sh. 6 of 6

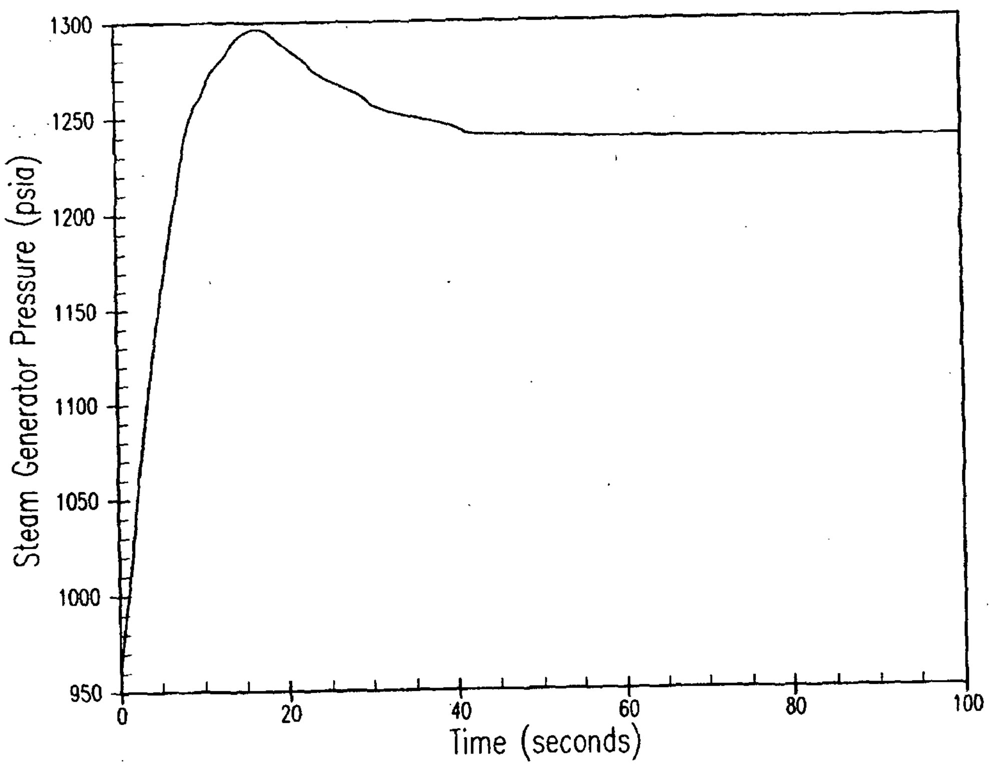
Deleted

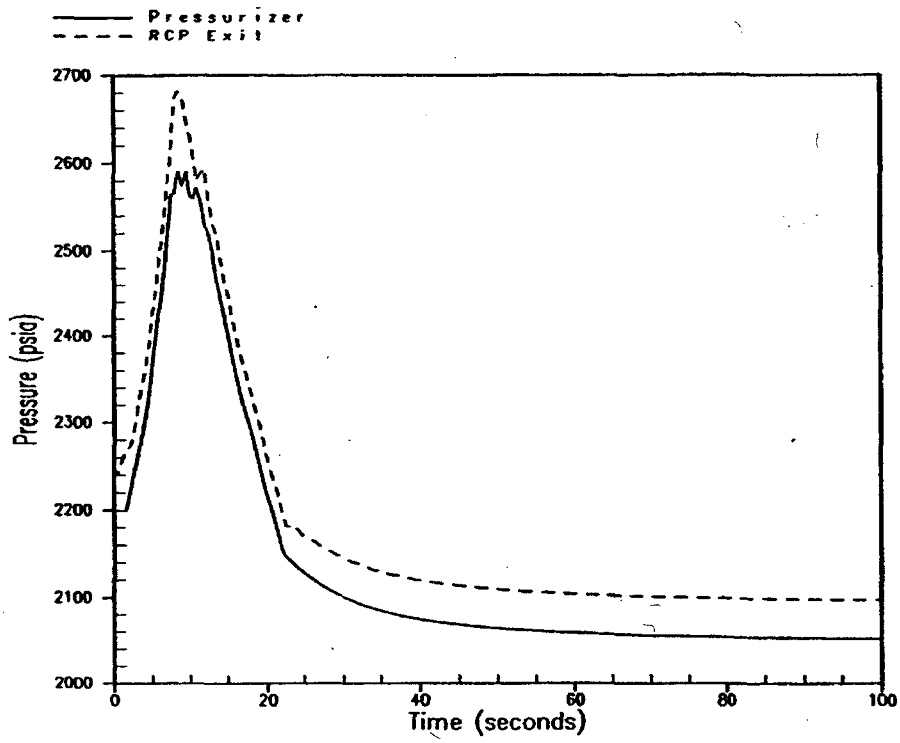
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Deleted	
		Figure 15.1-8

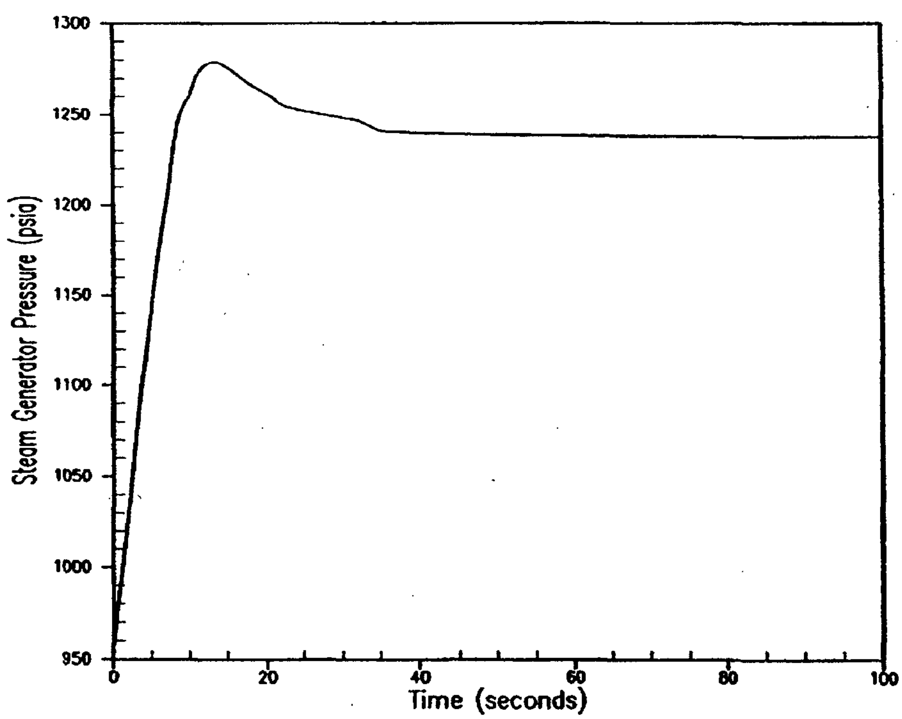
Deleted



SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Nuclear Power and RCS Temperature Transients for a Turbine Trip Event With Pressure Control	
		Figure 15.2-1 Sh. 1 of 5

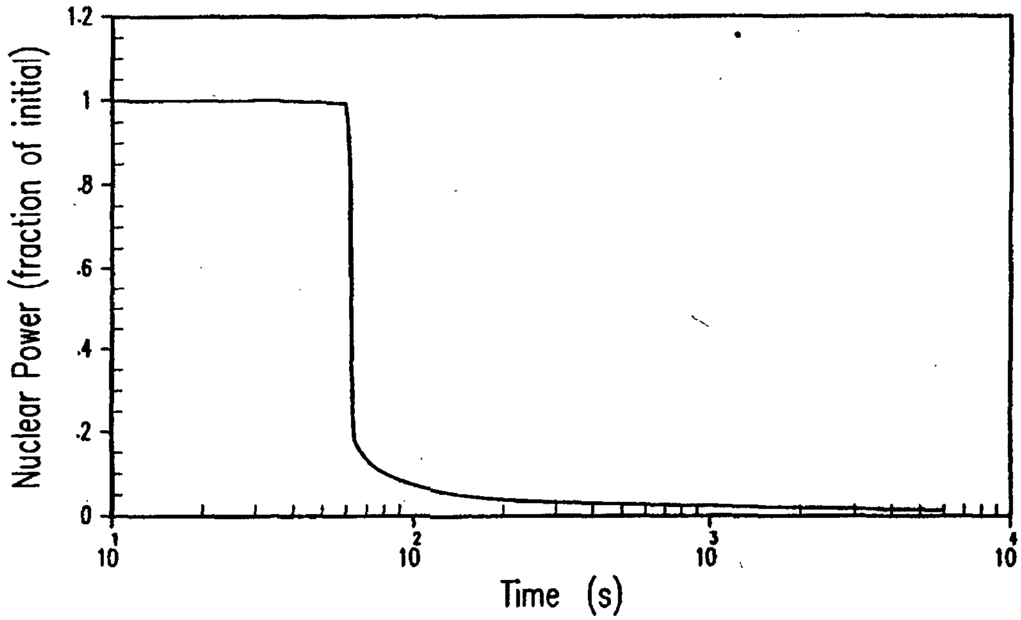

Steam Generator Pressure (psia)
1300
1250
1200
1150
1100
1050
1000
950
0
20
40
60
80
100
Time (seconds)

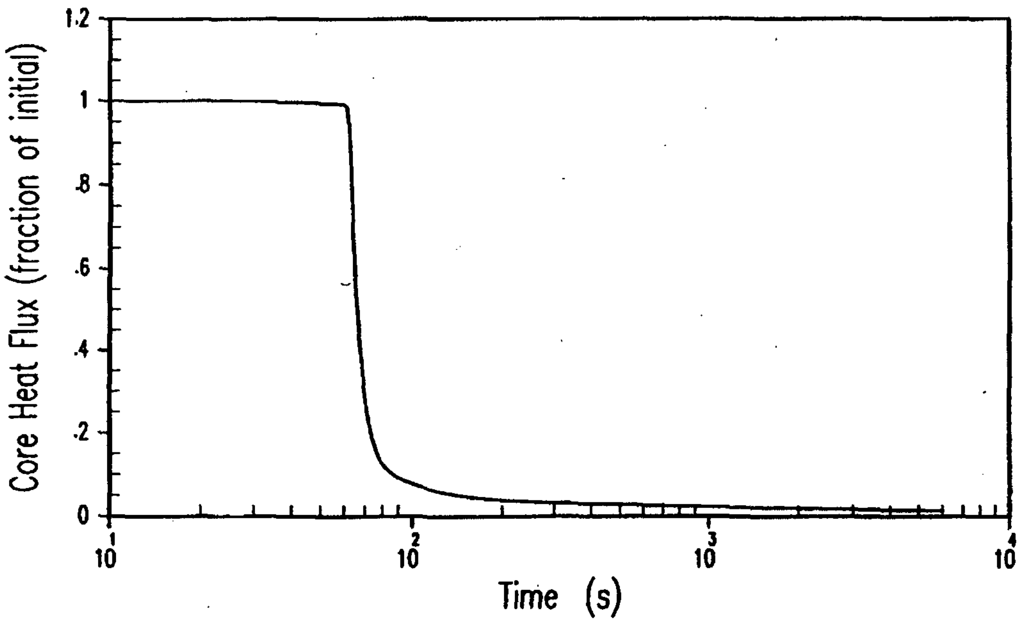




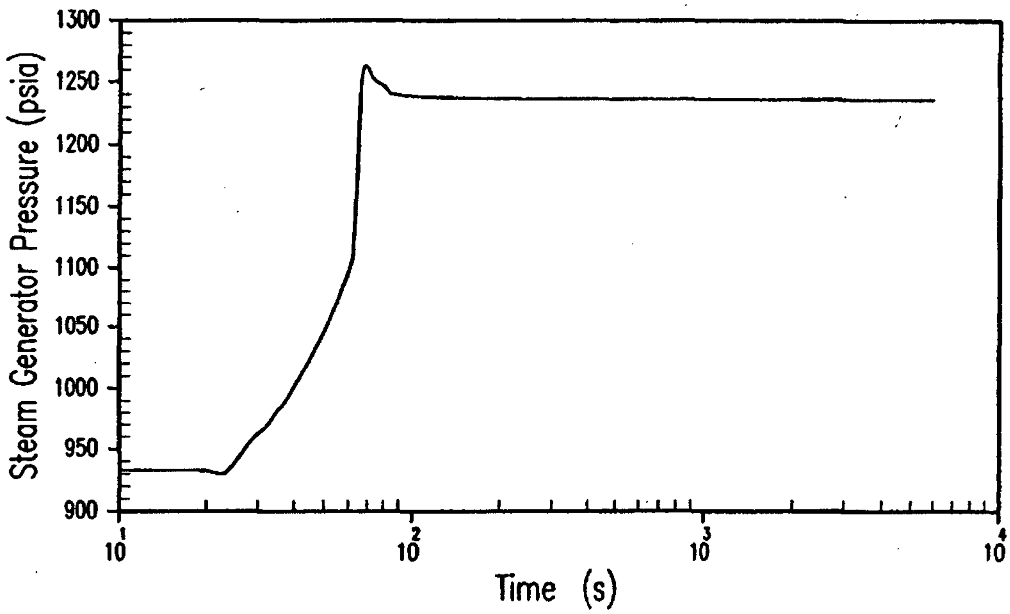
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Nuclear Power and RCS Temperature Transients for a Turbine Trip Event Without Pressure Control
	Figure 15.2-1 Sh. 5 of 5

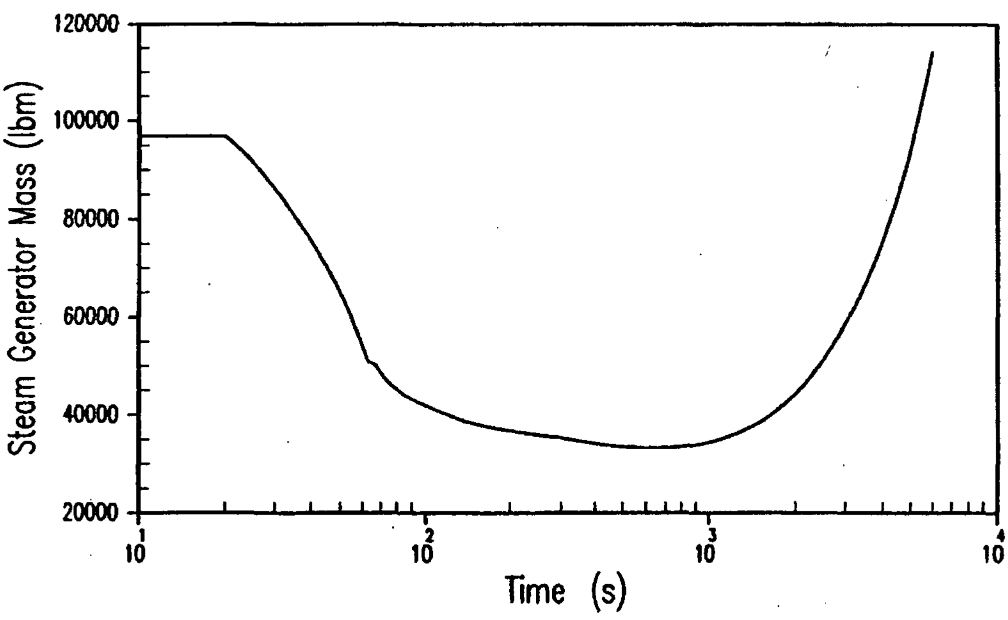
Deleted



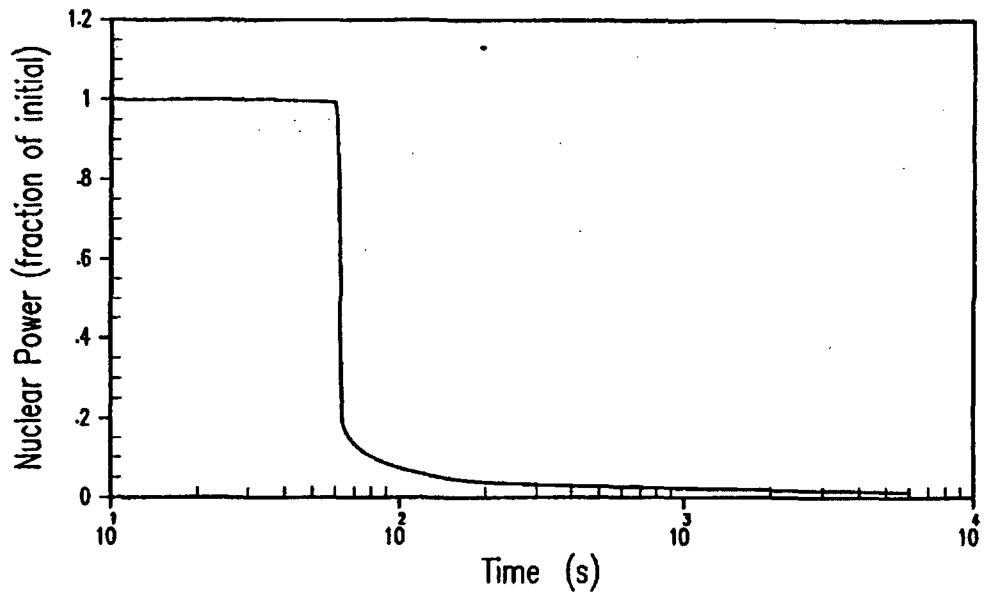


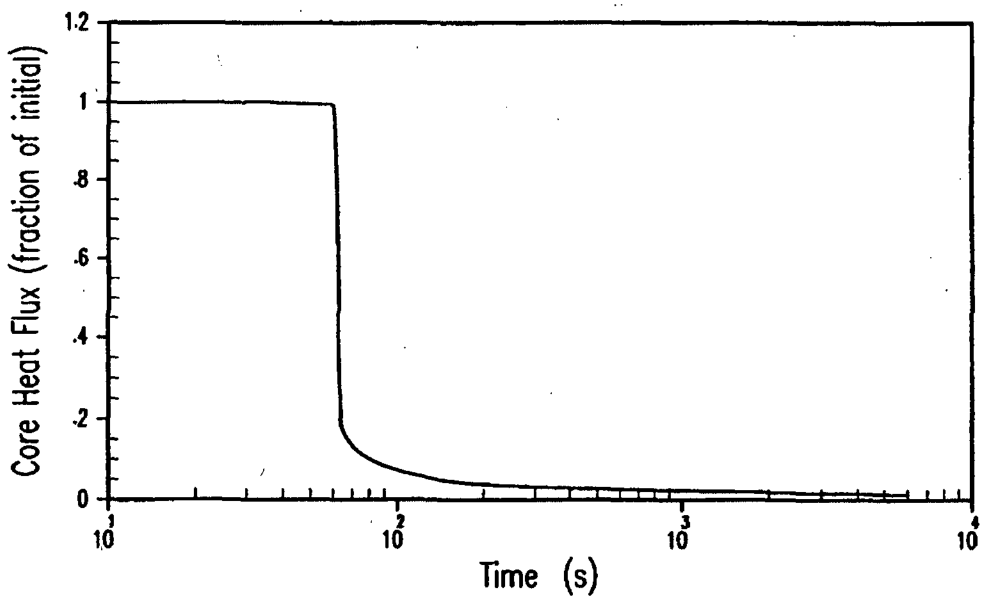
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Nuclear Power and Core Heat Flux Transients for a Loss of Non-Emergency AC to the Station Auxiliaries
	Figure 15.2-5 Sh. 1 of 4



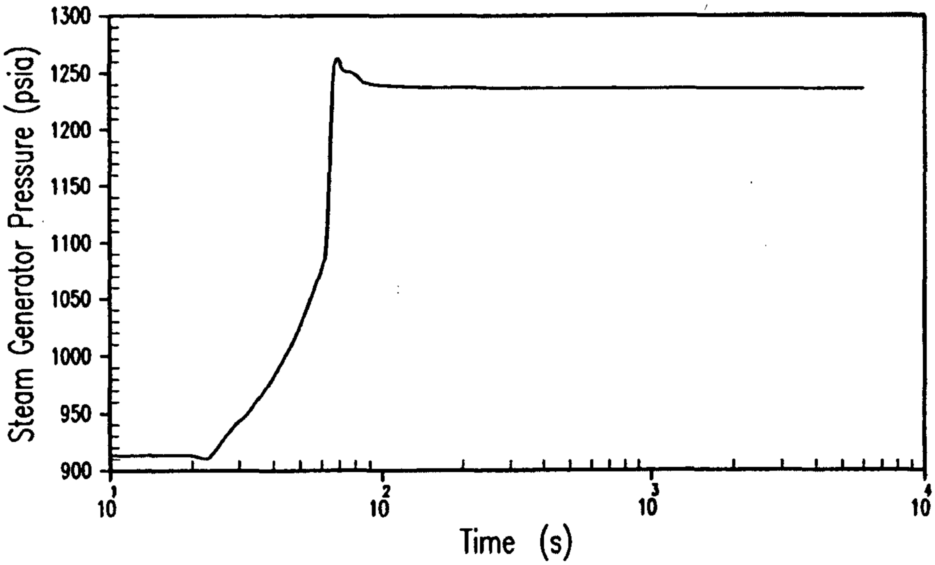


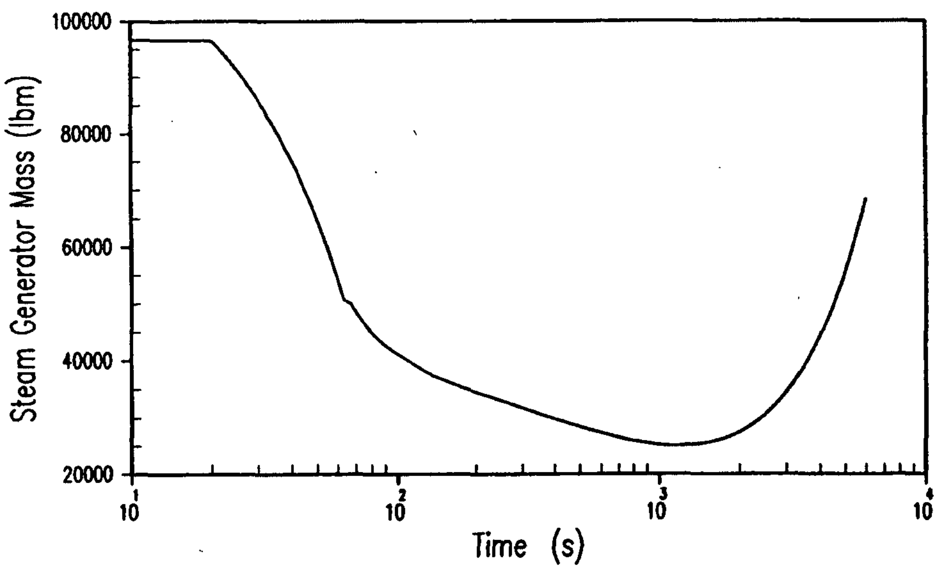
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	SG Pressure and SG Mass Transients for a Loss of Non-Emergency AC to the Station Auxiliaries	
		Figure 15.2-5 Sh. 3 of 4



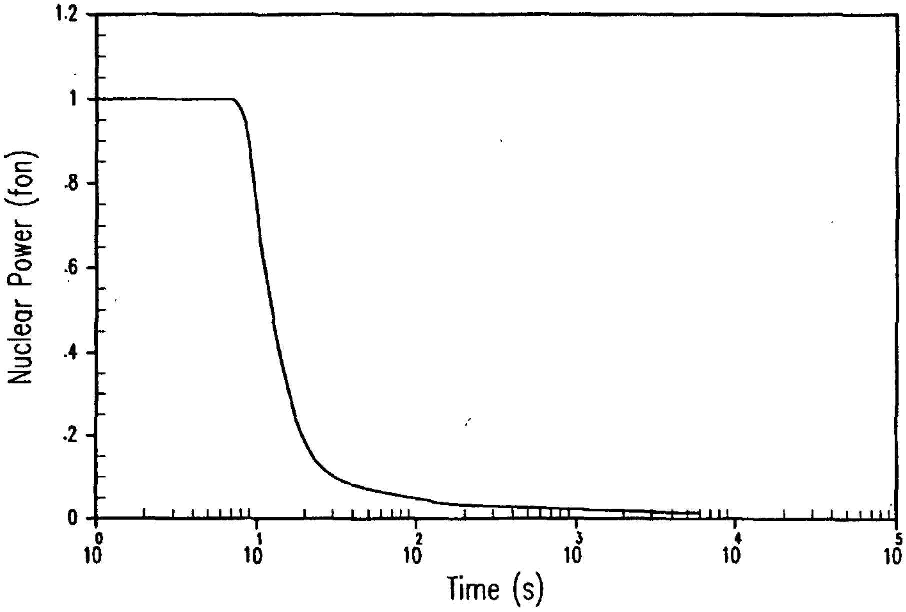


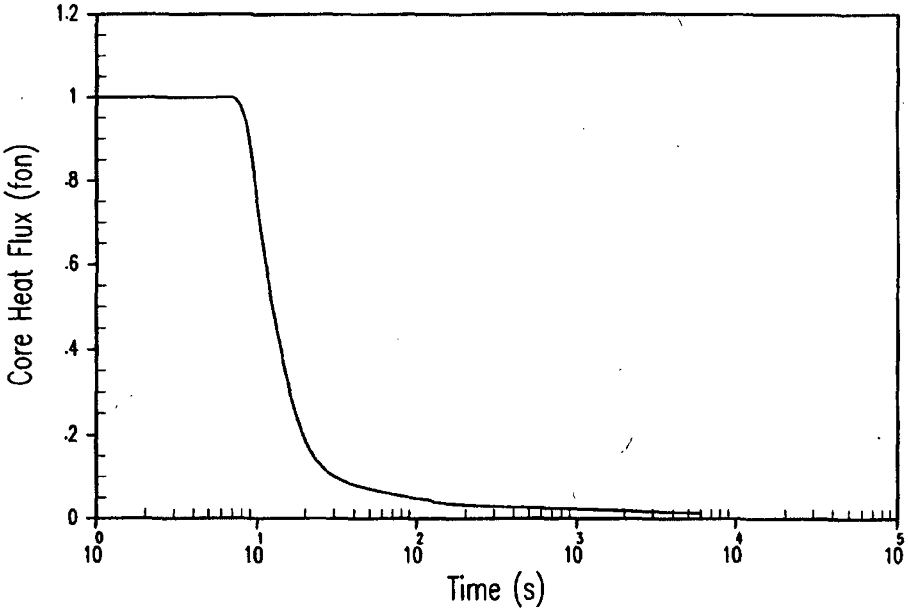
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Nuclear Power and Core Heat Flux Transients for a Loss of Normal Feedwater	
		Figure 15.2-6 Sh. 1 of 4



