5.1 SUMMARY DESCRIPTION

The reactor coolant system includes those systems and components which contain or transport fluids coming from, or going to the reactor core. These systems form a major portion of the reactor coolant pressure boundary. This chapter provides information regarding the reactor coolant system and pressure-containing appendages out to and including isolation valving. This grouping of components is defined as the reactor coolant pressure boundary (RCPB) as follows:

Reactor coolant pressure boundary (RCPB) includes all pressure containing components such as pressure vessels, piping, pumps, and valves, which are:

(1) Part of the reactor coolant system, or

(2) Connected to the reactor coolant system, up to and including any and all of the following:

   a. The outermost containment isolation valve in piping which penetrates primary reactor containment,

   b. The second of the two valves normally closed during normal reactor operation in system piping which does not penetrate primary reactor containment,

   c. The reactor coolant system safety/relief valves.

This chapter also describes various subsystems connected to the RCPB. Specifically Section 5.4 deals with these subsystems.

The nuclear system pressure relief system protects the reactor coolant pressure boundary from damage due to overpressure. To protect against overpressure, pressure-operated relief valves are provided that can discharge steam from the nuclear system to the suppression pool. The pressure relief system also acts to automatically depressurize the nuclear system in the event of a loss-of-coolant accident in which the high pressure coolant injection (HPCI) system fails to maintain reactor vessel water level. Depressurization of the nuclear system allows the low pressure core cooling systems to supply enough cooling water to adequately cool the fuel. In addition to these two modes to protect against overpressurization, the pressure relief system can also be remotely operated.

Limits on RCPB leakage inside the drywell are established so that appropriate action can be taken before the integrity of the RCPB is impaired (see Subsection 5.2.5).

The reactor vessel and appurtenances are described in the Section 5.3. The major safety consideration for the reactor vessel is the ability of the vessel to function as a radioactive material barrier. Various combinations of loading are considered in the vessel design. The vessel meets the requirements of applicable codes and criteria. The possibility of brittle fracture is considered, and suitable design, material selection, material surveillance activity and operational limits are established that avoid conditions where brittle fracture is possible.

The reactor recirculation system provides coolant flow through the core. Adjustment of the core coolant flow rate changes reactor power output, thus providing a means of adjusting power without moving control rods. The recirculation system is designed to provide a slow coastdown

of flow so that fuel thermal limits cannot be exceeded as a result of recirculation system malfunctions. The arrangement of the recirculation system routing is such that a piping failure cannot compromise the integrity of the floodable inner volume of the reactor vessel.

The main steamline flow restrictors of the venturi-type are installed in each main steamline inside the primary containment. The restrictors are designed to limit the loss of coolant resulting from a main steamline break outside the primary containment. The coolant loss is limited so that reactor vessel water level remains above the top of the core during the time required for the main steamline isolation valves to close. This action protects the fuel barrier.

Two isolation valves are installed on each main steamline; one is located inside, and the other is located outside the primary containment. In the event that a main steamline break occurs inside the containment, closure of the isolation valve outside the primary containment acts to seal the primary containment itself. The main steamline isolation valves automatically isolate the reactor coolant pressure boundary in the event a pipe break occurs downstream of the isolation valves. This action limits the loss of coolant and the release of radioactive materials from the nuclear system.

The reactor core isolation cooling (RCIC) system provides makeup water to the core during a reactor shutdown in which feedwater flow is not available. The system is started automatically upon receipt of a low reactor water level signal or manually by the operator. Water is pumped to the core by a turbine-pump set driven by reactor steam.

The residual heat removal (RHR) system includes a number of pumps and heat exchangers that can be used to cool the nuclear system under a variety of situations. During normal shutdown and reactor servicing, the RHR system removes residual and decay heat. One mode of RHR operation allows the removal of heat from the primary containment following a loss of coolant accident. Another operational mode of the RHR system is low pressure coolant injection (LPCI). LPCI operation is an engineered safety feature for use during a postulated loss-of-coolant accident (see Section 6.3). The core spray system (CS) is a redundant, low pressure coolant injection system which sprays water onto the top of the fuel assemblies to cool the core (Section 6.3.2.2.3).

The reactor water cleanup system (Section 5.4.8) recirculates a portion of reactor coolant through filter-demineralizers to remove particulate and dissolved impurities from the reactor coolant. It also removes excess coolant from the reactor system under controlled conditions.

Design and performance characteristics of the Reactor Coolant System and its various components will be found in Table 5.1-1.

5.1.1 Schematic Flow Diagram

Schematic flow diagrams of the reactor coolant system denoting all major components, principal pressures, temperatures, flow rates, and coolant volumes for normal steady-state operating conditions at rated power are presented in Figures 1.2-49, 1.2-49-1, 1.2-49-2, 1.2-49-3 and 5.1-2.

5.1.2 Piping and Instrumentation Diagram

Piping and instrumentation diagrams covering the systems included within the reactor coolant system and connected systems are presented in the following:

(1) the nuclear boiler and vessel instrumentation shown on Dwgs. M-141, Sh. 1, M-141, Sh. 2 and M-142, Sh. 1;

(2) recirculation system shown on Dwg. M-143, Sh. 1;

(3) reactor core isolation cooling system shown on Dwgs. M-149, Sh. 1 and M-150, Sh. 1;

(4) residual heat removal system shown on Dwgs. M-151, Sh. 1 and M-151, Sh. 3;

(5) reactor water cleanup system shown on Dwg. M-144, Sh. 1;

(6) high pressure coolant injection shown on Dwgs. M-155, Sh. 1 and M-156, Sh. 1;

(7) core spray system shown on Dwg. M-152, Sh. 1;

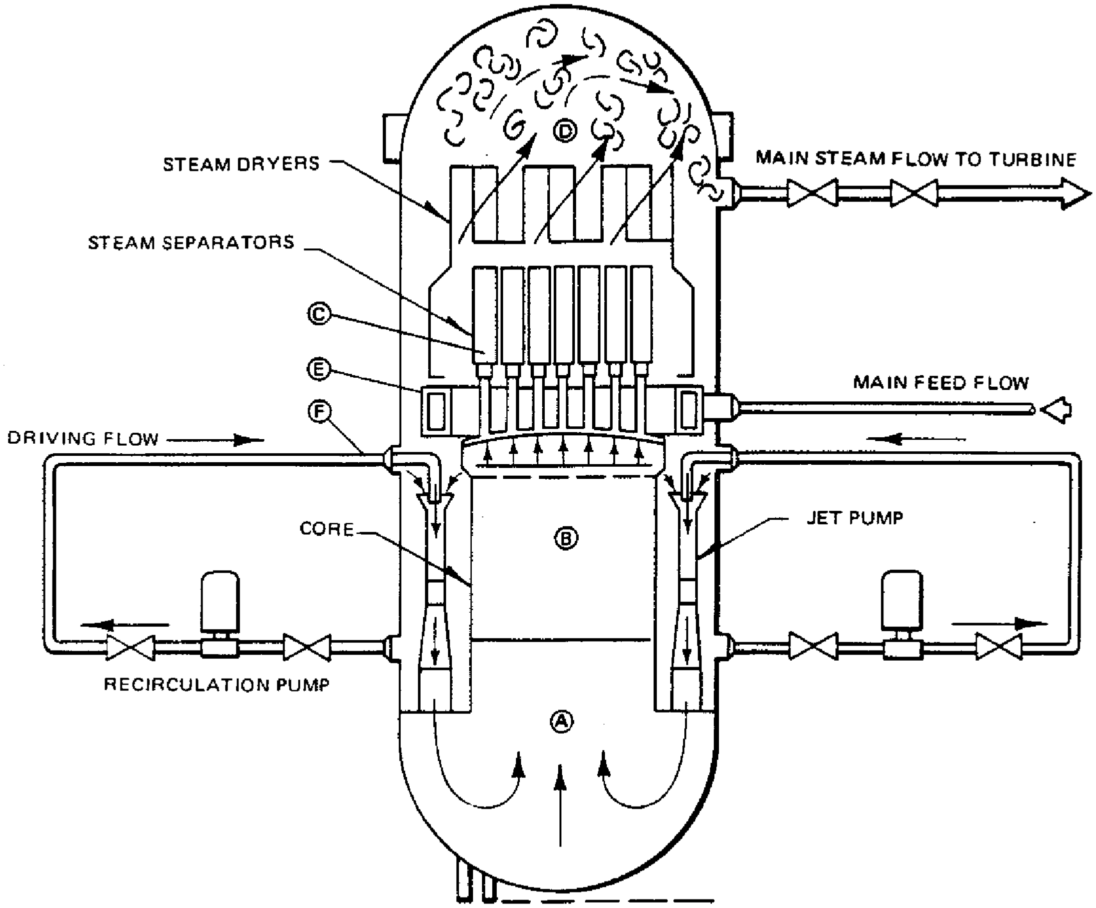
(8) standby liquid control system shown on Dwg. M-148, Sh. 1.