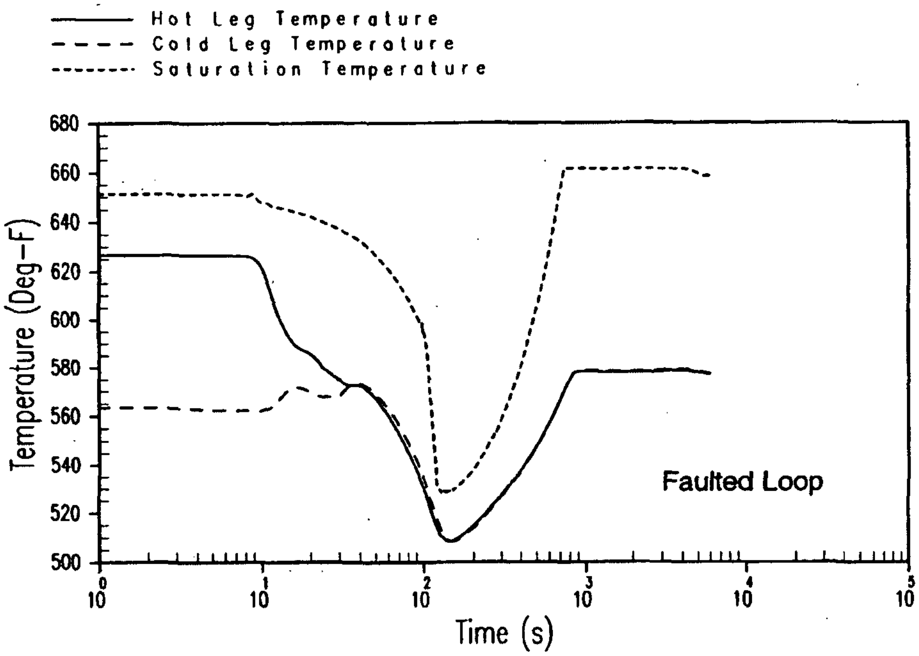


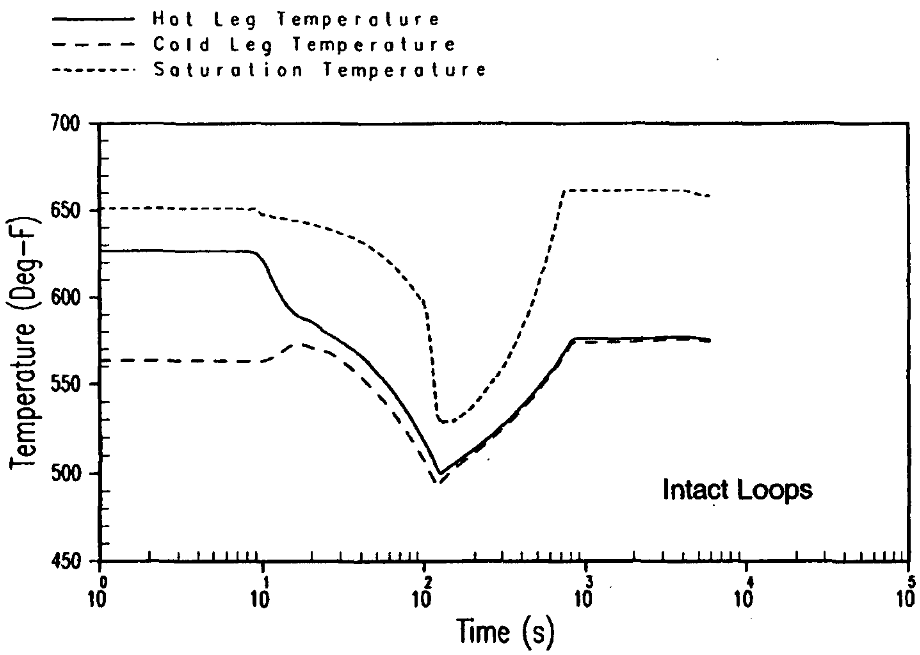
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	SG Pressure and SG Mass Transients for a Loss of Normal Feedwater	
		Figure 15.2-6 Sh. 3 of 4



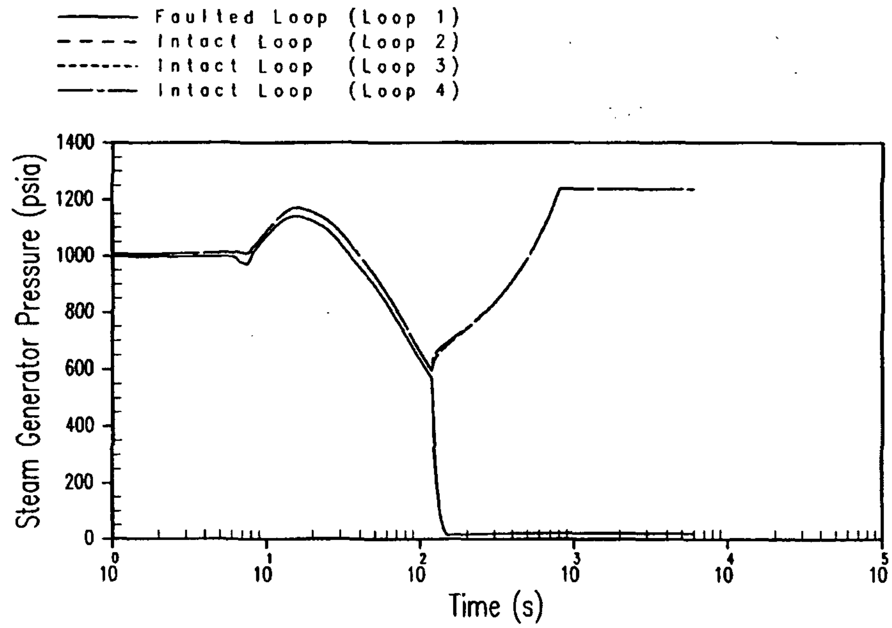


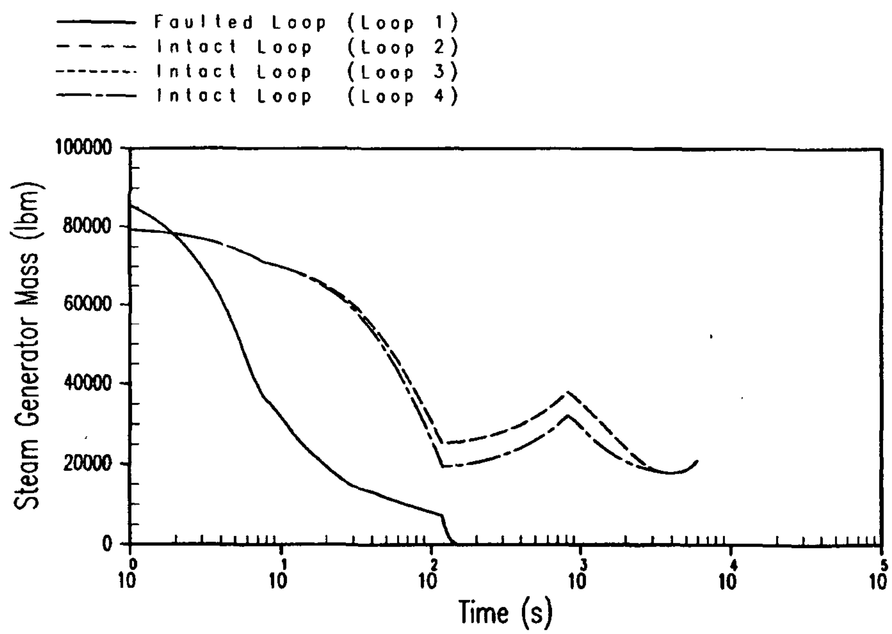
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Nuclear Power and Core Heat Flux Transients for a Feedwater System Pipe Break
	Figure 15.2-7 Sh. 1 of 5





SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Loop 1 (faulted) and Loop 2 (unfaulted) Temperature Transients for a Feedwater System Pipe Break
	Figure 15.2-7 Sh. 3 of 5





SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Steam Generator Pressure and Steam Generator Mass Transients for a Feedwater System Pipe Break
	Figure 15.2-7 Sh. 5 of 5

Deleted

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Deleted	
		Figure 15.2-9

Deleted

Deleted

Deleted

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Deleted	
		Figure 15.2-15

Deleted

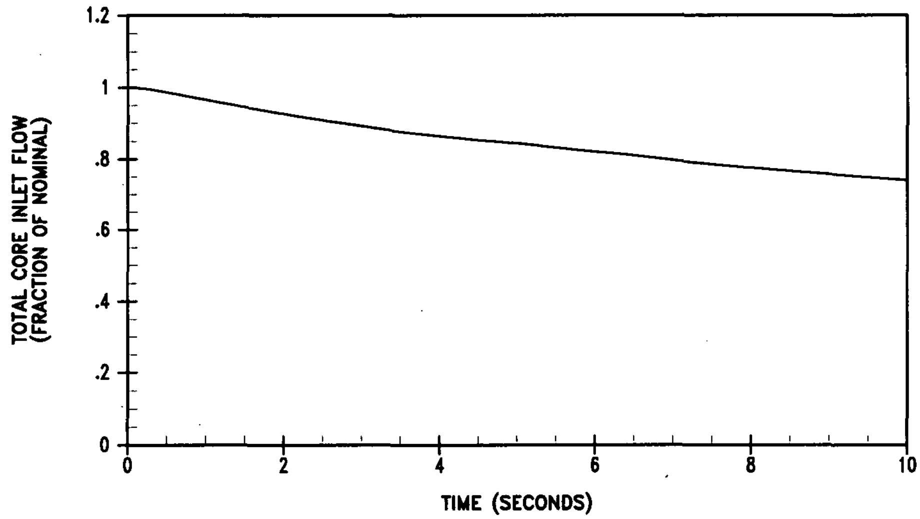
Deleted

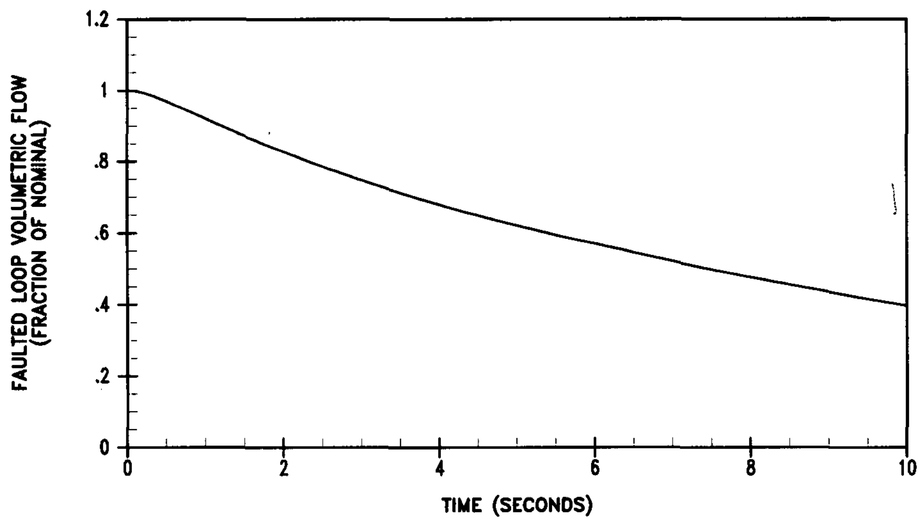
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Deleted	
		Figure 15.2-19

Deleted

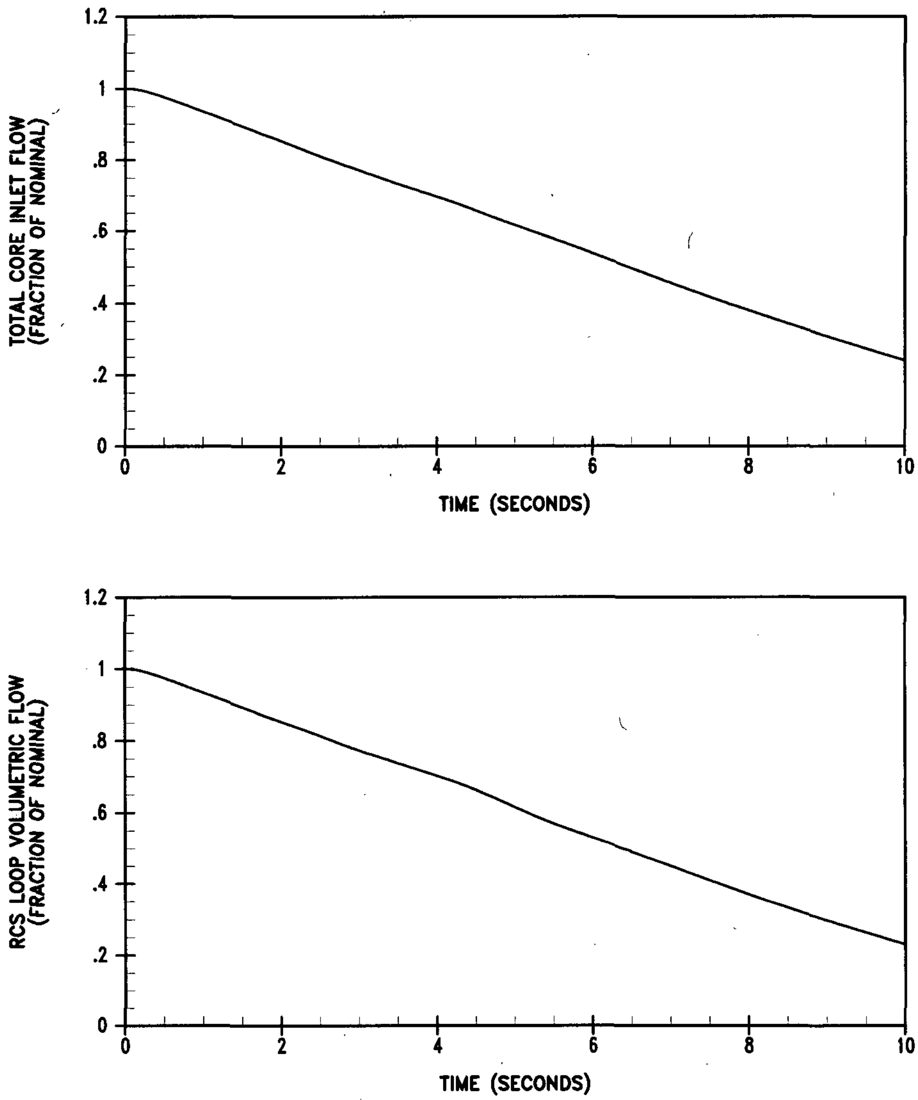
Deleted

Deleted

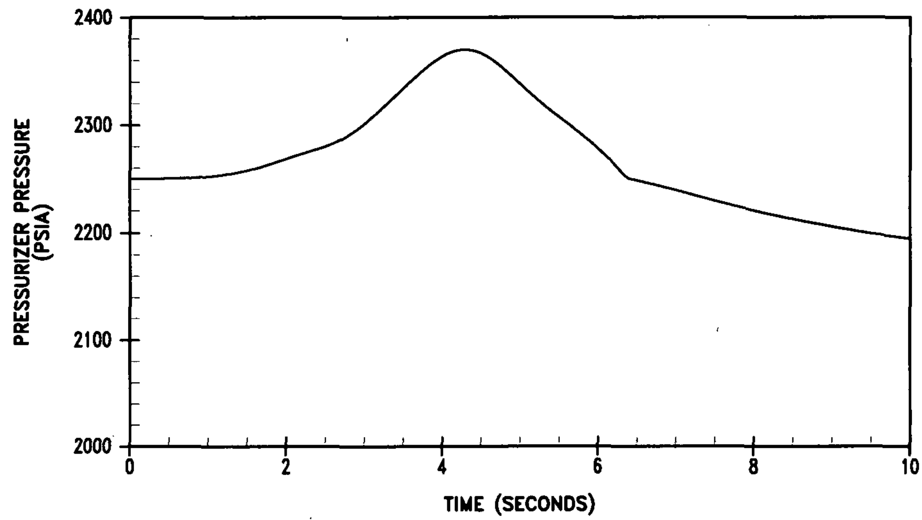


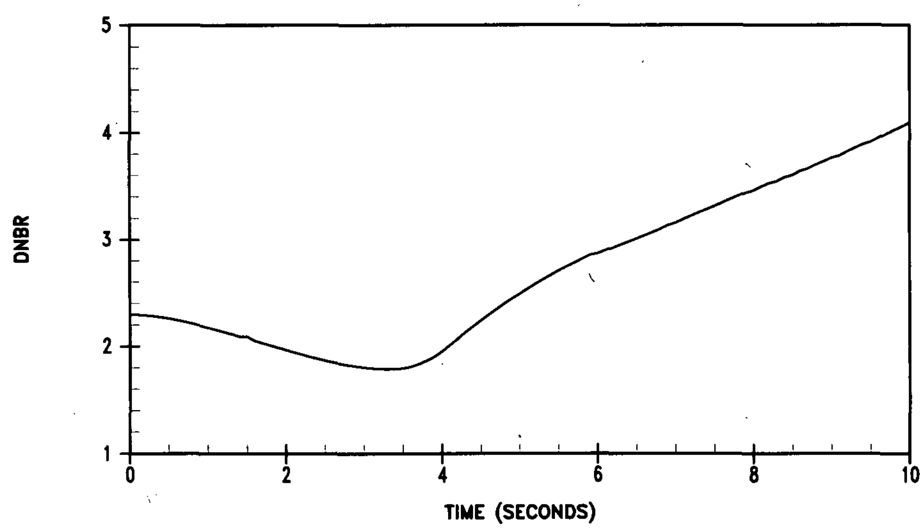


SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Core Flow and Faulted Loop Flow Transients for a Partial Loss of Forced Reactor Coolant Flow (4 loops in operation, 2 RCPs coasting down)	
	Rev. 12	Figure 15.3-1 Sh. 2 of 2

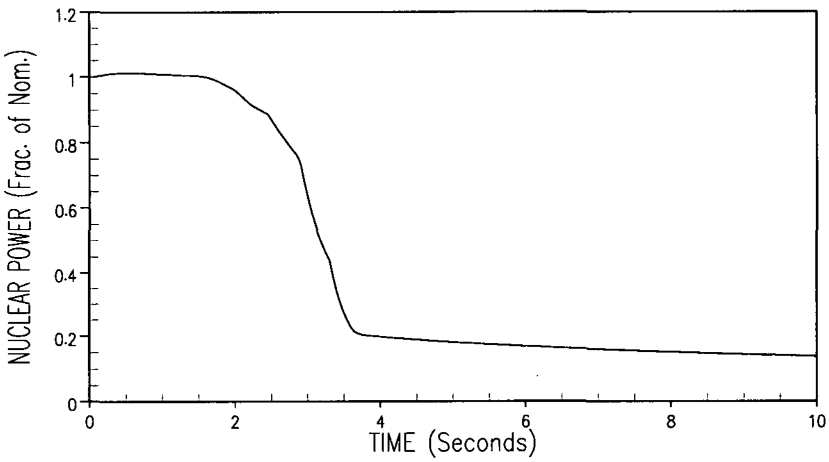


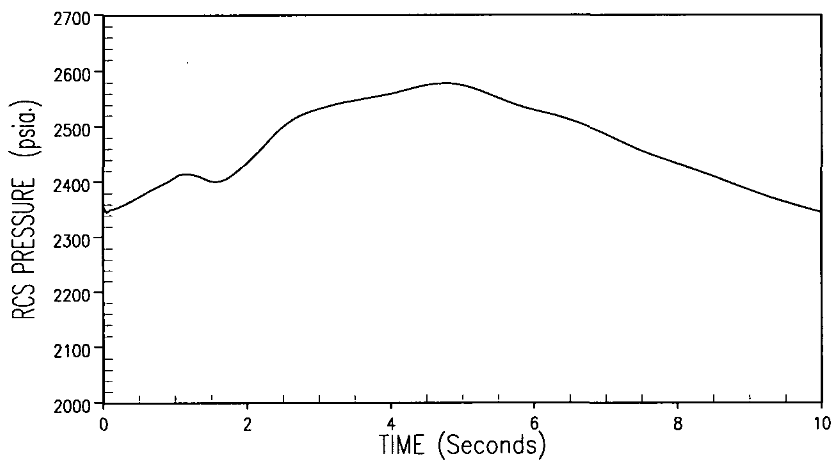
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Core Flow and Faulted Loop Flow Transients for a Complete Loss of Forced Reactor Coolant Flow (4 loops in operation, 4 RCPs coasting down)	
	Rev. 12	Figure 15.3-3





SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Reactor Coolant Pressure and DNBR Transients for a Complete Loss of Forced Reactor Coolant Flow (4 loops in operation, 4 RCPs coasting down)	
	Rev. 12	Figure 15.3-5





SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Nuclear Power and RCS Pressure Transients for a Locked Rotor	
	Rev. 12	Figure 15.3-6 Sh. 2 of 3

Deleted

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Deleted	
		Figure 15-3-7

Deleted

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Deleted	
		Figure 15.3-9

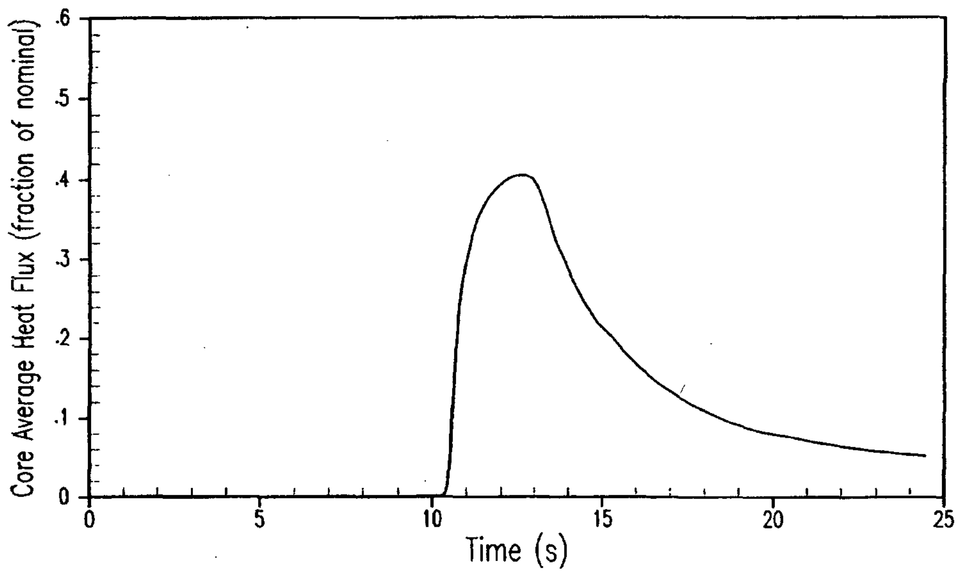
Deleted

Deleted

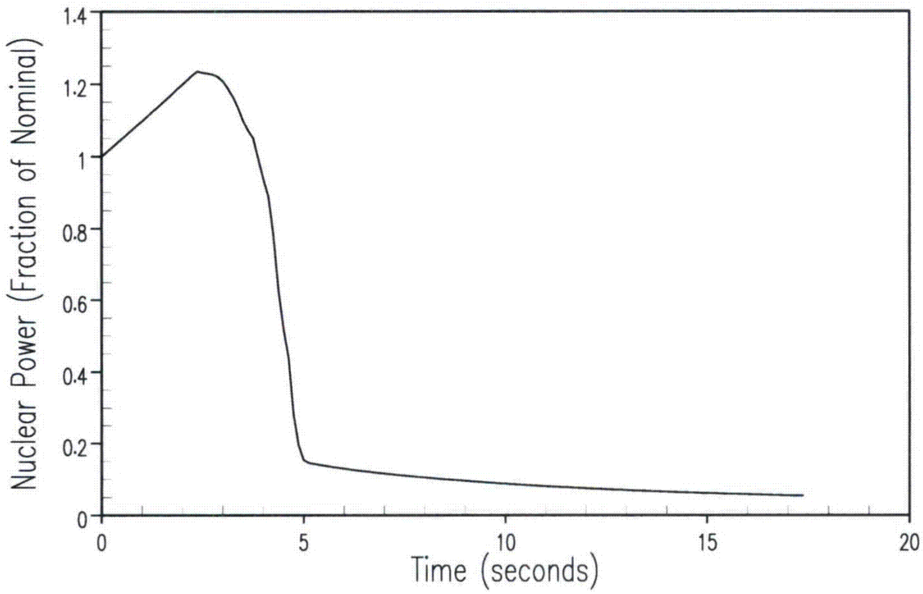
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Deleted	
		Figure 15.3-13

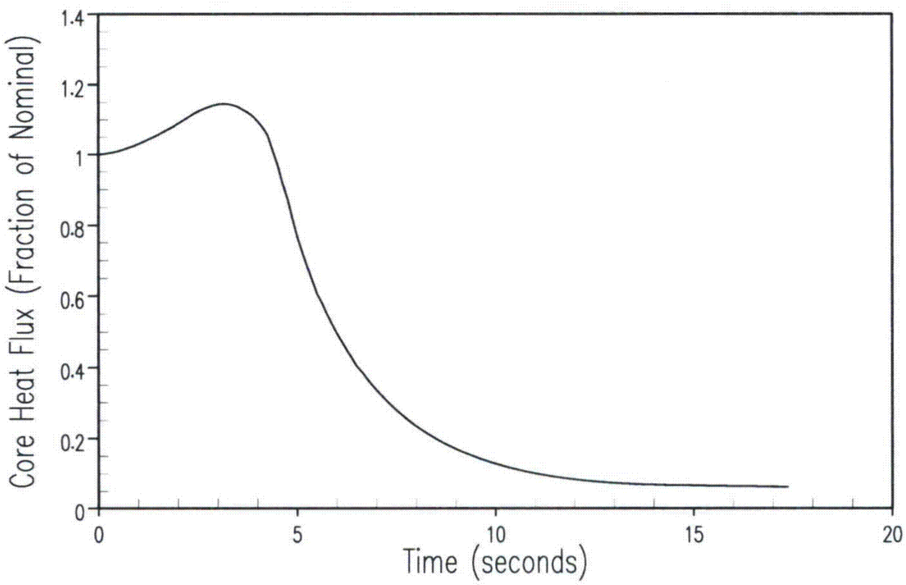
Deleted

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Deleted	
		Figure 15.3-15

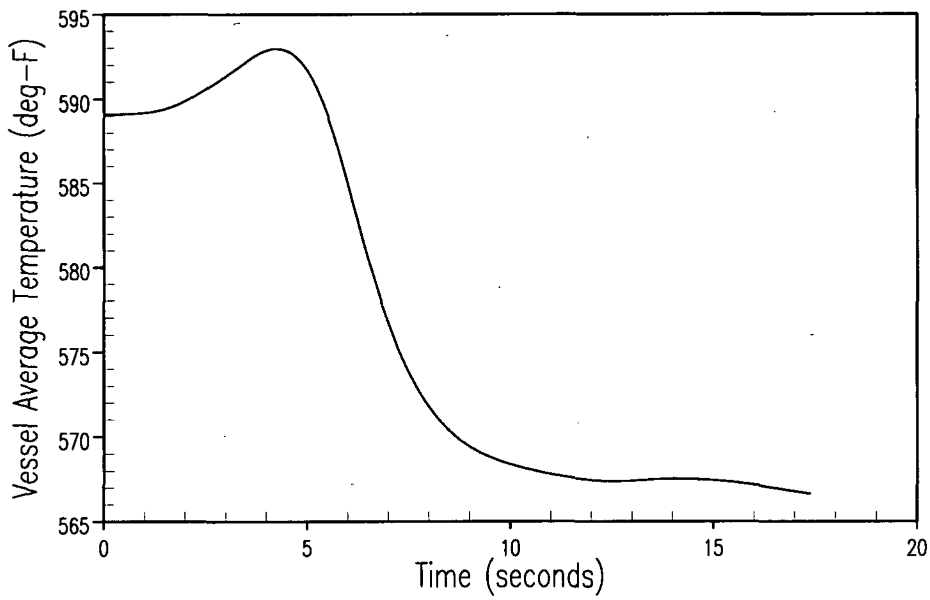


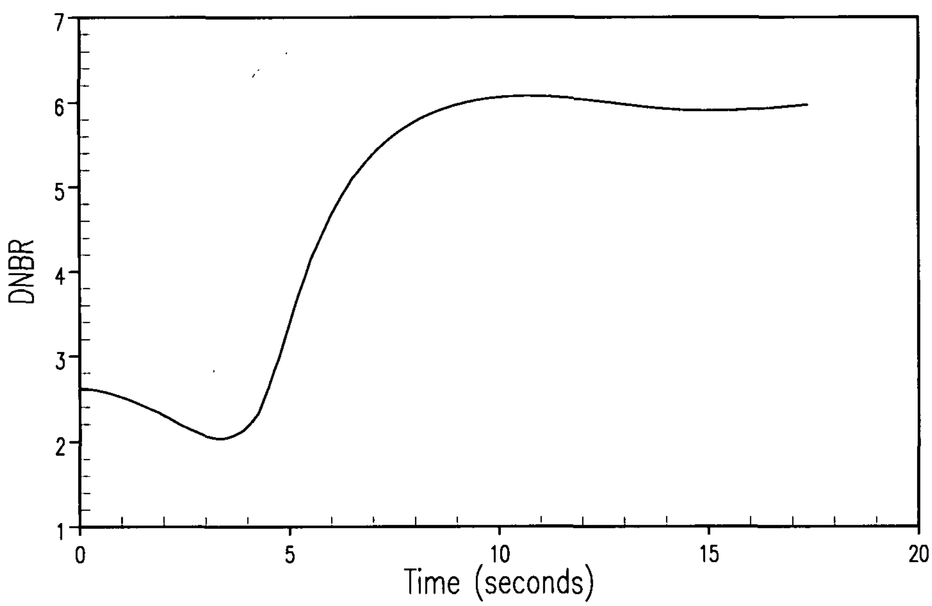
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Core Average Heat Flux Transients for Uncontrolled Rod Withdrawal from a Subcritical Condition	
		Figure 15.4-2



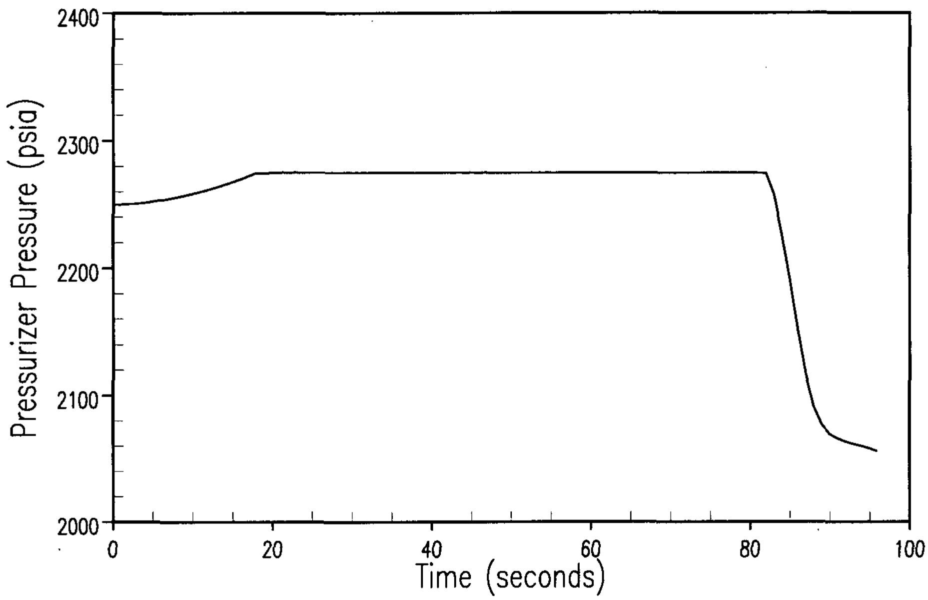


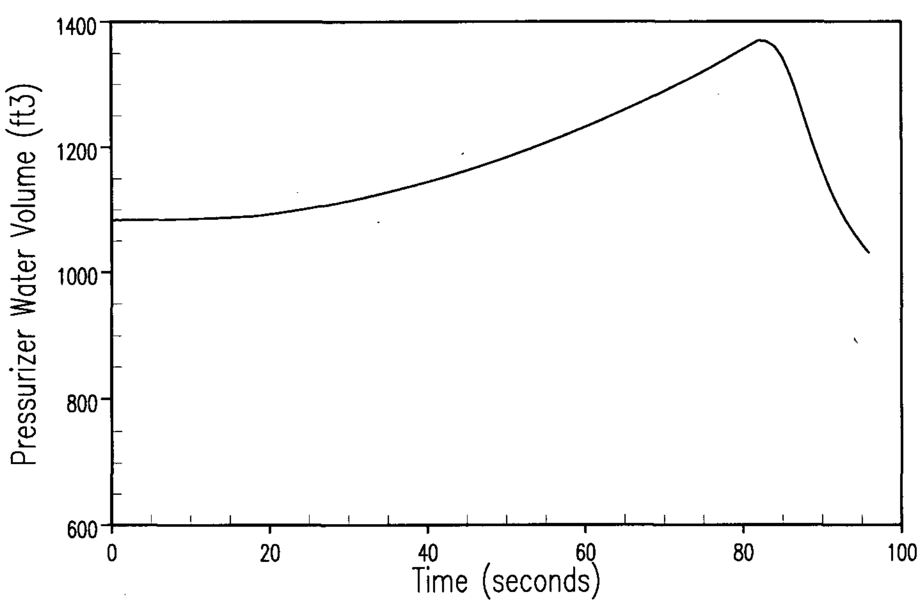
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Nuclear Power and Core Average Heat Flux Transients for an Uncontrolled RCCA Bank Withdrawal of 75 pcm/sec at 100% Power with Minimum Feedback	
	Rev. 12	Figure 15.4-4 Sh. 1 of 3



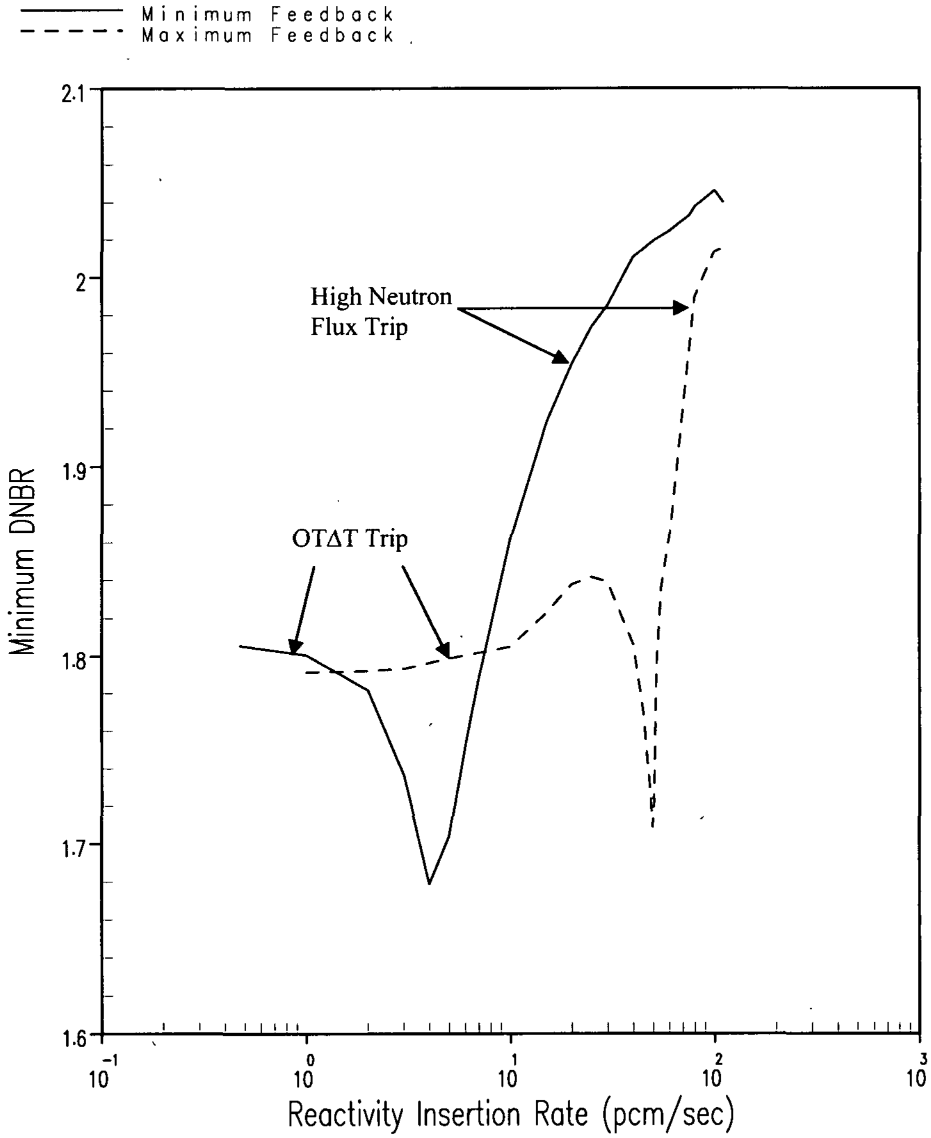


SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Core Average Temperature and DNBR Transients for an Uncontrolled RCCA Bank Withdrawal of 75 pcm/sec at 100% Power with Minimum Feedback	
	Rev. 12	Figure 15.4-4 Sh. 3 of 3

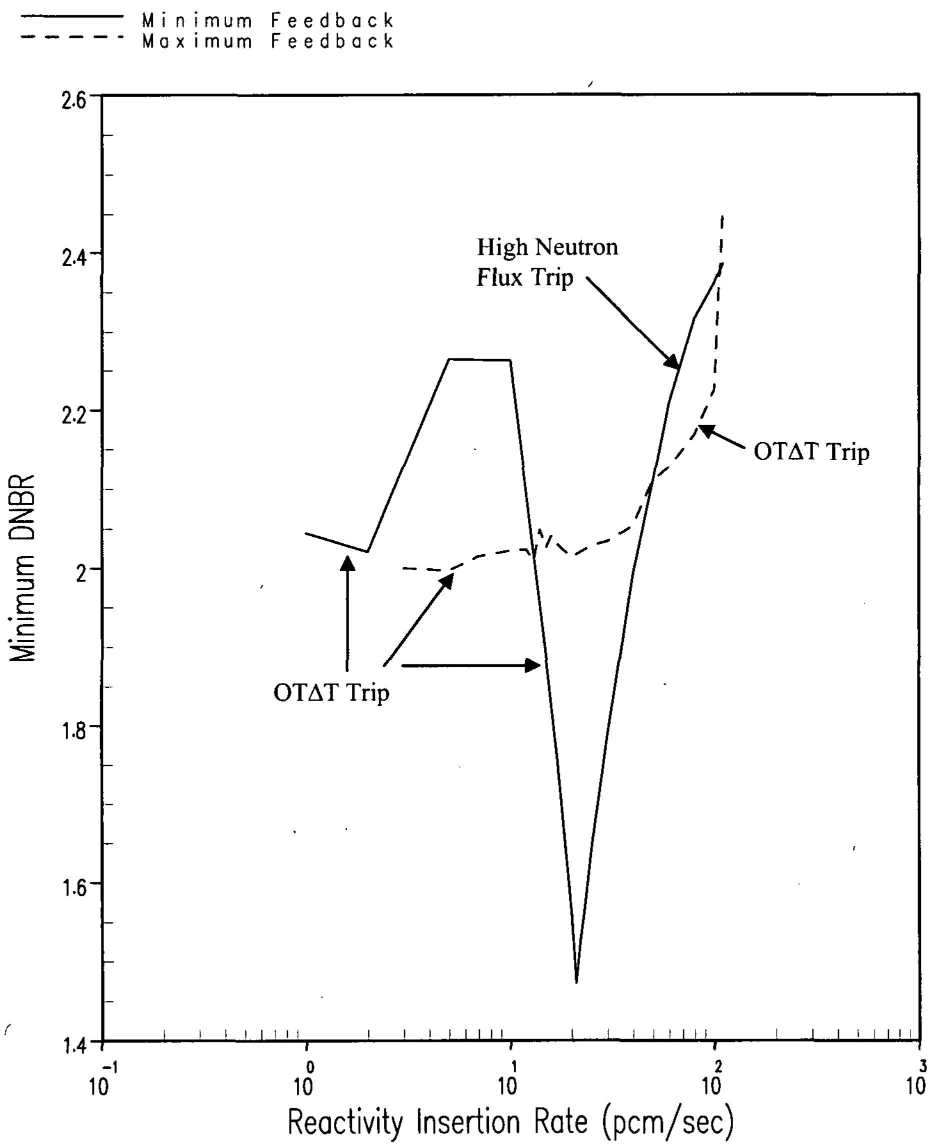




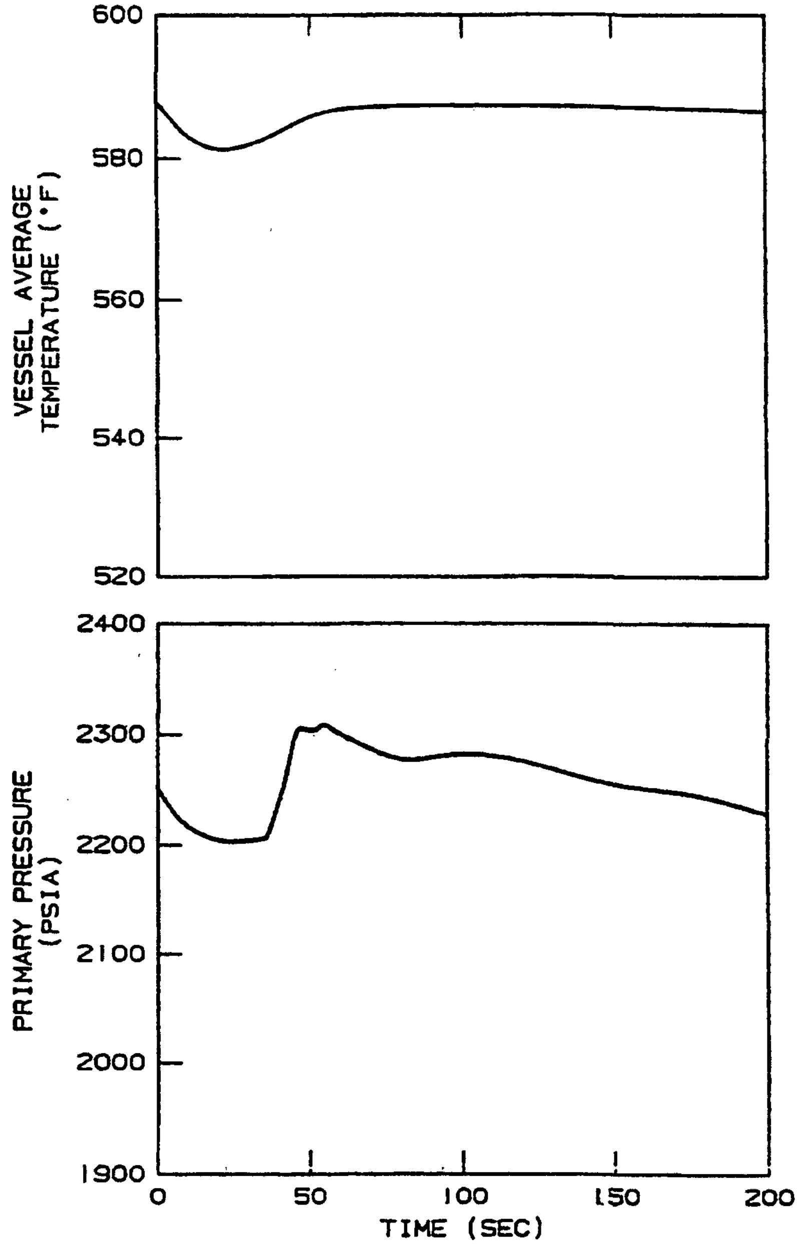
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Pressurizer Pressure and Pressurizer Water Volume Transients for an Uncontrolled RCCA Bank Withdrawal of 1 pcm/sec at 100% Power with Minimum Feedback	
	Rev. 12	Figure 15.4-5 Sh. 2 of 3



SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Minimum DNBR vs. Reactivity Insertion Rate for an Uncontrolled RCCA Bank Withdrawal at 100% Power	
	Rev. 12	Figure 15.4-6

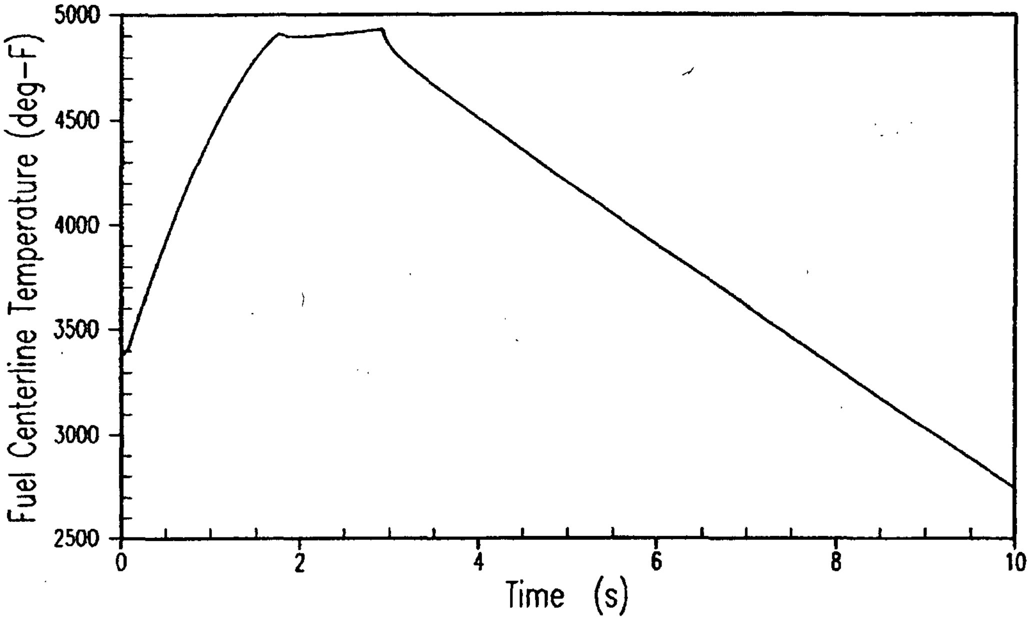


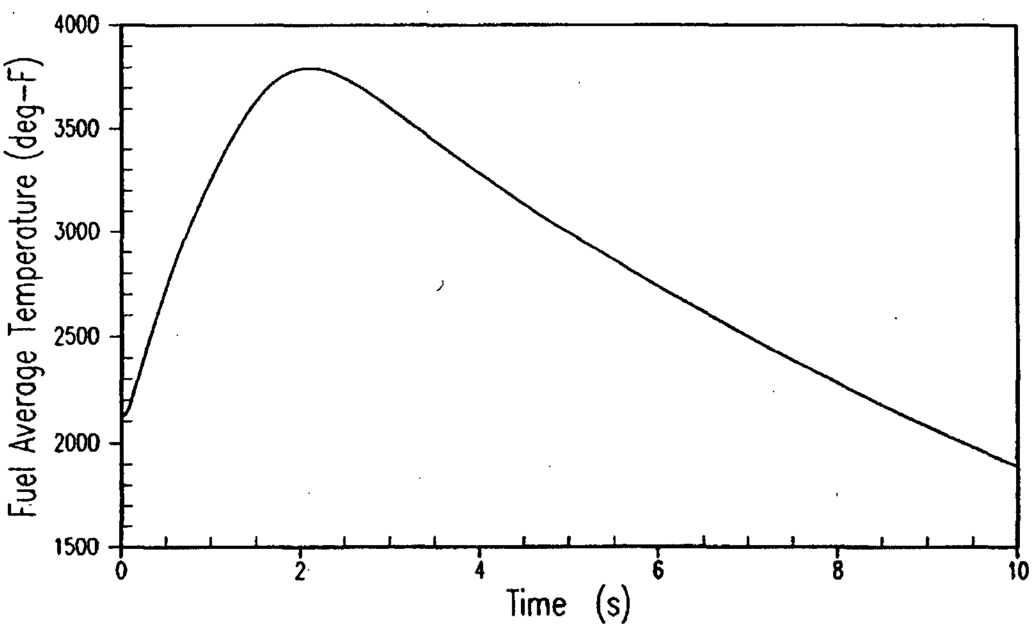
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Minimum DNBR vs. Reactivity Insertion Rate for an Uncontrolled RCCA Bank Withdrawal at 10% Power	
	Rev. 12	Figure 15.4-8



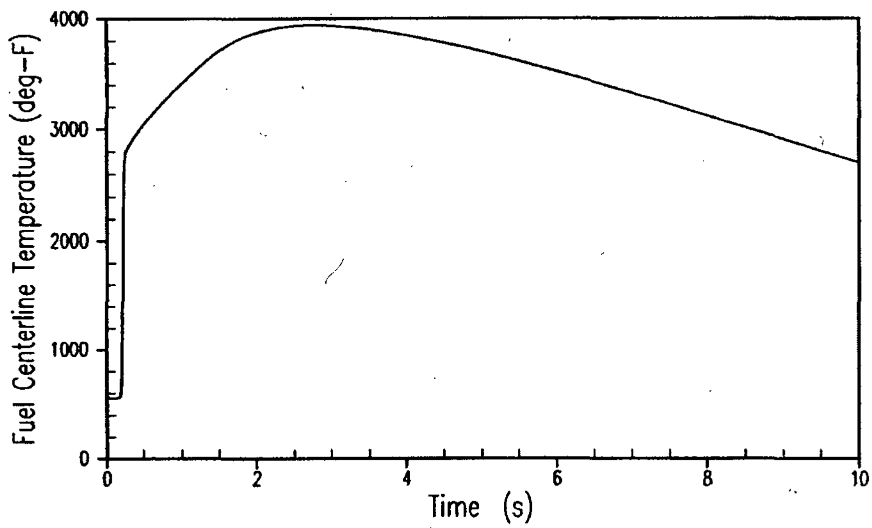
G:\Word\Images_P\Images_P.UFSAR\51162.ds4

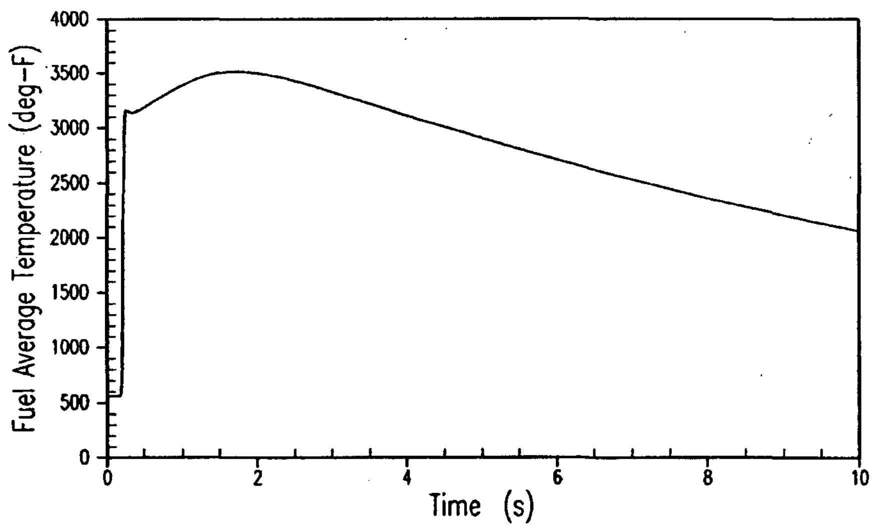
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Average Coolant Temperature and Pressurizer Pressure for a Dropped RCCA	
		Figure 15-4-9 Sh. 2 of 2





SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Fuel Centerline Temperature and Fuel Average Temperature Transients for an Uncontrolled RCCA Ejection at BOL HFP	
		Figure 15.4-10 Sh. 2 of 2





SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Fuel Centerline and Fuel Average Temperature Transients for an Uncontrolled RCCA Ejection at COL HZP	
		Figure 15.4-11 Sh. 2 of 2

Deleted

Deleted

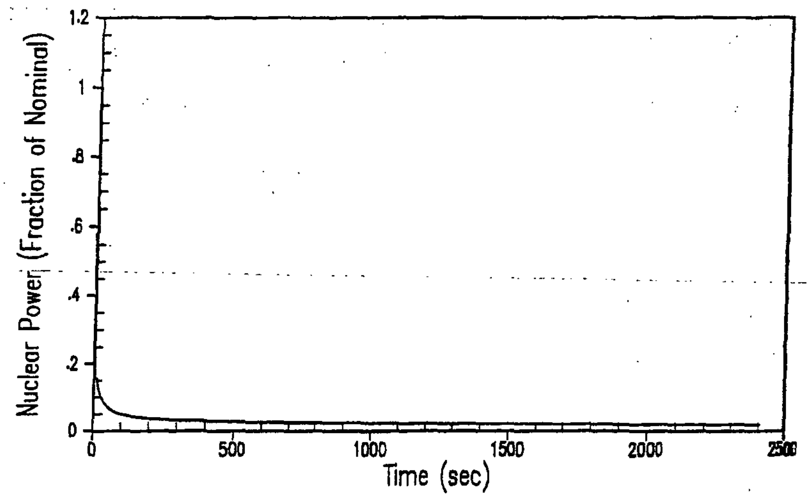
Deleted

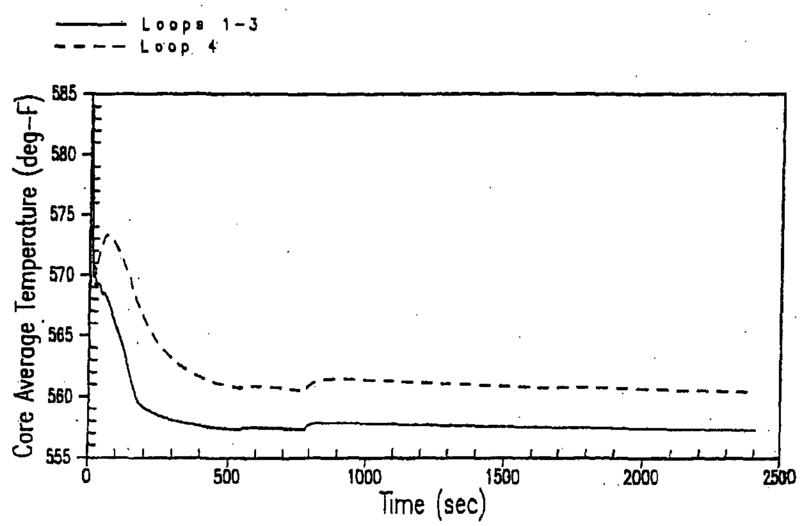
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Deleted	
		Figure 15-4-17

Deleted

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Deleted	
		Figure 15-4-19

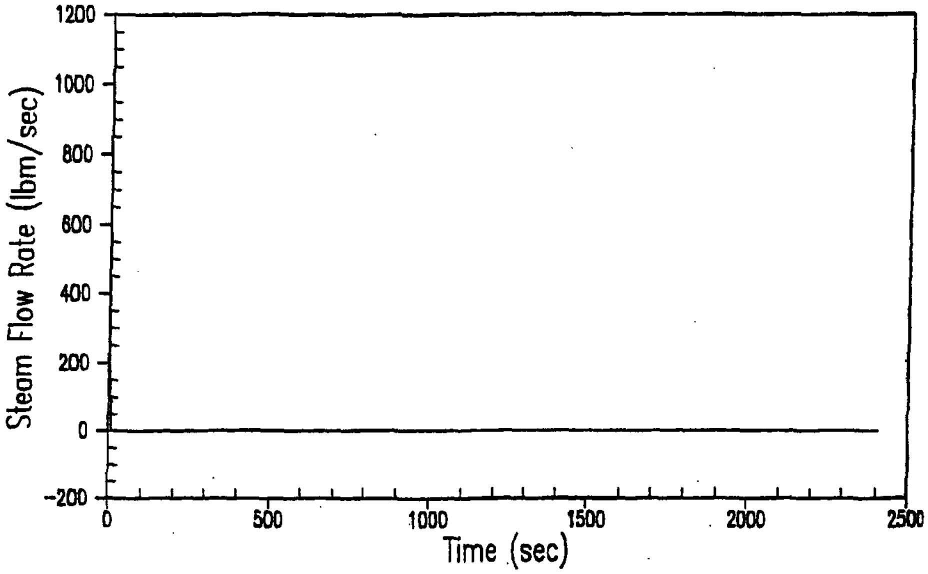
Deleted

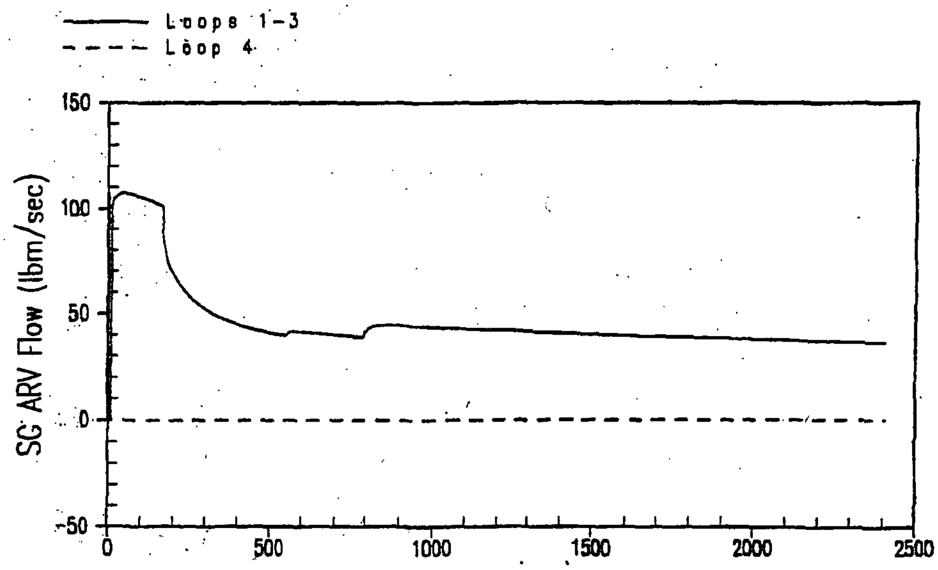




g:\WORK UFSAR_NATIVE\Drawings\15-5-1 Sh1.tif

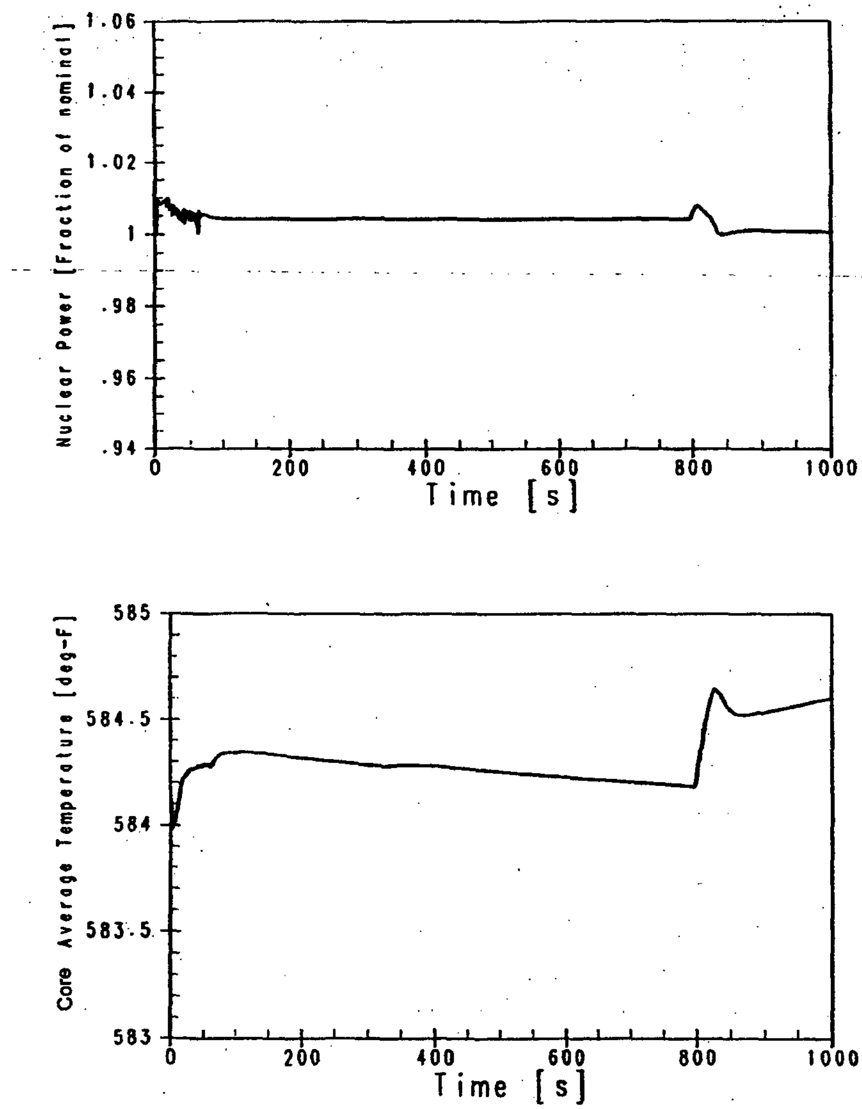
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Nuclear Power and Vessel Average Temperature Transients for an Inadvertent ECCS Actuation During Power Operation	
		Figure 15.5-1 Sh. 1 of 4





g:\WORK UFSAR_NATIVE\Drawings\15-5-1 Sh3.tif

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Main Steam Flow and Steam Generator Atmospheric Relief Valve Flow Transients for an Inadvertent ECCS Actuation During Power Operation	
		Figure 15.5-1 Sh. 3 of 4



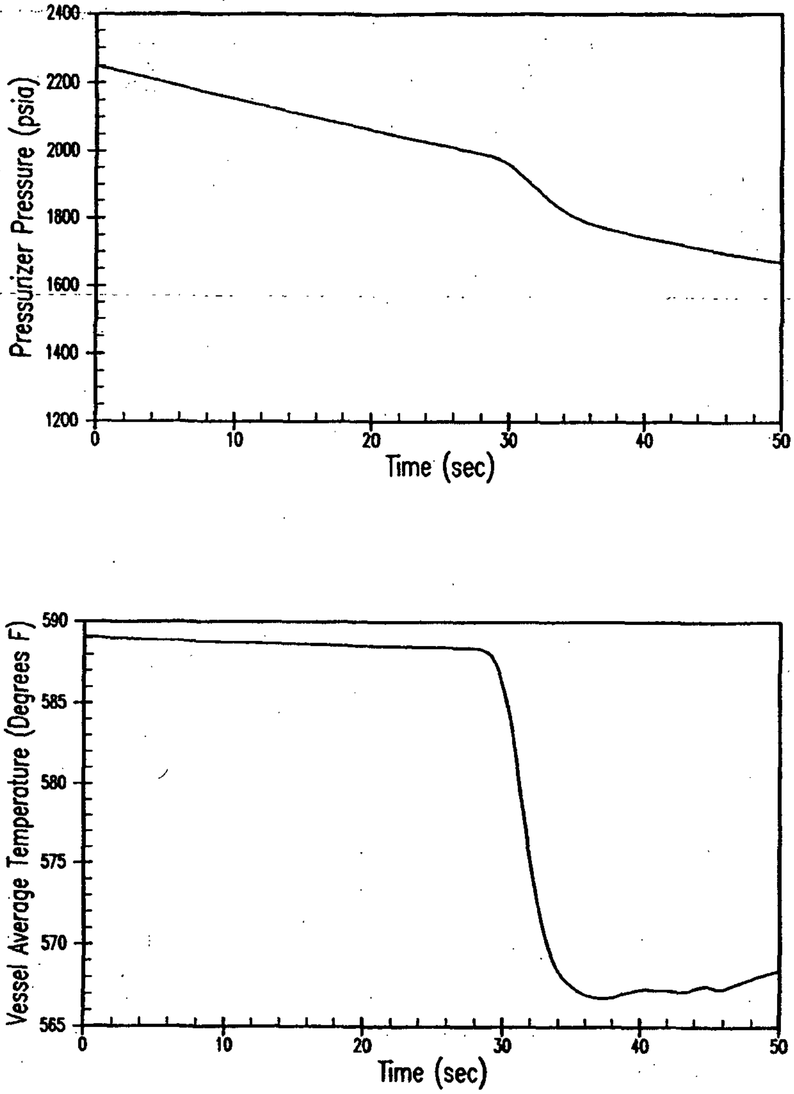
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Nuclear Power and Core Average Temperature Transients for a CVCS Malfunction	
		Figure 15.5-2 Sh. 1 of 2

Deleted

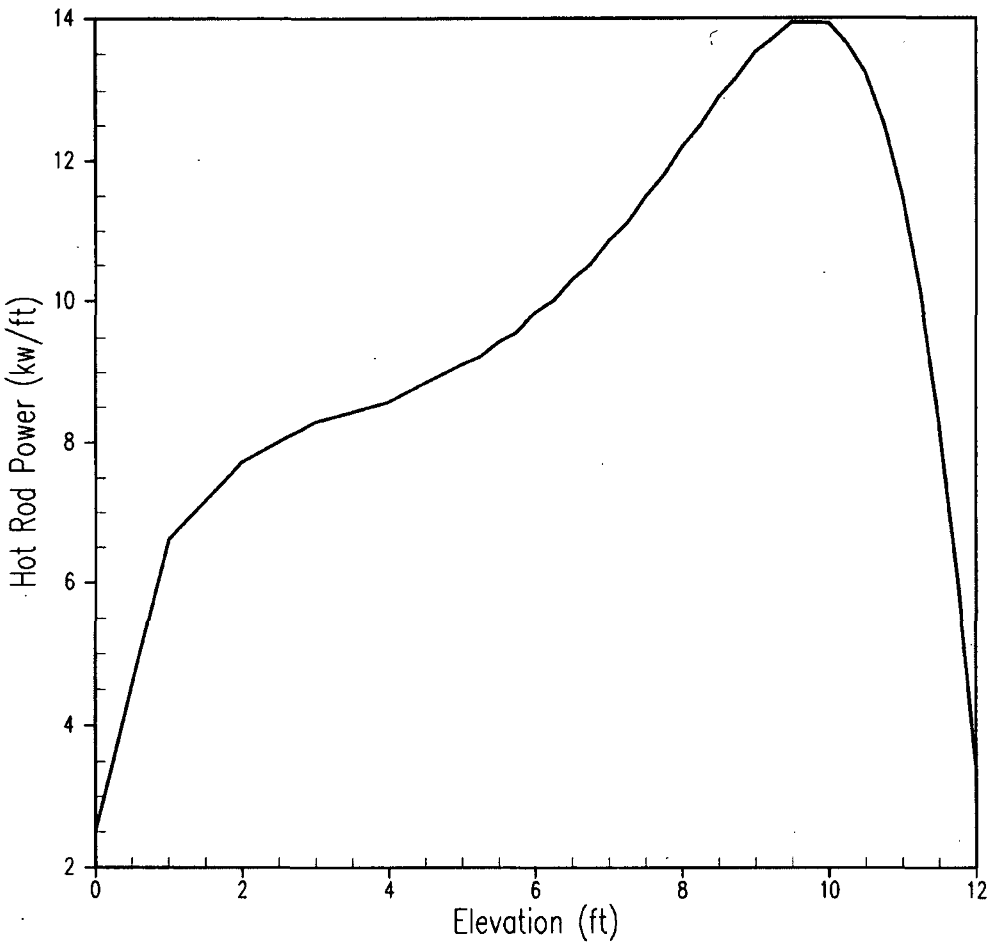
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Deleted	
		Figure 15.5-3

Deleted

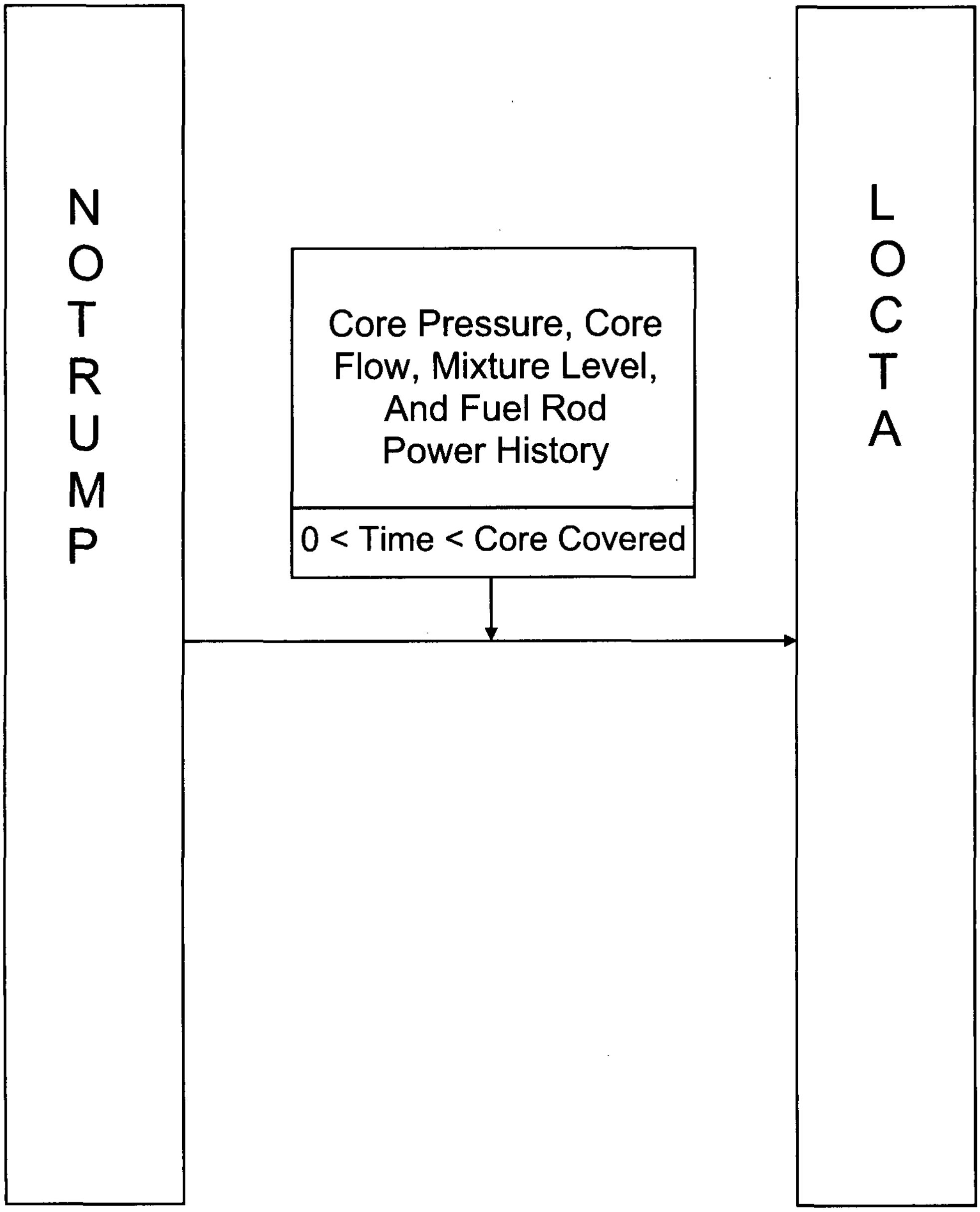
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	DELETED	
		Figure 15-5-5



SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Pressurizer Pressure and Vessel Average Temperature Transients for an Inadvertent Opening of a Pressurizer Safety Valve
	Figure 15.6-2

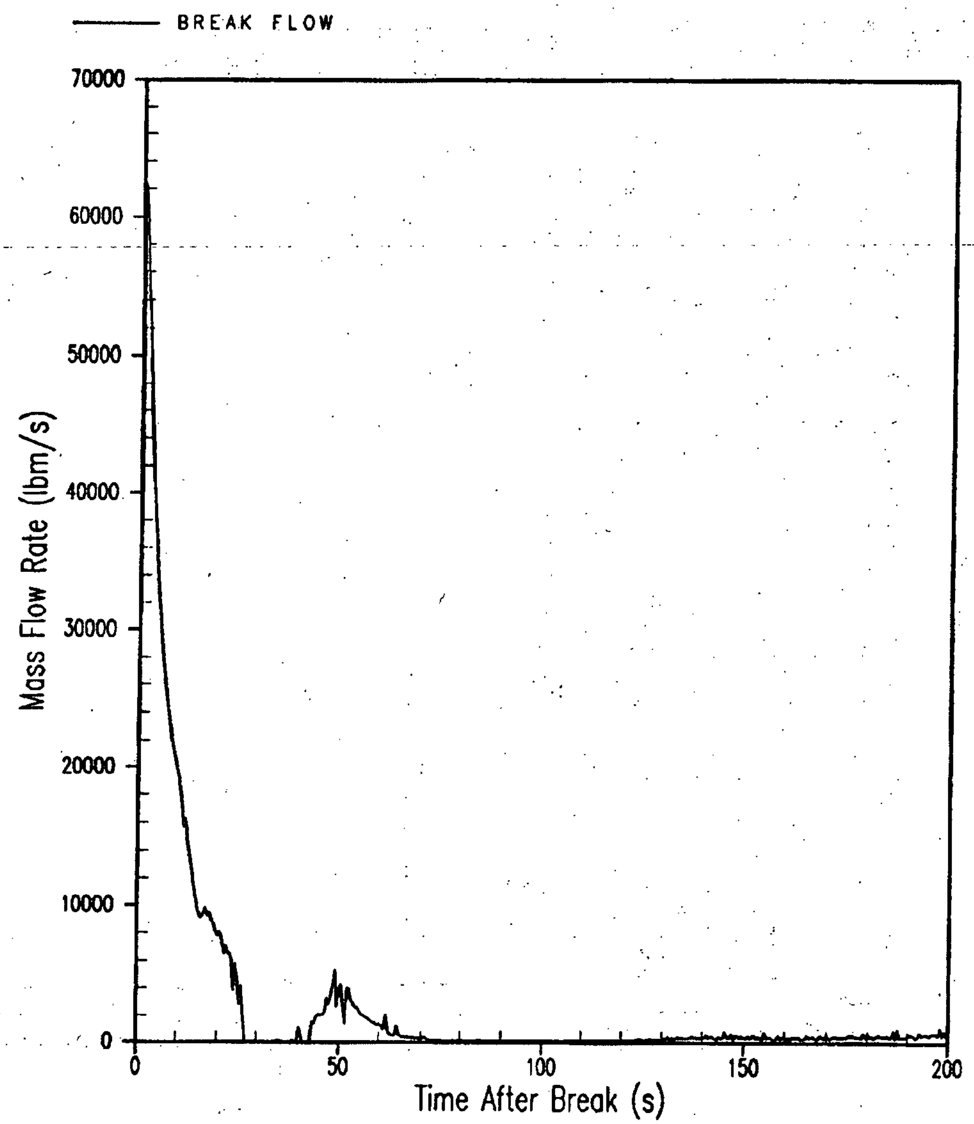


SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Small Break LOCA Hot Rod Power Shape	
		Figure 15.6-4

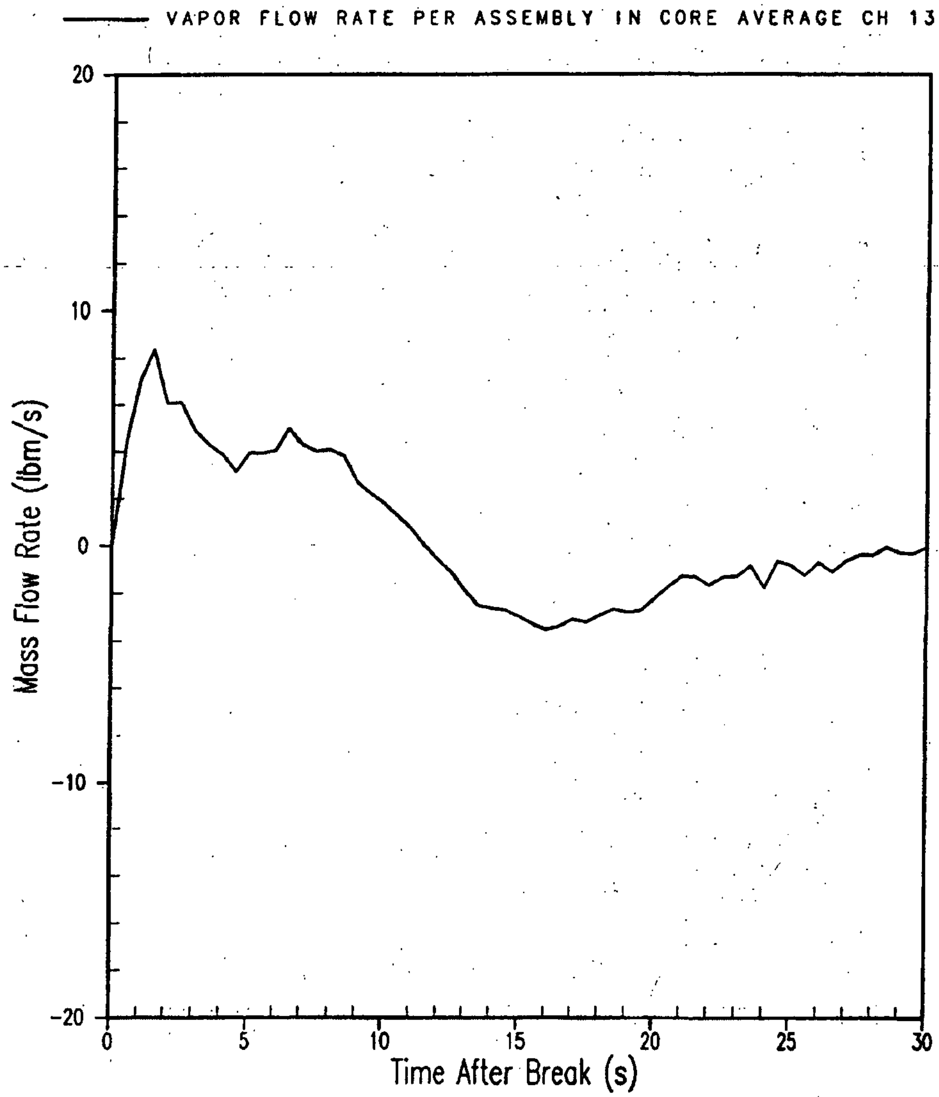


G:\Word\Images_P\UFSAR\1566.ds4

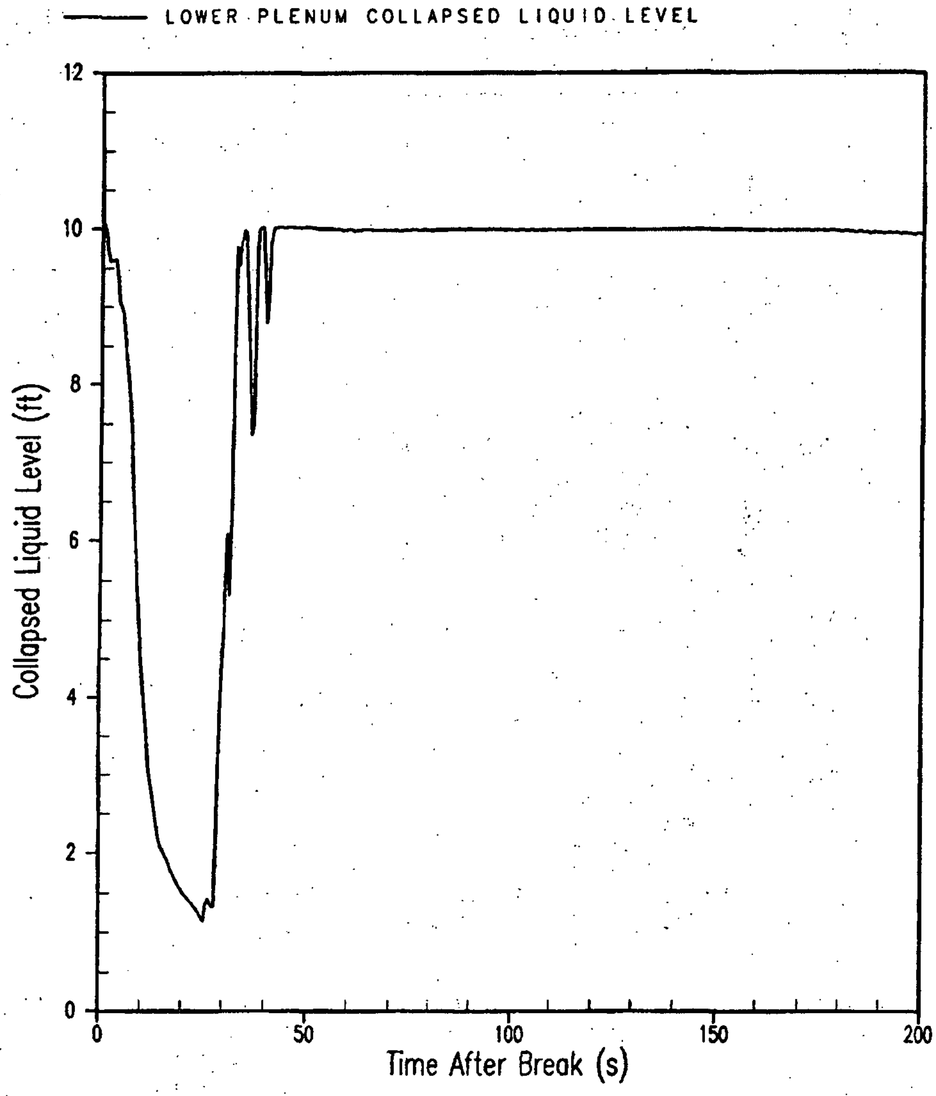
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Small Break LOCA Code Interface	
		Figure 15-6-6



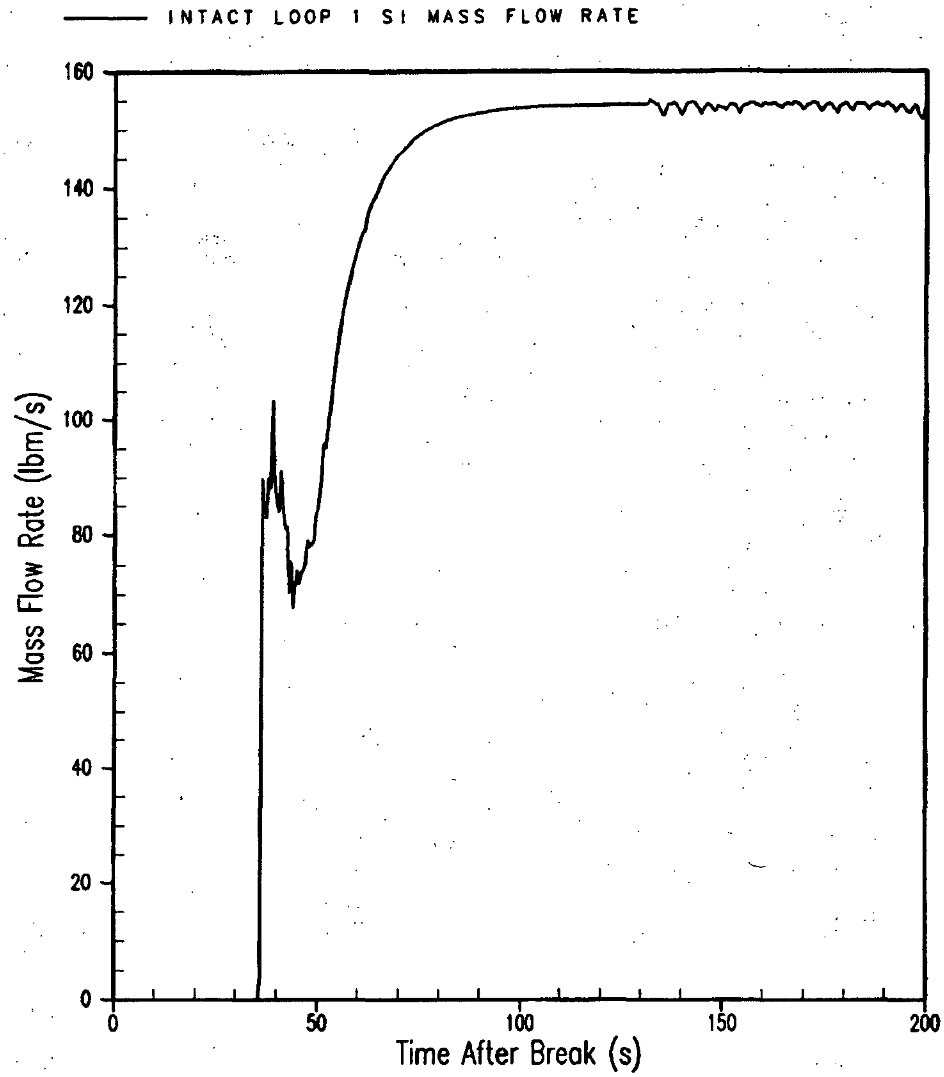
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Break Flow for Reference Split Transient – BE LBLOCA	
		Figure 15.6-8



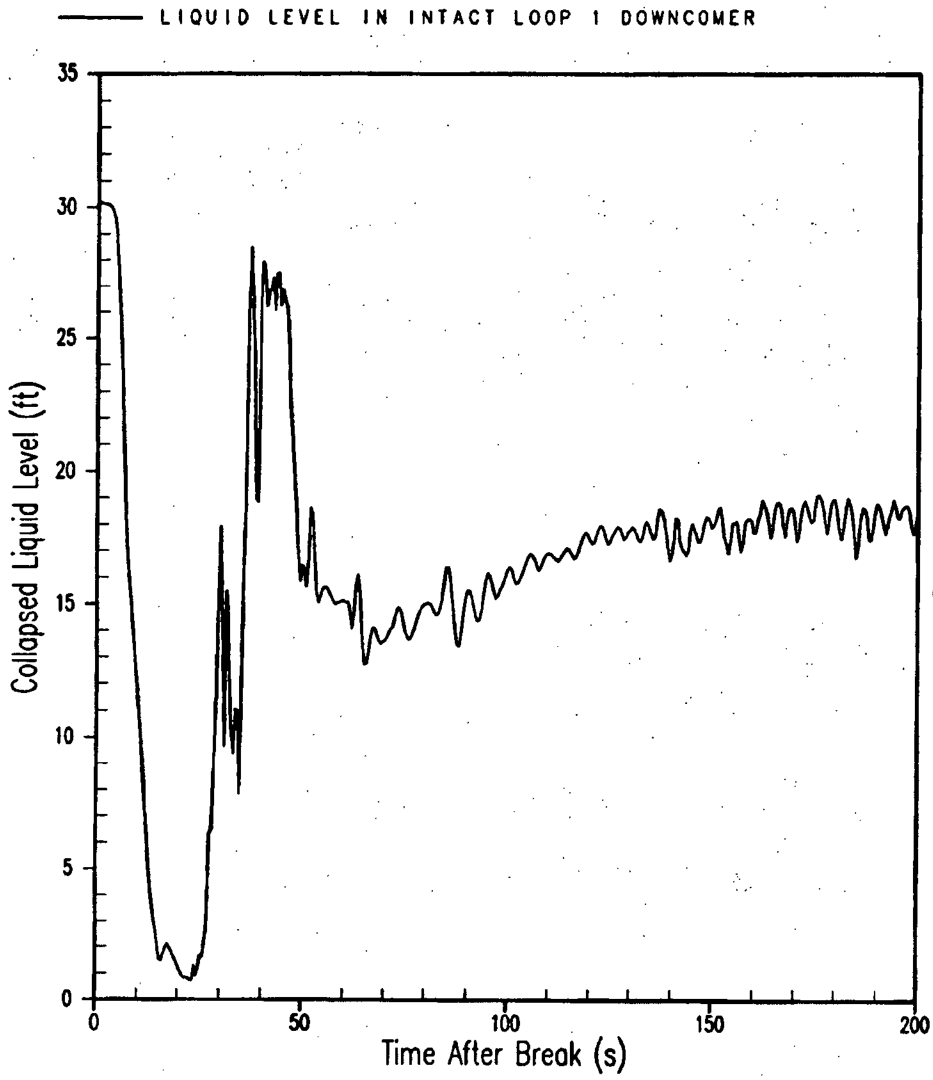
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Vapor Flow Rate at Midcore in Channel 13 During Blowdown for Reference Split Transient – BE LBLOCA	
		Figure 15.6-10



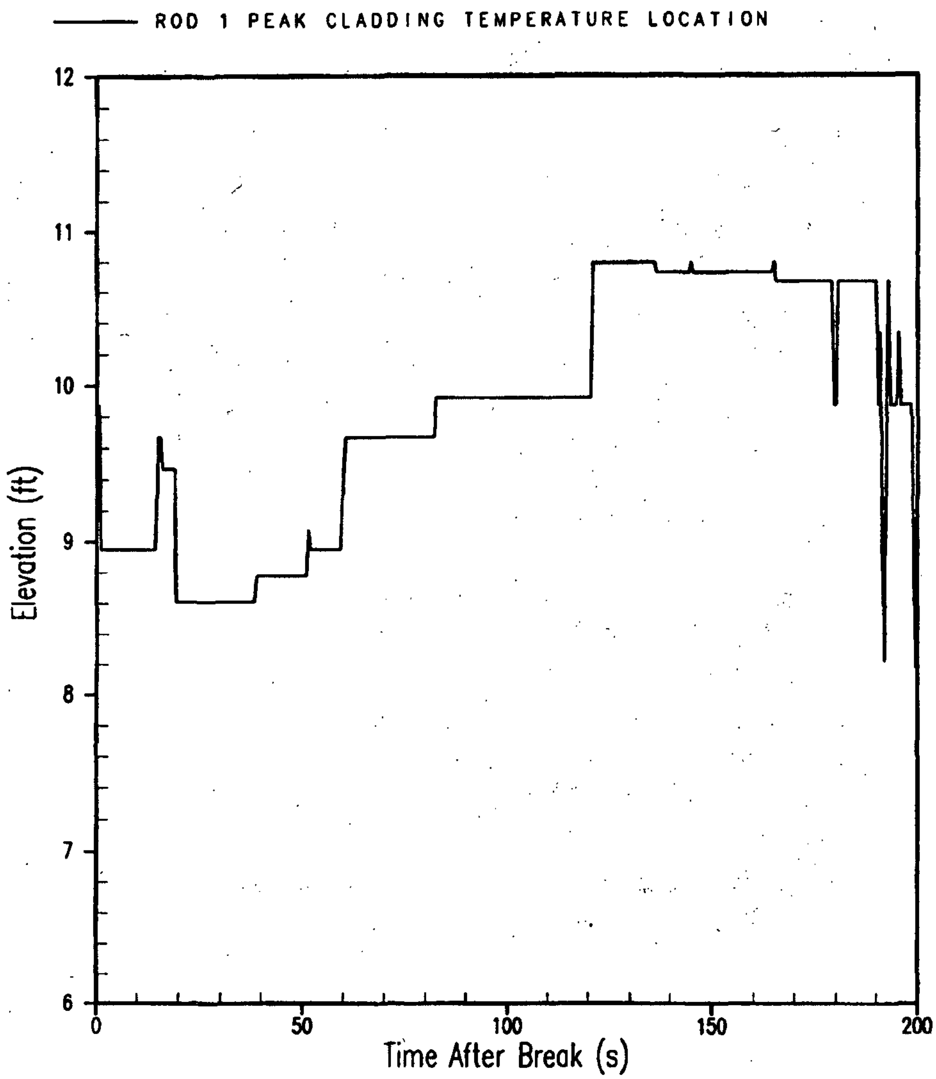
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Collapsed Liquid Level in Lower Plenum for Reference Split Transient – BE LBLOCA	
		Figure 15.6-12



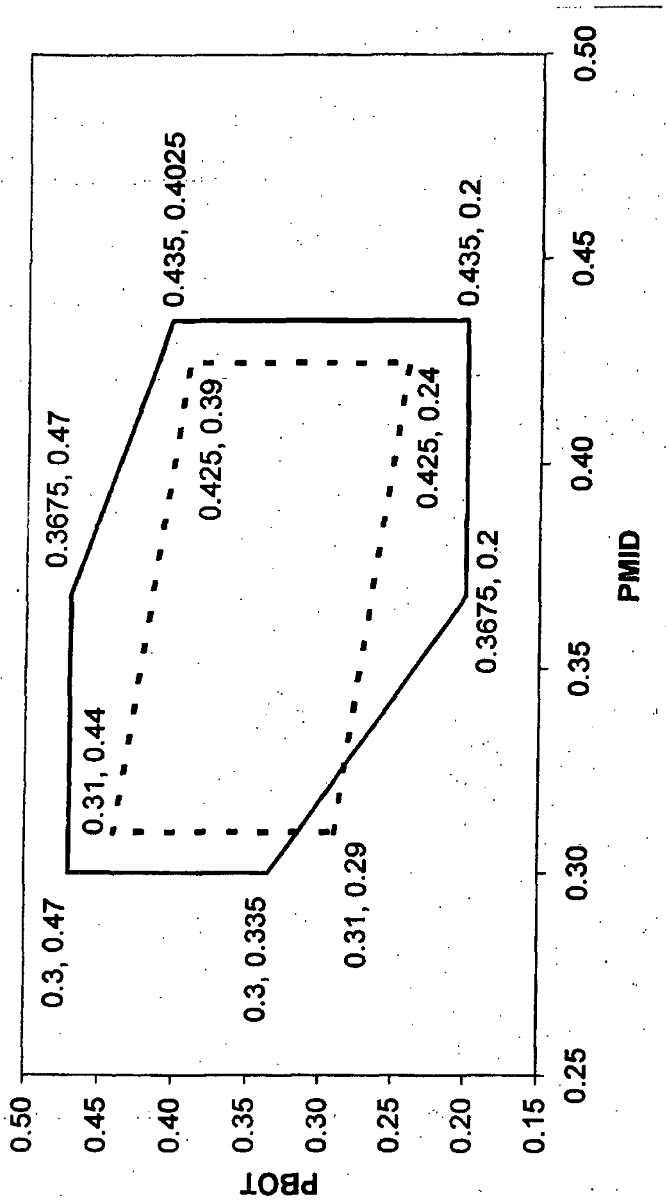
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	SI Mass Flow Rate for Reference Split Transient – BE LBLOCA	
		Figure 15.6-14



SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Collapsed Liquid Level in Downcomer for Reference Split Transient BE LBLOCA	
		Figure 15.6-16



SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Peak Cladding Temperature Location for Reference Split Transient – BE LBLOCA	
		Figure 15.6-18



SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	PBOT/PMID Sampling Limits (Plant Operating range indicated by dashed line; WC/T Range indicated by solid line) Figure 15.6-20

Deleted

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Deleted	
		Figure 15.6-22

Deleted

Deleted

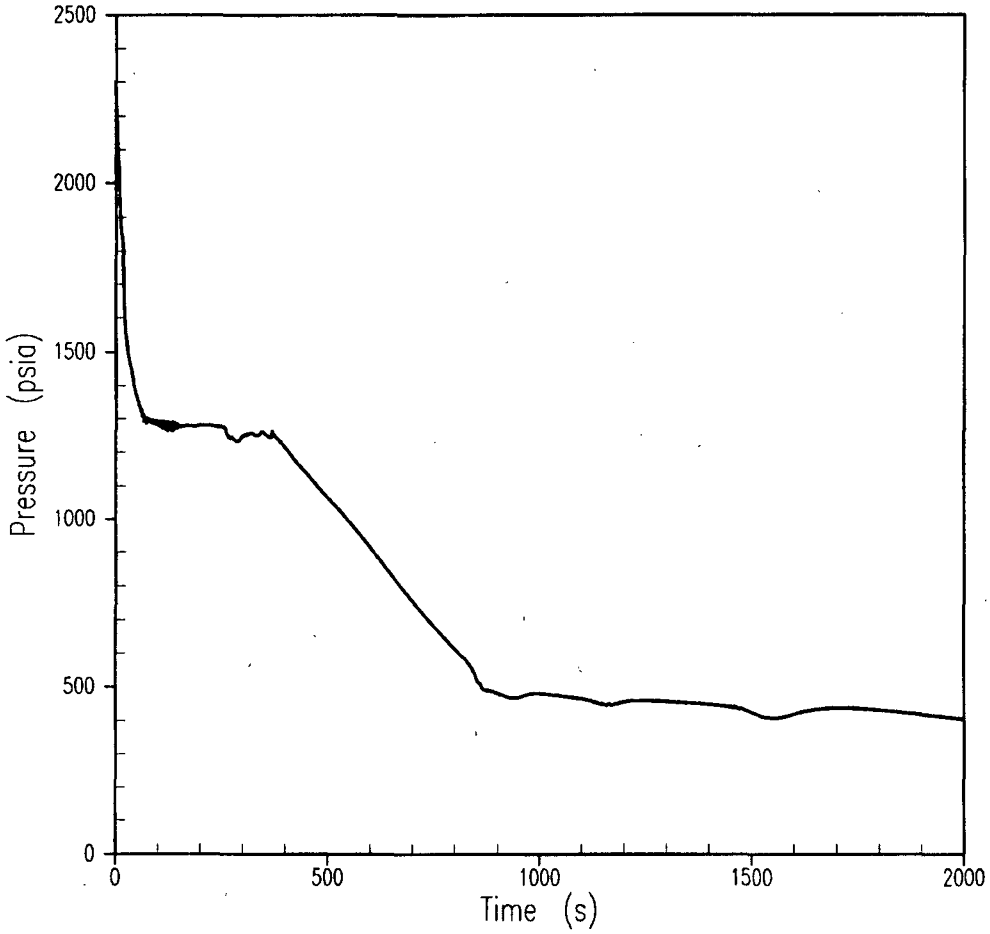
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Deleted	
		Figure 15.6-26

Deleted

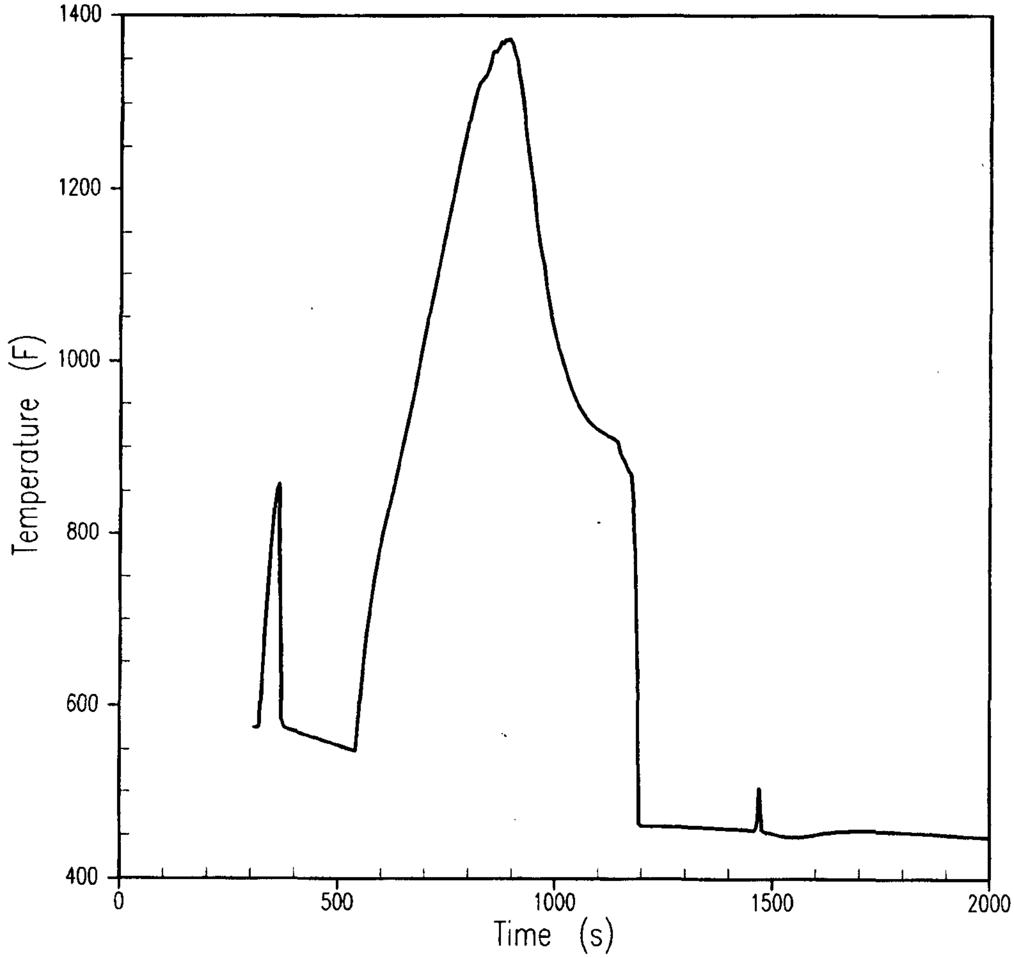
Deleted

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Deleted	
		Figure 15.6-30

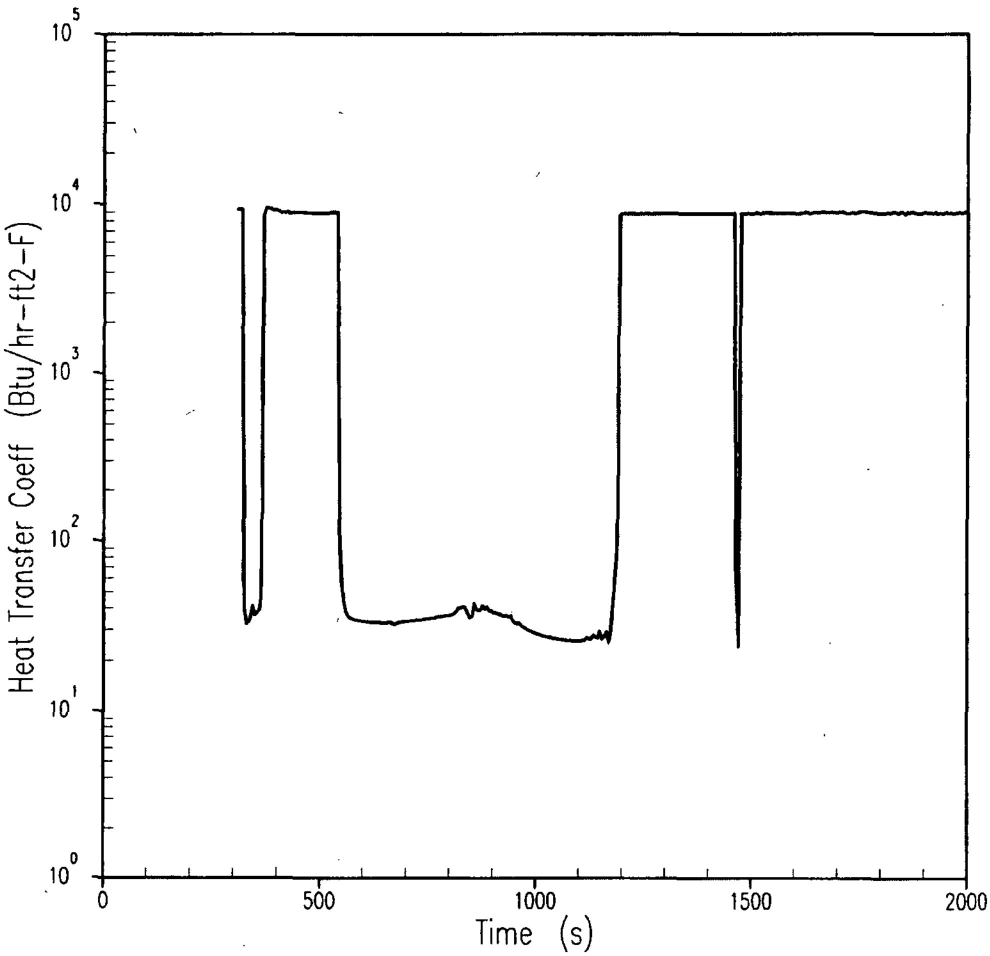
Deleted



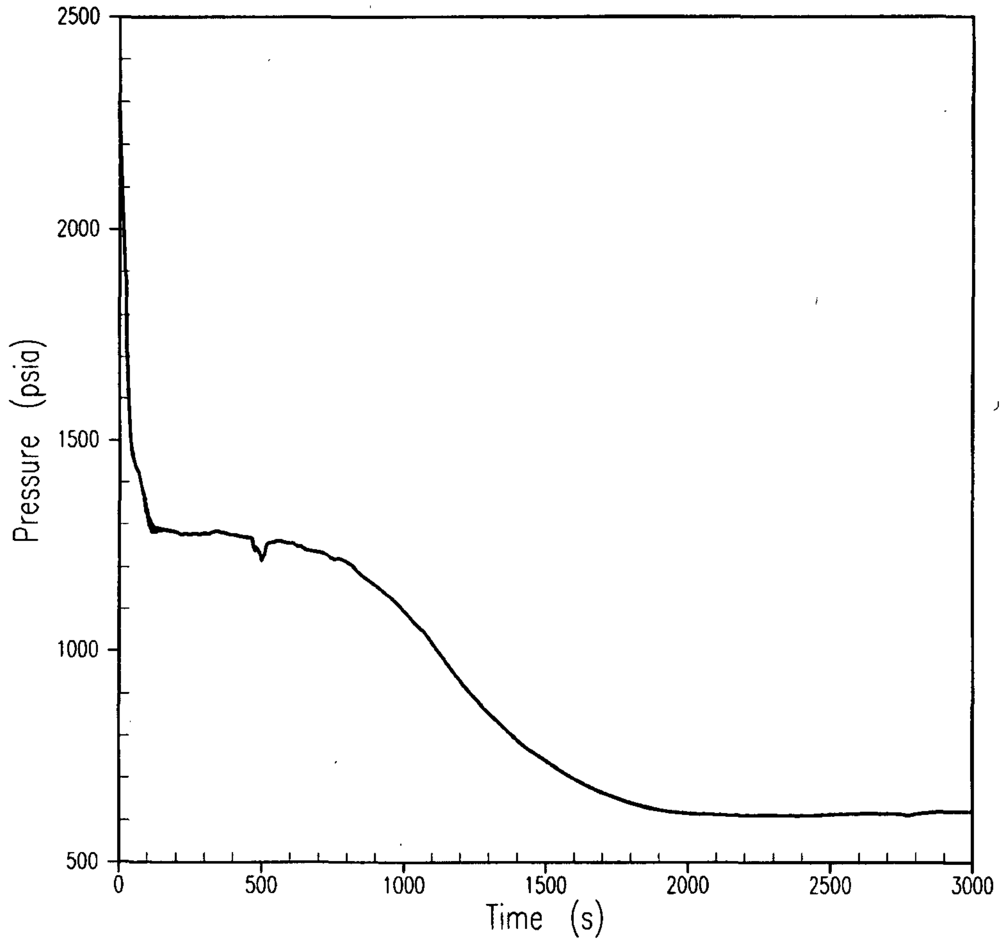
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	RCS Pressure (4 Inch Break)	
		Figure 15.6-34



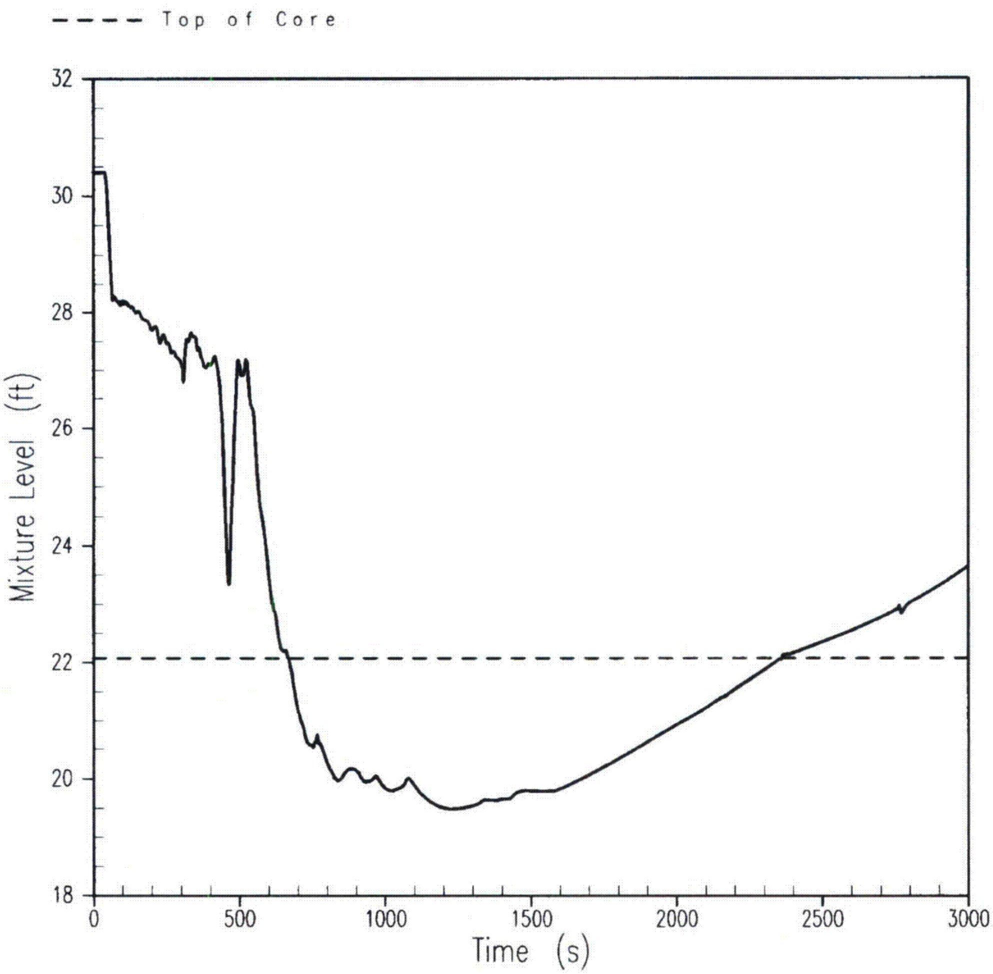
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Peak Clad Temperature (4 Inch Break)	
		Figure 15.6-36



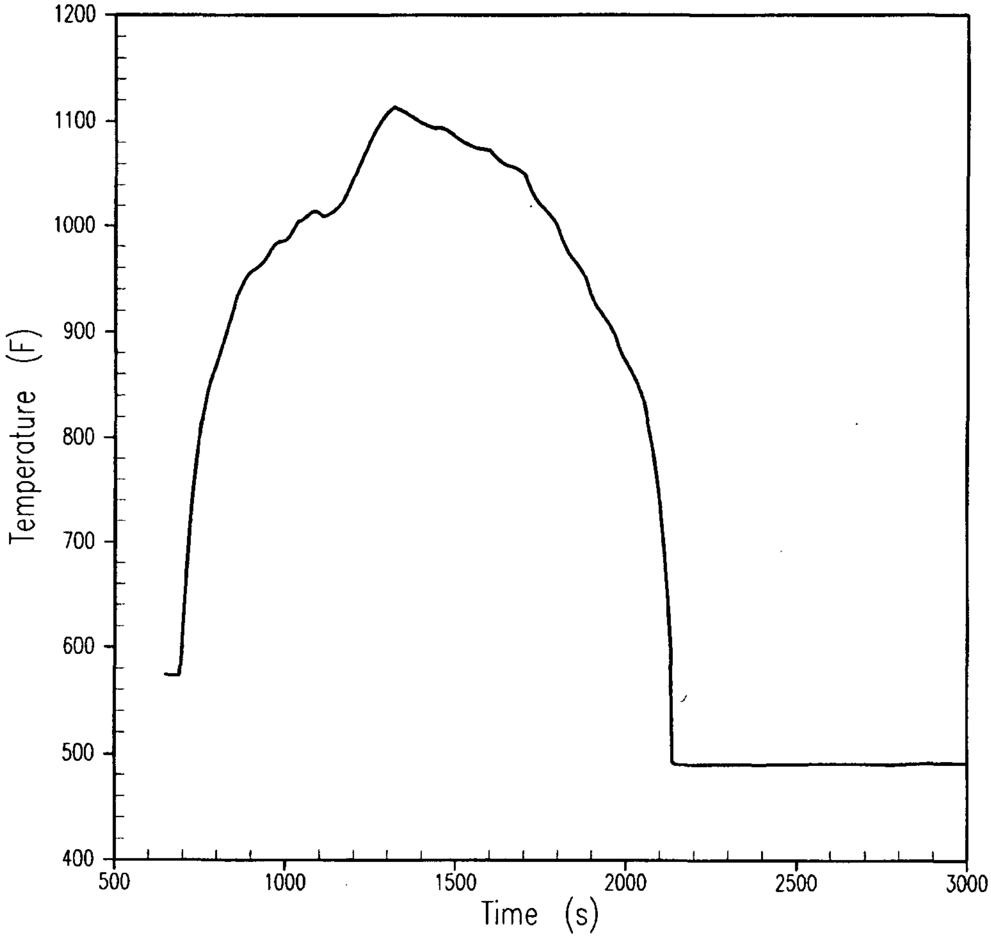
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Hot Rod Heat Transfer Coefficient (4 Inch Break)	
		Figure 15.6-38



SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	RCS Pressure (3 Inch Break)	
		Figure 15.6-40



---- Top of Core
32
30
28
26
24
22
20
18
Mixture Level (ft)
0
500
1000
1500
2000
2500
3000
Time (s)

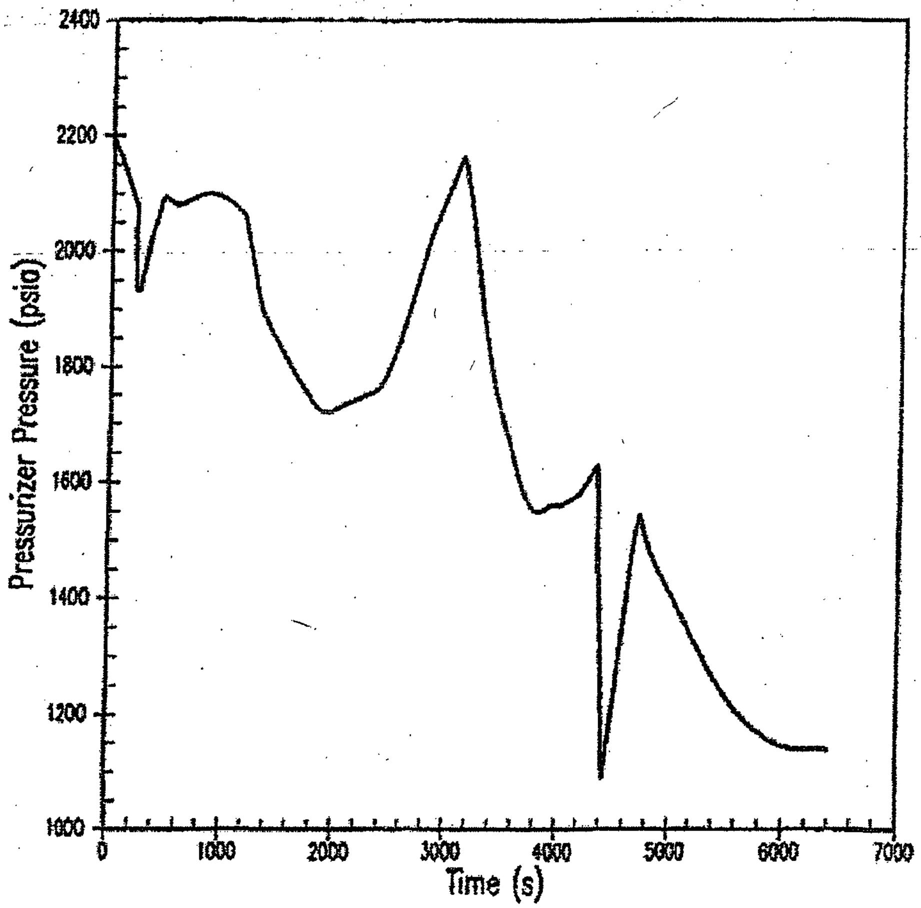


SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Peak Clad Temperature (3 Inch Break)	
		Figure 15.6-44

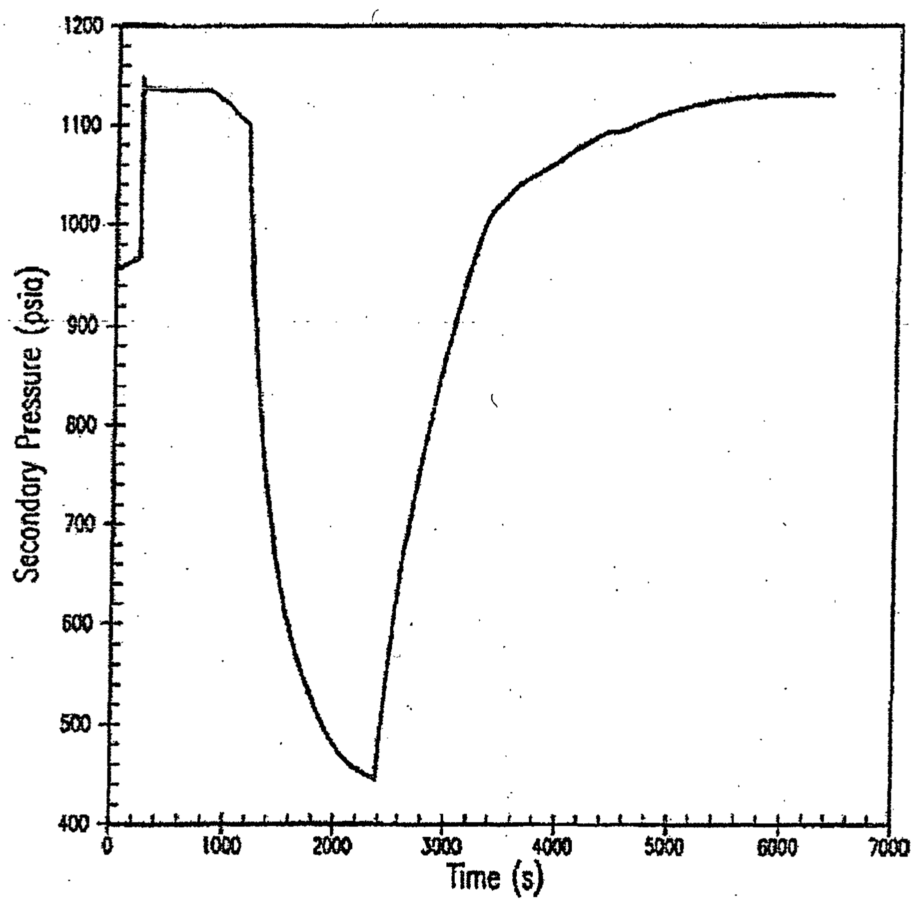
Deleted

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Deleted	
		Figure 15.6-46

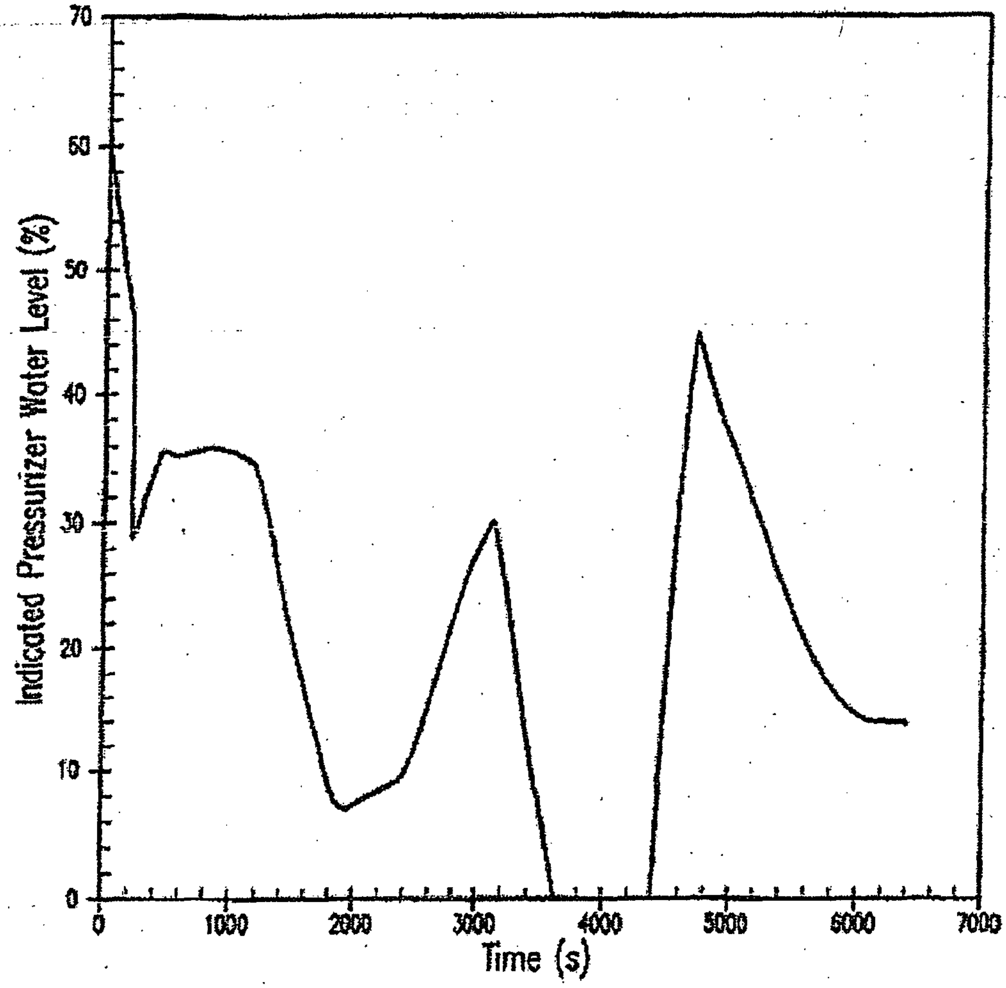
Deleted



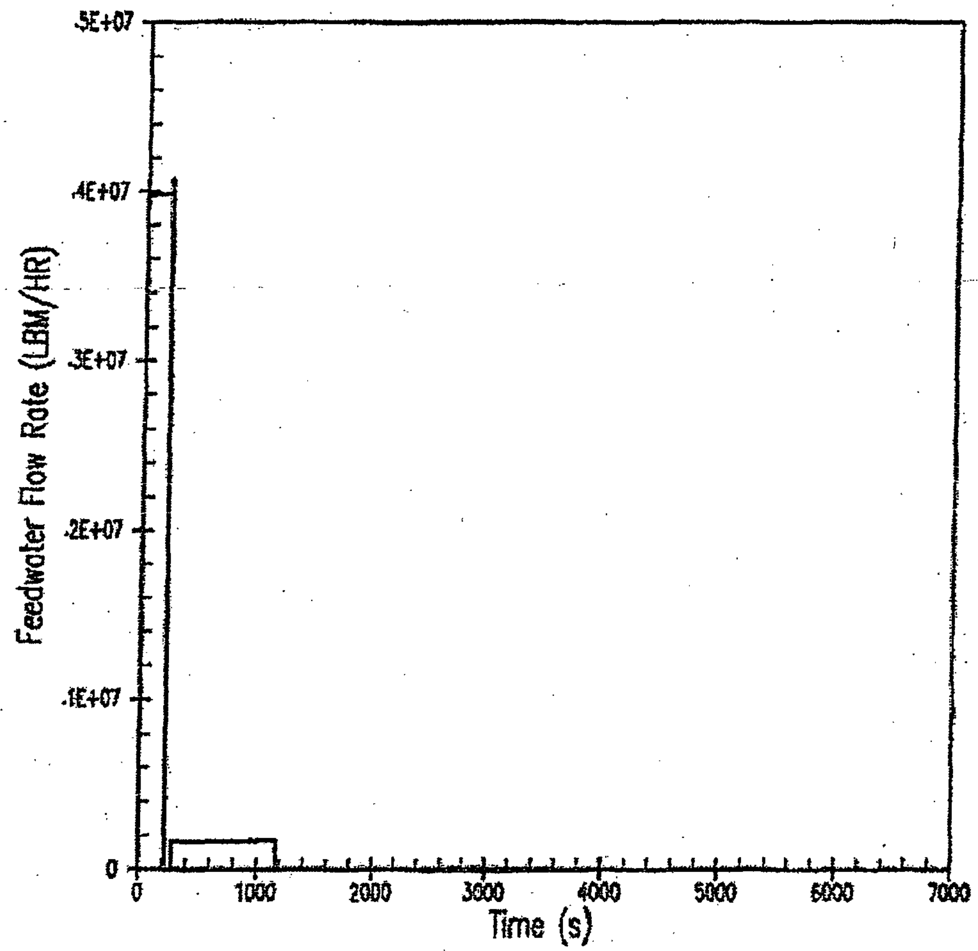
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Pressurizer Pressure as the Result of Steam Generator Tube Rupture	
	Rev. 12	Figure 15.6-50



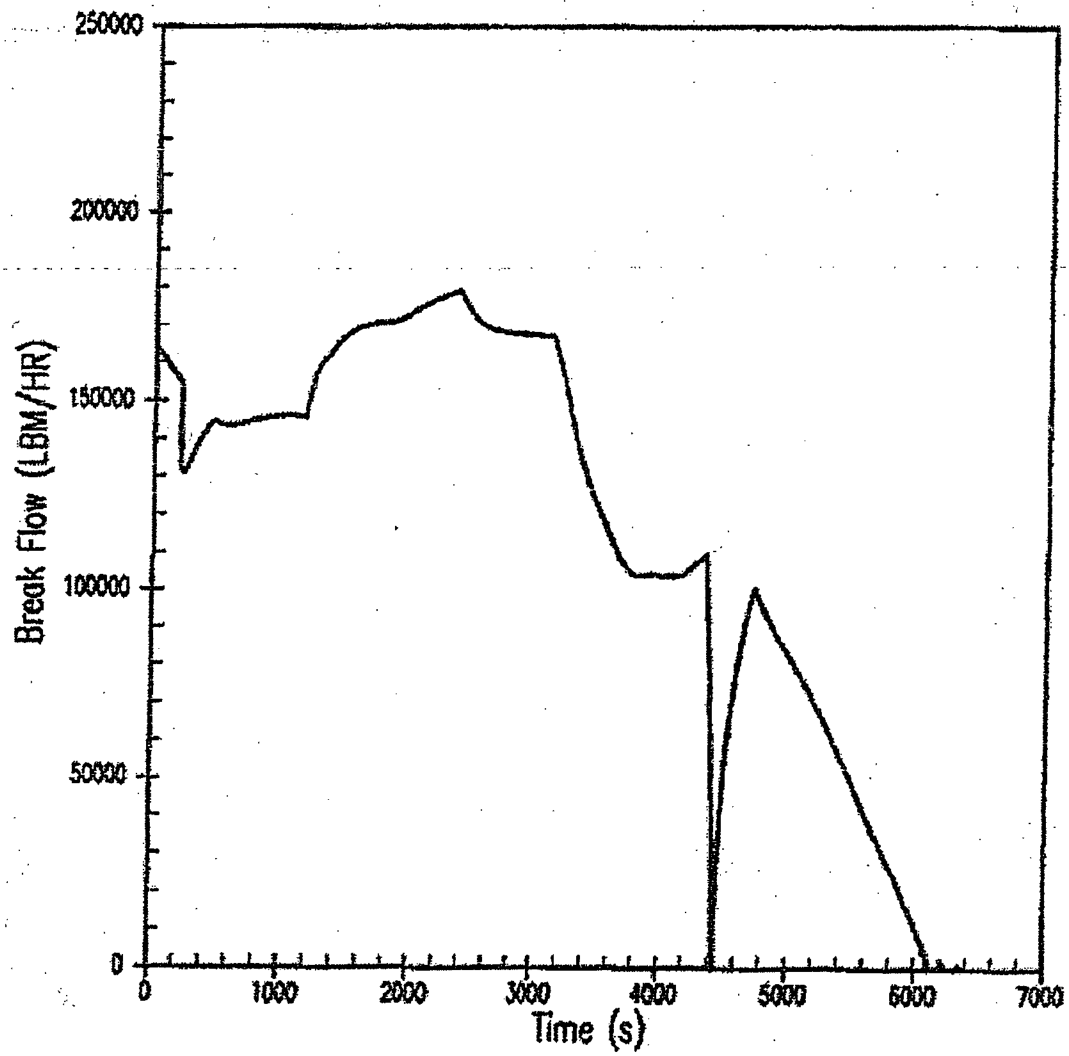
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Faulted Steam Generator Pressure as the Result of Steam Generator Tube Rupture	
	Rev. 12	Figure 15.6-52



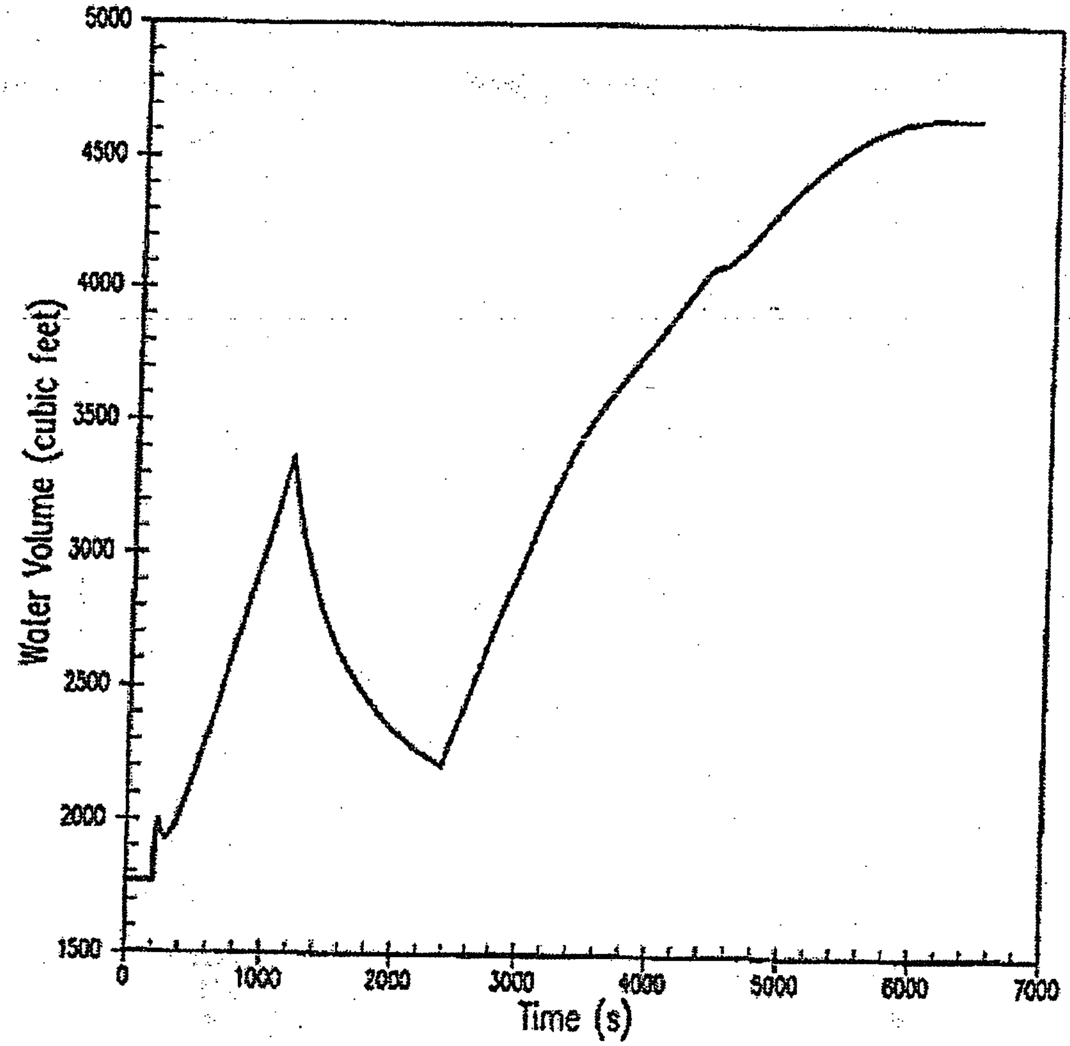
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Pressurizer Water Level as the Result of Steam Generator Tube Rupture	
	Rev. 12	Figure 15.6-54



SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Feedwater Flow Rate to Ruptured Steam Generator as the Result of Steam Generator Tube Rupture	
	Rev. 12	Figure 15.6-56



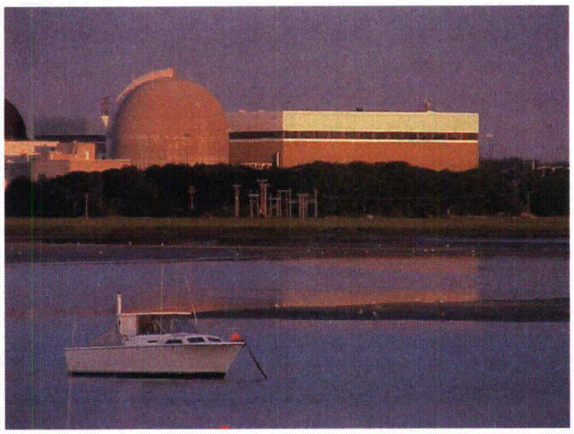
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Faulted Steam Generator Break Flow Rate to the Result of Steam Generator Tube Rupture	
	Rev. 12	Figure 15.6-58



SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Ruptured Steam Generator Water Volume as the Result of Steam Generator Tube Rupture	
	Rev. 12	Figure 15.6-60

SEABROOK STATION
UPDATED FINAL SAFETY ANALYSIS REPORT

CHAPTER 16
TECHNICAL SPECIFICATIONS

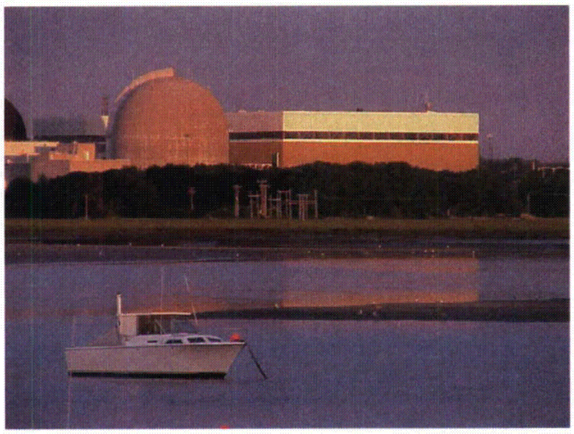


16.2 FINAL TECHNICAL SPECIFICATIONS

Refer to Technical Specifications, Seabrook Station, Unit I, Docket No. 50-443, Appendix A to License No. NPF-86, March 1990.

SEABROOK STATION
UPDATED FINAL SAFETY ANALYSIS REPORT

CHAPTER 17
QUALITY ASSURANCE



The responsibilities of YAEC and other key project management personnel are outlined below:

The Director of Quality Assurance who reported to the YAEC President was responsible for establishing policies under which the Yankee Quality Assurance organization works, and under which contractors comply. He approved the Seabrook Station Quality Assurance Manual which governed all YAEC program activities and received copies of correspondence and reports generated by the Quality Assurance Department (QAD). He evaluated and reported to the President on the effectiveness of the Quality Assurance Program. He reported on a quarterly basis to the NHY management to keep them advised of the program status. He coordinated the activities and program direction of quality assurance during design, construction and certain phases of startup operation to maintain consistency of the program and continuity of the effort.

The Construction Quality Assurance Manager (QAM) who reported to the Director of Quality Assurance, was responsible for the direction and supervision of work performed by the Construction Quality Assurance Group staff, at both the corporate office and at the plant site, and by consultants hired to supplement this staff. Offsite personnel (Home Office QA Engineers) performed staff functions, i.e., developed QA programs and procedures, reviewed technical and QA documentation submittals, provided training and indoctrination and performed audit and/or surveillance functions internally as well as over contractors, constructors, subcontractors and suppliers. Onsite personnel performed QA line functions, i.e., planned and developed verification procedures and controls, performed surveillance activities over constructors and subcontractors and reviewed contractor and subcontractor implementing procedures. Qualification requirements for the position responsible for establishing and implementing the Seabrook Station QA Program were:

1. Graduate of an accredited college or university, with a technical degree.

2. Ten years minimum experience consisting of:

   (a) Significant experience in a utility, nuclear, heavy construction or heavy equipment industry.

   (b) Experience in development and implementation of quality assurance programs, plans and procedures.

3. Familiarity with 10 CFR 50 and applicable codes and standards.

1. SNT-TC-IA Levels II and III

2. Quality Assurance for mechanical equipment

3. Quality Assurance for electrical equipment

4. Quality Assurance for instrumentation and control equipment

5. Quality Assurance for construction activities.

Personnel qualifications to review design and procurement documents and QA programs and to perform audits were reviewed annually at which time a determination was made for the need for further training. Responsibilities and duties were assigned to personnel having qualifications required for the assignments. The QAM, and personnel reporting to him, had the authority to order that work be stopped on any operation they found being performed contrary to approved procedures, specifications, instructions, or drawings.

<u>The NHY Construction Director</u> reported to the Senior Vice President of NHY and was responsible for managing all field personnel, thus ensuring that all construction-related activities are properly completed. The Construction Director was charged with the responsibility for completing the field construction activities in accordance with corporate guidelines, project planner and scheduler, project objectives, engineering drawings, specification, instructions, and procedures.

The Construction Director was responsible for performing the full range of management functions, including organizing, staffing, directing, leading and controlling the work of the assigned field personnel and contractors, as well as serving as the focal point for all groups involved with the construction-related work at the jobsite, both within and outside New Hampshire Yankee.

<u>The Director of Engineering and Licensing</u> reported to the Senior Vice President (NHY) and was responsible for providing direction and selected review of all project engineering, design, and NRC licensing work performed; for ensuring that the project engineering organization receives consistent direction and guidance, ensuring consistent and acceptable quality throughout the engineering organization; and for evaluating the impact of regulatory changes to the project.

The contractors were responsible for the review and approval of their supplier and subcontractor quality-related documents. The adequacy of the contractors' reviews was verified by YAEC audit and/or surveillance either at contractors' facilities or at the facilities of supplier and subcontractors. Audit and/or surveillance of contractors (AE) and suppliers was performed by YAEC home office personnel. Surveillance of subcontractors at the construction site was performed by the YAEC QA representatives assigned to the site. Audit at the construction site was performed by YAEC home office QA personnel.

b. Responsible Management Levels

Public Service Company of New Hampshire, had overall responsibility for quality assurance on the Seabrook Project. The Chief Executive Officer of PSNH delegated to YAEC the responsibility for estab-lishment and implementation of the Quality Assurance Program during construction, startup, and preoperation testing. He, or his staff, maintained cognizance of and evaluated the program activities in the following manner.

1. Reviewed and approved the YAEC Quality Assurance Program as described in the Seabrook Station Quality Assurance Manual.

2. Participated in major QA decisions and program changes.

3. Received copies of all YAEC audit reports (internal and external) pertaining to the Project. He received monthly the Status of Outstanding Items indicating the status of audit findings.

4. Participated on a quarterly basis in selected external audits by YAEC to assess YAEC performance in contractor activities. As an alternate to participating in the audit, he reviewed YAEC external audit reports. The diversity of audits were sufficient to ensure that YAEC complied with the requirements of Subsection 17.1.1.18. The NHY member of the audit team acted as an observer to assess the performance of the YAEC auditor(s).

c. <u>Scope of Delegation of Work</u>

YAEC delegated to the Project architect-engineers, United Engineers & Constructors Inc. (UE&C), and to the supplier of the Nuclear Steam Supply System, Westinghouse Electric Corporation (WRD) administration and execution of large portions of the quality assurance program associated with the design, procurement and installation of safety-related equipment as defined in Table 17.1-1, Table 17.1-2 and Table 17.1-3 of this program. Procurement of safety- related equipment was performed by either UE&C or Westinghouse under the provisions of their topical reports and QA programs. These were reviewed and concurred upon by YAEC personnel. Compliance to the aforementioned is ensured via a system of audits performed by YAEC Home Office personnel who also reviewed the UE&C and Westinghouse in- house departmental procedures. UE&C and WRD, and their vendors and subcontractors who were responsible for safety-related components or structures, were required to have quality assurance programs consistent with the requirements of 10 CFR 50 Appendix B, and of this program.

The UE&C and WRD quality assurance programs were extensions of the YAEC program and as such were reviewed and accepted by YAEC. The structure of the UE&C quality assurance organization is described in the UE&C Quality Assurance Programs (Topical Report No. UE&C-TR-001) and referenced in Subsection 17.1.2 of this Updated FSAR. The structure of the WRD quality assurance organization is described in the Westinghouse NES Division's Quality Assurance Plan Topical Report (WCAP-8370) and referenced in Subsection 17.1.3 of this Updated FSAR. Both Topical Reports were reviewed and approved by the Nuclear Regulatory Commission.

Conformance to approved requirements and programs was ensured through close liaison between the Project Managers of YAEC, NHY, WRD and UE&C and between their quality assurance organizations. Figure 17.1-2 depicts the managerial and quality assurance lines of authority, audit and communication between YAEC, NHY, WRD and UE&C and within these organizations. It also depicts responsibility to audit vendors and manufacturing divisions.

WRD as supplier of the Nuclear Steam Supply System (NSSS), presented its program in the Westinghouse WRD Divisions Quality Assurance Plan (WCAP-8370), and UE&C, the architect-engineer for this project, presented its program in the UE&C Quality Assurance Program (Topical Report No. UEC-TR-001). These programs were in effect and YAEC performed audits to ascertain WRD and UE&C compliance.

b. Safety-Related Structures, Systems, and Components

The safety-related structures, systems, and components listed in Table 17.1-1, Table 17.1-2, and Table 17.1-3 of this Updated FSAR were within the scope of the program. The contractors responsible for design and procurement were denoted in the table.

c. QA Program Schedule

All phases of the QA program were established at the earliest practical time consistent with the schedule for accomplishing activities affecting quality for the project. YAEC QAD reviewed contractor quality assurance procedures applicable to safety-related activities and performed audits or reviews, as required, to ensure implementation. Procedures concerning design and procurement activities were completed, reviewed and approved prior to that phase of the project. Construction procedures were prepared prior to the start of any quality-related activities at the site. Quality-related activities such as program and procedure reviews, specification reviews, procurement document reviews which were initiated prior to the submittal of the FSAR, were performed in accordance with approved procedures. Safety-related site studies (i.e., meteorology, geology) were performed in accordance with written approved procedures and were audited by YAEC or its agent(s).

Assurance by YAEC that these programs were properly implemented was accomplished by:

1. YAEC reviewed contractor QA programs to ensure compliance with the applicable criteria of 10 CFR 50, Appendix B

2. Audit programs conducted by YAEC and its contractors

3. YAEC's participation in periodic audits conducted by WRD and UE&C

4. Surveillance and audits at the construction site by YAEC Quality Assurance representatives.

4. Personnel were made aware of quality-related policies, manuals, and procedures that were mandatory requirements which must be implemented and enforced.

The measures which ensure that the YAEC indoctrination and training program defined the scope, objective and method of implementing the program and maintain proficiency of personnel include:

1. Section 17.1 of the Updated FSAR details the program objectives, scope and methods as required by the NRC Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants. The description meets the criteria of 10 CFR Part 50 Appendix B and of the NRC Standard Review Plan. The implementing program was prepared by the Construction Quality Assurance Manager and was approved by the YAEC Director of Quality Assurance.

2. Establishment of detailed YAEC departmental training programs which comply with FSAR commitments.

3. Auditing of departments performing quality affecting activities. The Quality Assurance Department performed the audits of other YAEC departments and it in turn was audited by NHY. During audits, the degree of compliance with policies and procedures was established and compliance with Project commitments was confirmed. Personnel become informed on the scope of Project technical and QA commitments by reviewing the FSAR and applicable referenced documents. They were instructed in the objectives, scope and details of manuals and instructions defining the YAEC control measures for work within individual departments and for interfacing, by both attendance at group indoctrination sessions conducted by QA personnel and by departmental supervision, meetings and directives. Special training in areas such as nondestructive testing, Boiler and Pressure Vessel Code, auditing and documentation was provided as required. The scope, objectives and methods employed to indoctrinate and train personnel were defined in departmental procedures. Assignments of work performed without direct supervision were made only to individuals who demonstrated that they were qualified, based on experience or training, to perform the tasks assigned.

k. Program Update

The Seabrook Station Quality Assurance Program was reviewed by YAEC QAD at least annually to ensure that it was kept current. YAEC performs audits on WRD and UE&C to ensure that their programs were kept up-to-date and effective.

The YAEC program for quality assurance normally involves three control levels:

Level 1 - Quality control by vendors, constructors and UE&C on the activities they perform, by YAEC on startup activities. This includes reviews, inspections and tests.

Level 2 - Surveillance of design, fabrication and construction activities, including Level 1 Quality Control. Contractors provided this level for the design and procurement phases. UE&C and YNSD provided additional surveillance on site construction activities.

Level 3 - Audits by YAEC QA Department of activities performed by Level 1 and 2 organizations.

YAEC provided the third level for all activities. At each level, the individual or group responsible for reviewing, inspecting, auditing or otherwise verifying that an activity has been correctly performed was independent of the individual or group responsible for performing the specific activity. The degree of control at each level reflected the importance of the activity to plant safety and reliability.

17.1.1.3 Design Control

WRD, as the nuclear steam system supplier, established a program for design control which is described in WCAP-8370. The program required WNES review of design specifications, appropriate drawings, calculations, procedures and instructions generated within the various Westinghouse organizations and by their suppliers.

UE&C, responsible for the balance-of-plant equipment up to the time of final acceptance by NHY, provided design control measures on other organizations within the scope of its responsibilities. Subsequent to final acceptance, design changes are controlled by the NHY Nuclear Production Operational Quality Assurance Program.

The UE&C program for design control is explained in UEC-TR-001.

d. Verification

Design verification or checking, such as design reviews, alternate calculations and qualification testing, were properly selected and performed. When a test program was used to verify the adequacy of a design, a qualification test of a prototype unit was performed under the most adverse design conditions. The individuals or groups who performed design verification or checking were other than those who performed the original design. These individuals or groups and their authority and responsibility were identified and controlled by written procedures. Compliance to Regulatory Guide 1.64, Revision 2, and ANSI N45.2.11 was ensured through the review of contractors QA manuals or procedures regarding Design Control. Implementation was verified via audits which sampled objective evidence of the design verification process. Selective review of contractor design documents by YAEC personnel was also performed and the verification and approval activity performed by the contractors was an element considered during this YAEC review and comment cycle. WRD and UE&C had the responsibility for assuring that proper design reviews or verifications were accomplished.

e. Interface Controls

Design interface controls of both external and internal participating organizations were procedurally described and controlled. Design documents, and revisions thereto, were distributed to the responsible persons in a timely and orderly manner and controlled to prevent inadvertent use of superseded documents. Design documents, design reviews, records, and changes thereto, were collected, stored and maintained in a systematic and controlled manner.

Interface material, including drawings, were provided to the Engineer-Constructor, UE&C, in the Westinghouse Project Information Package (PIP). The drawings included in the PIP provided information, such as location and safety class, which was used by UE&C in preparing balance-of-plant and nuclear system flow diagrams. The PIP also provided specific information on component design required by UE&C for their detailed design.

The UE&C drawing index lists all UE&C-originated drawings. Westinghouse reviewed this listing and selected those drawings which they required for information and their design purposes. As these drawings were issued and revised, they were sent to Westinghouse for their use. UE&C retains the responsibility for interface control.

All purchase specifications, and changes thereto, contained or referenced as applicable: design information and technical requirements including codes, standards, regulatory requirements, components and material identification; drawings, specifications, including their applicable revision; tests and inspection requirements; and special process instruction for such activities as fabrication, cleaning, erecting, packaging, handling, shipping, storing, and inspecting. The specifications contained requirements which identified the documents to be prepared, maintained, submitted, and made available to the buying agent for review and/or approval. The specifications contain, as appropriate, the requirement for reporting and disposition of nonconformances from procurement requirements. These documents included, as applicable, drawings, specifications, procedures, inspection and fabrication plans, inspection and test records, personnel and procedure qualifications, and materials, chemical and physical test results. The specifications also contained applicable requirements for the retention, control and maintenance of records, and the procuring agency's right of access to the vendor's facilities and records for source inspection and audit. Contract Procurement Documents specified the records that were to be delivered to the purchaser prior to use or installation of the hardware. Compliance to ANSI N45.2.9, N45.2.13 and ASME Section III was ensured via review of Contractor (UE&C & W) procedures, specifications and records requirements specified in procurement documentation. Verification was performed via a system of audits and/or surveillance performed by YAEC Home Office and/or Site QA personnel. The specifications contained provisions for extending applicable requirements of the document to subcontractors and suppliers, including purchaser's right of access to such subvendors' facilities and records.

Prior to their release to suppliers, procurement documents for safety-related items, equipment and services were subject to review by the originating organization. The review, conducted by qualified personnel, determined that quality requirements were correctly stated, inspectable, and controllable, that there were adequate acceptance and rejection criteria, and that the procurement document had been prepared and approved in accordance with program requirements. Prior to the issuance of purchase orders, WRD and UE&C submitted lists of potential suppliers (bidders list) to YAEC for technical review and approval. Subsequent to obtaining YAEC approval of the potential supplier list, WRD issued purchase orders to those suppliers listed. UE&C submitted bid packages (including a recommended supplier) to NHY for approval prior to releasing purchase documents.

17.1.1.5 Instructions, Procedures and Drawings

a. Quality Control Instructions

The Seabrook Station Quality Assurance Program included a system for controlling all documents, procedures, instructions, or drawings that were required for quality-related activities associated with the design, procurement, testing, inspecting, construction, preoperational testing and auditing of all safety-related material, structures, systems and components.

WRD and UE&C, as major suppliers, were responsible for establishing systems for controlling instructions, procedures, and drawings within their own organizations and those of their suppliers. The WRD program is described in WCAP-8370 and the UE&C program is described in UFC-TR-001. The control of instructions, procedures, and drawings, with YAEC, is described in the Seabrook Station Quality Assurance Manual. Those instructions, procedures and drawings prepared by the Startup Test Group were handled in accordance with written, approved procedures.

b. Acceptance Criteria

Activities affecting quality were defined in instructions, procedures, and drawings and included appropriate qualitative and quantitative acceptance criteria to ensure that specific activities were satisfactorily accomplished. References to these documents, when pertinent, identified the applicable revision. Instructions, procedures and drawings were reviewed and approved by appropriate supervisors or management. YAEC reviewed contractor quality assurance manuals to ensure incorporation of program requirements.

c. Audit and Surveillance

YAEC was responsible for auditing and surveillance of the WRD, UE&C, selected suppliers and site constructor programs to ensure that the instructions, procedures, and drawings used on safety-related equipment were controlled and met the requirements of 10 CFR 50, Appendix B.

	Document	Responsible Organization(s)
4.	Quality Assurance Manuals, procedures, and instructions	WRD-UE&C-YAEC
5.	Manufacturing and construction inspection, test and special process instructions and procedures	WRD-UE&C-YAEC
6.	FSAR and related design criteria documents	YAEC

c. Audits

YAEC Quality Assurance Department periodically verified, by audits and surveillance, that WRD, UE&C, selected suppliers, site constructors and the YAEC organization complied with these requirements.

17.1.1.7 Control of Purchased Material, Equipment, and Services

a. Selection and Control of Suppliers

The Seabrook Station Quality Assurance Program established controls to ensure that purchased construction material, equipment, and services, whether purchased directly or through contractors and subcontractors, conformed to the procurement document requirements. These measures included provisions for source evaluation and selection of vendors, objective evidence of quality furnished by the contractor or subcontractor, inspection and audit at the supplier source, and examination of products prior to or upon delivery.

Purchasing of safety-related items was the responsibility of WRD and UE&C as applicable. YAEC and Project contractors, WRD and UE&C, established measures for the control of purchased safety-related material, equipment and services applicable to the scope of their contracts. The UE&C measures were contained in Subsection 17.1.2 and the WRD measures were contained in WCAP-8370. These measures include:

1. Evaluation of suppliers prior to the award of procurement orders or contracts by qualified personnel using written procedures and check lists. Quality assurance and engineering personnel participated in the evaluation of those suppliers providing critical components.

(b) Inspection of the material, component or equipment, and acceptance records was performed and judged acceptable in accordance with predetermined inspection instructions. Items and their records were approved prior to installation or use.

(c) Inspection records or certificates of conformance attesting to the acceptance of material, components, and equipment were available at the nuclear power plant prior to installation or use.

(d) Items accepted and released were identified as to their approved inspection status prior to forwarding them to a controlled storage area or releasing them for installation or further work.

(e) Nonconforming items were segregated, controlled, and clearly identified until proper disposition was made.

Purchased material, equipment and services originated by the NHY station staff was controlled by the NHY Nuclear Production Operational Quality Assurance Program.

The YAEC Quality Assurance Department evaluates the control measures in the quality assurance programs of WRD and UE&C by reviewing their quality assurance programs and by a system of periodic audits. These evaluations provide assurance that they were capable of providing equipment, material and services which meet the applicable regulatory requirements. The audits were performed to verify that WRD and UE&C comply with the control measure applicable to the material, equipment, and services involved.

WRD and UE&C, based upon the complexity of purchased items and supplier performance history, performs source inspections or audits of vendors as necessary to ensure that the required quality of the items was obtained. Surveillance of supplier's fabrication, testing, inspection and shipment of materials, equipment and components was planned, performed and reported in accordance with written procedures which ensured conformance to the purchase order requirements.

Prior to the solicitation of bids, WRD and UE&C submitted lists of prospective suppliers to YAEC for review and approval and for suggested additions. YAEC recommended addition or removal of suppliers based on prior YAEC experience with the suppliers.

Upon receipt of items, material, or equipment at the site, UE&C performed receiving inspection. Receiving inspection verified that all required documentation was received, that the item, material or equipment conformed to the purchase order requirements, that the documentation was traceable to the item, material or equipment, and that the item, material or equipment was inspected for shipping damage. Certification was furnished by the supplier which identified any procurement requirements which have not been met together with a description of the disposition of each nonconformance. Appropriate records were maintained to indicate completion of these activities. Material, equipment or items lacking the required documentation were identified as nonconforming and placed in a "hold" status pending receipt of the necessary documentation.

b. Audits

The YAEC Quality Assurance Department audits WRD, UE&C and selected suppliers, and participated during WRD and UE&C audits of selected suppliers to assess the adequacy of supplier control measures for purchased material, equipment and services and of the WRD and UE&C audit systems. These YAEC audits, performed in accordance with requirements contained in the Seabrook Station Quality Assurance Manual, occurred at intervals consistent with the importance, complexity and quality of the item or service.

17.1.1.8 Identification and Control of Material, Parts and Components

The Seabrook Station Quality Assurance Program required that all organizations performing safety-related activities establish procedures to provide identification and control of materials, parts, and components, including partially fabricated assemblies, to prevent the use of incorrect or defective material, parts, and components and that measures ensured that identification of the item was maintained by a unique number either on the item or on records traceable to the item throughout fabrication, erection, installation, and use of the item. The location and method of identification did not affect the function or quality of the item being identified. Verification of identification was accomplished at appropriate stages throughout manufacturing, shipping, receipt, and installation.

WRD and UE&C developed methods for identification and control of materials, parts and components within the scope of their responsibilities. The UE&C program was detailed in UEC-TR-001 and the WRD program was detailed in WCAP-8370.

During the design stages, WRD and UE&C developed systems identification and assigned unique identification numbers, as appropriate, to items in a system. These numbers provide traceability of all associated documentation such as manufacturing and inspection documents, deviation reports, and material test reports.

17.1.1.10 Inspection

All organizations responsible for inspection of safety-related equipment and systems were required to have a documented program which includes the use of qualified inspection personnel and written inspection instructions.

The WRD inspection program for the manufacture of the NSSS equipment was detailed in WCAP-8370. The UE&C program for inspection of safety-related items for the balance of plant and for site activities was detailed in UEC-TR-001.

a. Inspection Program Implementation

The programs required that design specifications, drawings, purchase orders, procedures or instructions included the necessary inspection requirements with acceptance and rejection criteria. These inspection requirements were translated into inspection programs, procedures, and check lists, by manufacturing, construction, installation and test activities in order to specify, control and provide documented evidence of inspection activities. Inspection procedures, instruction and check lists contained the following:

1. Identification of characteristics to be inspected
2. Identification of individuals or groups responsible for performing the inspection operation
3. Acceptance and rejection criteria
4. Definition of the inspection method
5. Verification of inspection completion and certification
6. A record of the results of the inspection operation.

Inspections were performed for each work operation as necessary to verify quality.

d. Audits

YAEC performed audits of WRD and UE&C and participated in inspections at selected vendor facilities to verify implementation with specifications, applicable codes, standards, and regulatory guides. YAEC also performed surveillance of site constructor activities in accordance with this program.

17.1.1.11 Test Control

a. Test Control Implementation

YAEC assigned to WRD and UE&C the control of testing of safety-related materials, equipment, and structures during all phases of manufacturing, construction and installation.

The UE&C test program for material, equipment, and structures within the balance of plant and for site activities was detailed in UEC-TR-001. The WRD test program for the nuclear steam supply system components was detailed in WCAP-8370.

Supplier and subcontractor test procedures were subject to review and approval by the contractor having responsibility for the item.

The WRD and UE&C programs required that all testing necessary to demonstrate that materials, equipment, and structures perform satisfactorily in service was identified, accomplished, and documented in accordance with written controlled procedures. These procedures are based on the requirements of the codes and standards referenced in Table 3.2-3. These written procedures include requirements for the following:

1. Instructions for testing method and test equipment and instrumentation
2. Calibrated instrumentation
3. Adequate and appropriate equipment
4. Trained, qualified, and licensed or certified personnel
5. Preparation, condition, and completeness of the item to be tested
6. Suitable and controlled environmental conditions

These programs required that all organizations performing measuring or testing operations on safety-related materials, components, systems and structures had written procedures describing the calibration technique and frequency, maintenance, and control of all measuring and test instruments, tools, gages, fixtures, reference and transfer standards, and nondestructive test equipment which were used. Reference and transfer standards were required to have traceability to nationally recognized standards, or, where national standards did not exist, provisions were established to document the basis for calibration.

All measuring and test equipment was identified and the calibration test data was identified for the equipment to which it applied. The contractors (UE&C and WRD) were required to conform to a calibration requirement of marking, labeling or tagging of measuring and test equipment indicating date of next calibration. UE&C and WRD were committed to this requirement in their Topical Reports UEC-TR-001 and WCAP-8370. Suppliers and subcontractors were required to a similar provision in their QA programs which were approved by the Contractors and Constructor. The calibration frequency depended upon the required accuracy, purpose, degree of usage, stability characteristics and the manufacturers' recommendation. Records of the status of all items under the calibration system were maintained as required by ANSI N45.2.9.

UE&C, who was responsible for the procurement of equipment for the balance-of-plant, required in their quality assurance program that suppliers maintain a system which ensured that calibrating standards have an uncertainty (error) requirement of no more than 1/4 of the tolerance of electrical equipment being calibrated and 1/10 of the tolerance for all other equipment being calibrated, except where limited by the state-of-the-art.

Westinghouse, who was responsible for procurement of the NSSS equipment, required that when calibrating measuring and test equipment, typical transfer ratios of 10-1 were used for mechanical equipment; 4-1 for electrical equipment; and 4 or 5-1 for precision mechanical measuring equipment.

An investigation was conducted to determine the validity of previous inspections performed when measuring and test equipment was found to be out of calibration. The results of this investigation were documented. Inspections were repeated, as necessary, using calibrated equipment to establish acceptability of suspect items.

WRD and UE&C were responsible for imposing these requirements on their internal operations and on their vendors and constructors. WRD and UE&C performed audits and surveillance to ensure the adequacy of the program.

YAEC performed audits of WRD, UE&C, selected vendors, and site constructors, to ensure conformance with the program requirements.

Each organization supplying safety-related material, equipment, and structures established a system for identification of the inspection, test and operating status during all phases of their operation. The system was implemented by procedures which describe the use of status indicators such as labels, tags, stamps or routing cards that identify the status of the equipment at any given time.

The program ensured that operations performed out of sequence were controlled through documented measures under the cognizance of the applicable QA organization. Only authorized personnel were permitted to apply or remove tags, markings, or stamps used to indicate inspection, test, or operating status. Stamps used by personnel for completing items such as welds, inspections, and test were controlled and traceable to the user.

The operating status of nonconforming, inoperative, or malfunctioning structures, systems, or components was identified to prevent their inadvertent use in accordance with Subsection 17.1.1.15.

WRD and UE&C were responsible for imposing these requirements on their respective operations and on their vendors.

YAEC, through a system of audits at WRD, UE&C, selected vendors, and site constructors ensured timely implementation of these requirements.

17.1.1.15 Nonconforming Materials, Parts and Components

WRD and UE&C were delegated the responsibility for specifying requirements for the control of nonconforming materials, parts and components during the design, procurement, and construction phases of the project.

The WRD program for the handling, disposition and control of nonconforming items is detailed in WCAP-8370 and the UE&C program is detailed in UEC-TR-001.

The programs established requirements for the following measures by WRD, UE&C, vendors, and constructors:

a. Procedures to control the identification, documentation, segregation, review, disposition and notification to affected organizations of nonconforming materials, parts, components, or services.

b. Documentation which identified the nonconforming item described the nonconformance, the disposition of the nonconformance, and the inspection requirements, and included signature approval of the disposition.

When a review by WRD or UE&C indicated that a nonconformance was reportable as defined in Paragraph 50.55(e) of Part 50, Title 10, of the Code of Federal Regulations, WRD or UE&C, as applicable, submitted a report of the nonconformance to the YAEC Project Manager clearly identifying the nonconformance as a possible reportable deficiency. Following disposition of the nonconformance report (reportable as defined in 10 CFR 50, 50.55(e)), but prior to any repair or rework, WRD or UE&C, as applicable, submitted a copy of the report to YAEC with the recommendation that the occurrence be reported. Sufficient justification and data of the proposed action was included to allow preparation of the report required for the regulatory authorities. Repair or rework required YAEC approval.

Nonconformance reports were reviewed by YAEC, WRD and UE&C to ascertain quality trends, and the results were documented and reported to the appropriate management.

WRD and UE&C performed audits and surveillance within their internal operations and those of their vendors and contractors to ensure compliance with the program requirements.

Items identified as nonconforming during the conduct of plant preoperational and startup tests were identified and documented as described in Chapter 14.

YAEC, through audits of WRD, UE&C, site contractors, and selected vendors, ensured overall conformance to these requirements.

17.1.1.16 Corrective Action

YAEC and their contractors, WRD and UE&C, developed programs for the control and implementation of corrective action for all safety-related activities.

The WRD program for the control and implementation of corrective action was detailed in WCAP-8370 and the UE&C program was detailed in UEC-TR-001. The YAEC program was contained in the Seabrook Station Quality Assurance Manual.

a. Identification and Correction

The Seabrook Station Quality Assurance Program required that conditions adverse to quality such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and nonconformances were promptly identified and corrected.

f. <u>Audits</u>

WRD and UE&C performed audits and surveillance on their own operations and those of their vendors to ensure compliance with these requirements.

YAEC, through a system of planned audits of WRD, UE&C, and selected vendors, and a program of internal audits of YAEC departments ensured their conformance to the program requirements.

17.1.1.17 Quality Assurance Records

WRD and UE&C were responsible for the collection of all quality assurance records generated within the scope of their responsibilities and submittal of these records to YAEC prior to fuel loading. The quality assurance records and the required storage and retrieval system were designed to fulfill the requirements of ANSI 45.2.9 "Requirements for Collection, Storage, and Maintenance of Quality Assurance Records for Nuclear Power Plants," with the exception that the records storage facility at the plant site was not tornado proof. The Startup Test Group was responsible for the control of records associated with the initial test program and for the transfer of all relevant data in accordance with Project procedures.

The WRD program providing for collection, storage and maintenance of quality records was described in WCAP-8370 and the UE&C program was described in UEC-TR-001.

a. <u>Objective Evidence</u>

The programs required that records documenting evidence of quality of items and activities include operating logs and the results of reviews, inspections, tests, audits, monitoring of work performance, and material analyses. The records included closely related data such as qualification of personnel, procedures, and equipment. Other documents retained include drawings, specifications, procurement documents, special process and calibration procedures, calibration reports, and nonconforming and corrective action reports. Requirements and responsibilities for record transmittals, retention, and maintenance, subsequent to completion of work or prior to release of material or equipment for installation, were indicated in specifications, procedures and quality programs.

b. <u>Identification and Retrievability</u>

The program specified that inspection and test records contain at least the following:

1. A description of the type of operation

17.1.1.18 Audits

WRD and UE&C had a comprehensive system of planned and periodic audits to determine the effectiveness and implementation of their respective programs and those of their vendors. The WRD audit program is described in WCAP-8370 and the UE&C program was described in UEC-TR-001.

The Seabrook Station Quality Assurance Program includes a comprehensive system of planned and periodic audits carried out by the YAEC quality organization as activities were performed to verify compliance with the program requirements. The system provides data for a continuing evaluation of the program effectiveness.

NHY, YAEC, WRD and UE&C established audit programs which comply with 10 CFR 50, Appendix B, and ANSI 45.2.12, "Requirements for Auditing of Quality Assurance Programs for Nuclear Power Plants." The program included the following requirements:

a. Performance of the following types of audits by NHY, YAEC, WRD and UE&C:

   1. Management audits which provided verification and evaluation of the Quality Assurance Program procedures, and activities to ensure that they effectively complied with corporate policy and with codes, standards and applicable regulatory guides.

   2. Internal audits by the Quality Assurance organization to provide independent verification and evaluation of quality-related procedures and activities to ensure that they effectively complied with the QA program.

   3. External audits performed on suppliers. These audits included verification and evaluation of the supplier's QA program, procedures, and activities to ensure that they effectively complied with all aspects of the QA program and procurement requirements.

b. Establishment of the requirement that audits were performed in those areas where the requirements of Appendix B to 10 CFR 50 were implemented. These areas included, as a minimum, those activities associated with:

   1. Site-related studies which affected plant safety analyses

   2. The preparation, review, approval, and control of the FSAR, designs, specifications, procurement documents, instructions, procedures, and drawings

d. YAEC internal audits.

Audits of WRD and UE&C were conducted as early as possible in the program to ensure compliance with the requirements of codes, standards, applicable regulatory guides and quality assurance provisions. Program areas were subsequently audited consistent with the project schedule or where quality concerns were noted, but, as a minimum, they were audited annually or at least once during the life of the contract, whichever was shorter. The audit frequency may be increased based on experience obtained. Generally, two or three partial audits per year were conducted at both WRD and UE&C to ensure compliance with contract and regulatory requirements and to permit early verification of corrective action.

Independently, or as participants in WRD or UE&C audit teams, YAEC audited selected vendors such as equipment fabricators, material suppliers, consultants and organizations working onsite preparation activities. The audits were based on the safety and code class of the item involved, the complexity of the item, and prior YAEC experience with the supplier. These audits were in addition to these performed by YAEC in conjunction with visits associated with pre-established notification points.

Regularly scheduled audits were supplemented when one or more of the following conditions existed:

a. When it was necessary to determine the capability of a contractor's quality assurance program prior to award of a purchase order.

b. When, after award of a purchase order, sufficient time had elapsed for implementing the quality assurance program and it was appropriate to determine that the organization was adequately performing the functions as defined in the quality assurance program description, codes, standards, and other contract documents.

c. When significant changes were made in functional areas of the quality assurance program such as significant reorganization or procedure revisions.

d. When it was suspected that the quality of the item was in jeopardy due to nonconformance in the quality assurance program.

e. When a systematic, independent assessment of program effectiveness was considered necessary.

f. When it was necessary to verify implementation of required corrective action.

17.1.3 Westinghouse Quality Assurance Program

The original Quality Assurance Program implemented by Westinghouse for Seabrook was described in RESAR-3, Amendment 4, as referenced by Updated FSAR Section 17.3. Over the course of performing the design and initial procurement activities for Seabrook, the Westinghouse Quality Assurance Program was upgraded to reflect changes in regulatory requirements and industry standards. These changes first culminated in Westinghouse topical report, WCAP-8370, Revision 7A (Reference 1) which was applicable to activities from January 1, 1975 to October 1, 1977, as documented in PSAR Amendment 24. This was superseded by Westinghouse topical report, WCAP-8370, Revision 8A (Reference 2) which was applicable to activities from October 1, 1977 to October 31, 1979, and by WCAP-8370, Revision 9A, which was applicable from October 31, 1979 to November 30, 1984.

The Westinghouse Nuclear Fuel Division Quality Assurance Program was described in Westinghouse Topical Report, WCAP-7800, Revision 5 (Reference 4).

The present Westinghouse Water Reactor Divisions Quality Assurance Plan was described in WCAP 8370/7800, Revision 10A/6A (Reference 5) and applies to all Westinghouse Water Reactor Division's (including Nuclear Fuel Division's) activities subsequent to November 30, 1984.

17.1.4 References

1. "Quality Assurance Plan Westinghouse Nuclear Energy Systems Divisions," WCAP-8370, Revision 7A, February 1975.

2. "Westinghouse Water Reactor Divisions Quality Assurance Plan," WCAP-8370, Revision 8A, September 1977.

3. "Westinghouse Water Reactor Divisions Quality Assurance Plan." WCAP-8370, Revision 9A, October 1979.

4. "Nuclear Fuel Division Quality Assurance Program Plan," WCAP-7800, Revision 5, December 1977.

5. "Westinghouse Water Reactor Division Quality Assurance Plan," WCAP-8370/7800, Revision 10A/6A, November 1984.

17A EXCEPTIONS, ALTERNATIVES, AND CLARIFICATIONS TO PROGRAM STANDARDS, INDUSTRY CODES, FEDERAL REGULATIONS AND GUIDES

General

This information is located in the FPL Quality Assurance Topical Report.

17C COMPANY NUCLEAR REVIEW BOARD (CNRB)

17C.1 FUNCTION

The CNRB functions have been transferred to the Station Operation Review Committee as described in the FPL Quality Assurance Topical Report.

SEABROOK STATION
UPDATED FINAL SAFETY ANALYSIS REPORT

CHAPTER 17
QUALITY ASSURANCE

TABLES

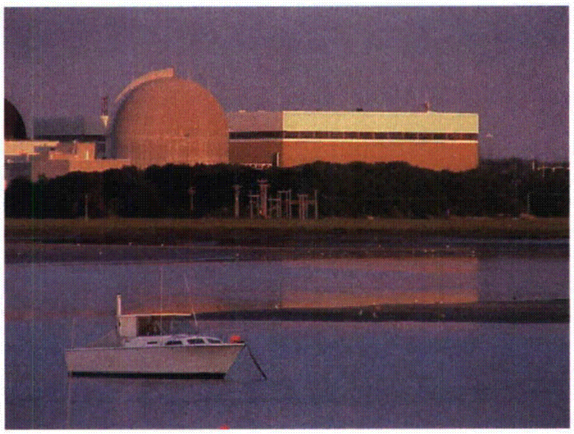


Table 17.1-2 Safety-Related Electrical Systems And Instrumentation

Description	Contractor
4160 Volt Switchgear (Engineered Safety Features Buses)	UE&C
4160 - 480 Volt Transformer (Associated with Engineered Safety Features)	UE&C
4000 and 460 Volt Motors (Associated with Engineered Safety Features)	UE&C/WRD
4160 Volt Nonsegregated Group Phase Buses (Associated with Engineering Safety Features)	UE&C
480 Volt Load Centers (Associated with Engineered Safety Features)	UE&C
125 Volt DC Batteries (Associated with Engineered Safety Features)	UE&C
Battery Chargers (Associated with Engineered Safety Features)	UE&C
Inverters, 125 Volt DC to 120 Volt AC (Vital Instrument Buses)	UE&C/WRD
Vital Instrument Bus Panels	UE&C
125 Volt DC Power Panels (Associated with Engineered Safety Features)	UE&C
125 Volt DC Switchboards (Associated with Engineered Safety Features)	UE&C
Electrical Tray and Conduit Supports, Fittings and Accessories (Associated with Engineered Safety Features)	UE&C
Containment Penetration Assemblies	UE&C
Power Cables (Associated with Engineered Safety Features System)	UE&C
Instrumentation and Control Cables (Associated with Engineered Safety Feature System)	UE&C
Diesel Generators	UE&C

TABLE 17.1-3 SAFETY-RELATED MECHANICAL EQUIPMENT

	Contractor
Reactor Coolant System	
Reactor	WRD
Full Length Control Rod Drive Mechanism Housing	WRD
Steam Generator	WRD
Pressurizer	WRD
Reactor Coolant Piping, Fittings, and Fabrication	WRD
Surge Pipe, Fittings and Fabrication	WRD
Bypass Manifold	WRD
Reactor Coolant Thermowells	WRD
Safety Valves	WRD
Relief Valves	WRD
Valves to Reactor Coolant System Boundary	WRD/UE&C
Control Rod Drive Mechanism Head Adapter Plugs	WRD
Reactor Coolant Pump	WRD
Internals	WRD
Fuel	WRD
Handling Equipment for Fuel and Reactor Vessel Internals	
Fuel Transfer Tube Outer Sleeve	UE&C
Expansion Joints	UE&C
Reactor Vessel Head Lifting Device	WRD
Fuel Transfer System	WRD
Fuel Transfer Tube and Flange	WRD
Conveyor System and Controls	WRD
Spent Fuel Racks	UE&C

	Contractor
Mixed Bed Demineralizer	UE&C
Cation Bed Demineralizer	UE&C
Reactor Coolant Filter	UE&C
Charging Pumps Centrifugal	WRD
Positive Displacement Charging Pump	WRD
Seal Water Injection Filter	UE&C
Excess Letdown Heat Exchanger	WRD
Seal Water Return Filter	UE&C
Seal Water Heat Exchanger	WRD
Boric Acid Tanks	UE&C
Boric Acid Transfer Pumps	WRD
Boric Acid Blender	UE&C
Boric Acid Filter	UE&C
Volume Control Tank	WRD
Bypass Orifice	WRD
Letdown Flow Control Valves	WRD
Boric Acid Transfer Pump Bypass Orifice	UE&C
Demineralizer Prefilter	UE&C
Letdown Strainers	UE&C
Boron Thermal Regeneration Subsystem	
Moderating Heat Exchanger	WRD
Letdown Chiller Heat Exchanger	WRD
Letdown Ceheat Heat Exchanger	WRD
Thermal Regeneration Demineralizer	WRD

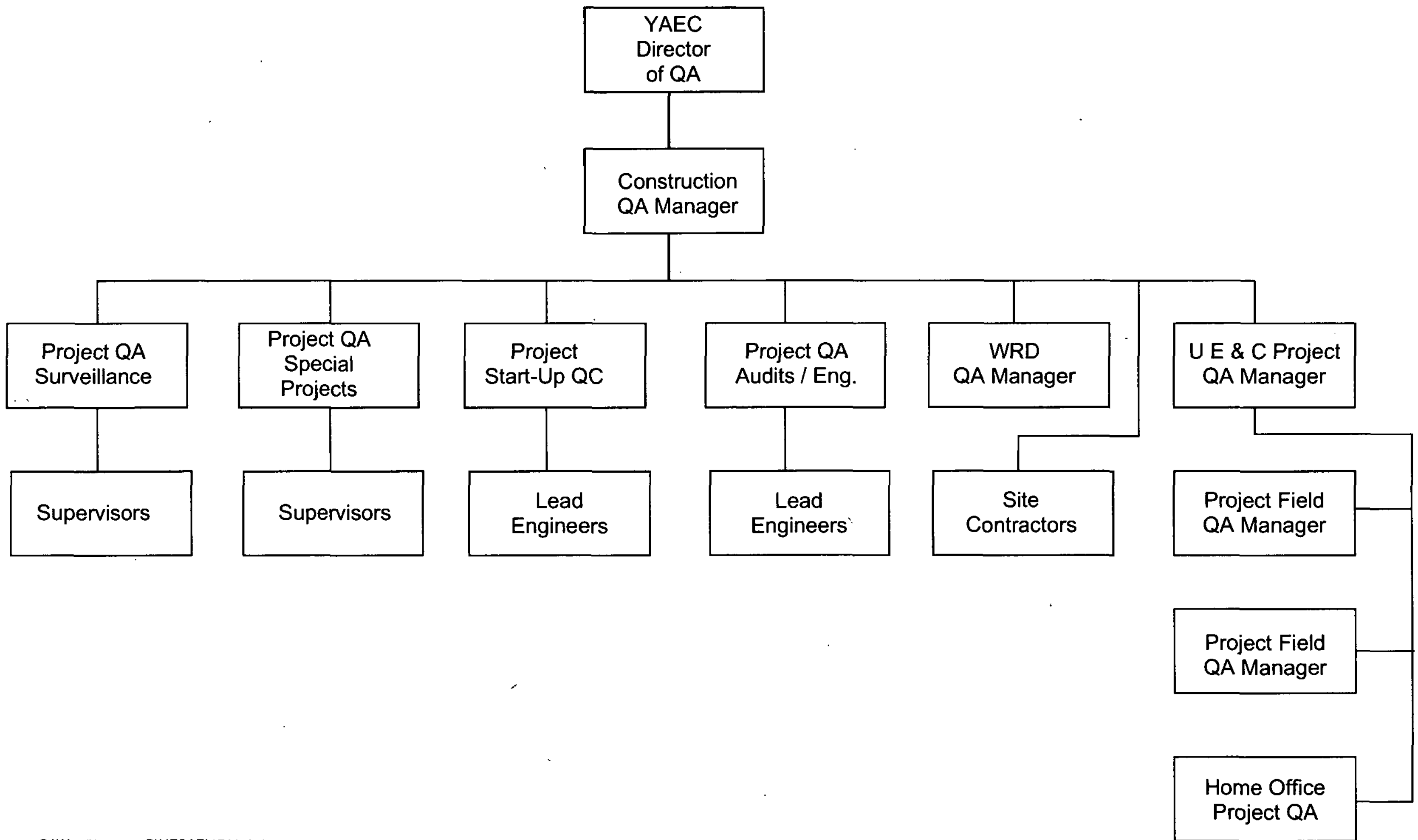
	Contractor
Sample System	
Isolation Valves	UE&C
Piping (up to and including isolation valves)	UE&C
Ventilation Cleanup and Air Conditioning Systems	
Containment Enclosure Exhaust	UE&C
Diesel Generator Building Ventilation	UE&C
Battery Room Ventilation	UE&C
Control Room Air Conditioning	UE&C
Control Room Emergency Filters	UE&C
Fuel Storage Building Emergency Purge System	UE&C
Safeguard Pump Rooms Cooling	UE&C
Control Room Complex Makeup Air System	UE&C
Emergency Diesel Generator System	
Diesel Fuel Storage Tank	UE&C
Diesel Fuel Day Tank	UE&C
Diesel Generator Air Tank	UE&C
Diesel Fuel Transfer Pump	UE&C
Diesel Fuel Filter	UE&C
Diesel Engines	UE&C
Spent Fuel Pool Cooling and Cleanup System	
Spent Fuel Pool Pump	UE&C
Spent Fuel Pool Heat Exchanger	UE&C

TABLE 17.1-4 SEABROOK STATION QUALITY ASSURANCE MANUAL COMPLIANCE WITH 10 CFR 50 APPENDIX B

Table of Contents			Compliance with Applicable Section 10 CFR 50 Appendix B
Preface			1,2
Table of Contents			
Definitions			
1.0	Program		
	1.1	Design and Procurement	1 through 18
	1.2	Construction	1 through 18
2.0	Organization		1
3.0	Review Control		
	3.1	External Interface Controls	3,6
	3.2	Review Control	3,6
	3.3	Review Procedure	3,6
		Appendix A - Engineering Specification	3
		Appendix B - Engineering Drawing	3
		Appendix C - Purchase Documents	3
		Appendix D - QA/QC Program/Manual/Procedure	5
4.0	Document Control		
	4.1	Project Document Control	5,6
	4.2	Manual Control	5,6
5.0	Purchased Material Control		
	5.1	Control of Purchased Material, Equipment and Services	4,7,8,10
8.0	Corrective Action		
	8.1	Corrective Action	16

Table 17.1-5 Supplemental Procedures

Procedure	Proc. No.
Administration of QA Program	WE-001
Indoctrination of Personnel	WE-003
Training	WE-004
Calculations and Analysis	WE-103
Computer Codes	WE-108
Quality Assurance Training	Q-101
QA Training Program	Q-102
Audit Primer	Q-103
Qualification and Certification of Inspection and Testing Personnel to ANSI N45.2.6	Q-106
Document Control Center Interface	Q-107
Quality Assurance for Fire Protection	Q-110
Project Policy Manual (Seabrook)	
Document Control Center Manual	
Seabrook Station Field Quality Assurance Manual and Procedure	
Procedure for Blast Monitoring	PSY Proc. 1
Procedure for Monthly Maintenance Program - Blast Monitoring Equipment	
Procedure for Operator Training Program (Blast Monitoring)	PSY Proc. 3
Procedure for Control of YAEC Generated Procedures (Site Related)	PSY Proc. 4
Procedure for Geological Mapping Program	PSY Proc. 5
Procedure for Procurement Control	PSY Proc. 6



G:\Word\Images_P\UFSAR\1711.ds4

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	YAEC Quality Assurance Interfaces Figure 17.1-1

Deleted