5.1.3 Plan and Elevation Drawings

Plan and elevation drawings showing the principal dimensions of the reactor coolant system in relation to the containment are shown in Figure 5.1-4-1, 5.1-4-2 and 5.1-4-3.

Table 5.1-1
DESIGN AND PERFORMANCE CHARACTERISTICS OF
THE REACTOR COOLANT SYSTEM AND ITS COMPONENTS

Security-Related Information
Table Withheld Under 10 CFR 2.390



	VOLUME OF FLUID (ft^3)
A. LOWER PLENUM	3887
B. CORE	2054
C. UPPER PLENUM AND SEPARATORS	1321
D. DOME (ABOVE NORMAL WATER LEVEL)	7934
E. DOWNCOMER REGION	5830
F. RECIRC LOOPS AND JET PUMPS	1398

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 AND 2
FINAL SAFETY ANALYSIS REPORT

COOLANT VOLUMES
OF THE
BOILING WATER REACTOR

FIGURE 5.1-2, Rev. 49

Auto Cad: Figure Fsar 5_1_2.dwg

THIS FIGURE HAS BEEN
REPLACED BY DWG.
M-141, Sh. 1

FSAR REV. 65

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Figure 5.1-3A-1 replaced by dwg.
M-141, Sh. 1

FIGURE 5.1-3A-1, Rev. 56

AutoCAD Figure 5_1_3A_1.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
M-141, Sh. 2

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
Figure 5.1-3A-2 replaced by dwg. M-141, Sh. 2 FIGURE 5.1-3A-2, Rev. 55

AutoCAD Figure 5_1_3A_2.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
M-142, Sh. 1

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
Figure 5.1-3B replaced by dwg. M-142, Sh. 1 FIGURE 5.1-3B, Rev. 55

AutoCAD Figure 5_1_3B.doc

Security-Related Information
Figure Withheld Under 10 CFR 2.390

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
REACTOR COOLANT SYSTEM PLAN
FIGURE 5.1-4-1

Security-Related Information
Figure Withheld Under 10 CFR 2.390

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
REACTOR COOLANT SYSTEM ELEVATION DIAGRAM FIGURE 5.1-4-2

Security-Related Information
Figure Withheld Under 10 CFR 2.390

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
REACTOR COOLANT SYSTEM ELEVATION DIAGRAM FIGURE 5.1-4-3

5.2 INTEGRITY OF REACTOR COOLANT PRESSURE BOUNDARY

This section discusses measures employed to provide and maintain the integrity of the reactor coolant pressure boundary (RCPB) for the plant design lifetime.

5.2.1 COMPLIANCE WITH CODES AND CODE CASES

5.2.1.1 Compliance with 10CFR50, Section 50.55a

A table which shows compliance with the rules of 10CFR50 is included in Section 3.2, (See Table 3.2-1). Code edition, applicable addenda, and component dates are in accordance with 10CFR50.55a. Table 5.2-10 lists those RCPB components which comply with the rules of 10CFR50 in accordance with 10CFR50.55(a)(2)(ii).

5.2.1.2 Applicable Code Cases

The reactor pressure vessel and appurtenances, and the RCPB piping, pumps and valves, have been designed, fabricated, and tested in accordance with the applicable edition of the ASME Code, including addenda that were mandatory at the order date for the applicable components. Section 50.55a of 10CFR50 requires code case approval only for Class 1 components. These code cases contain requirements or special rules which may be used for the construction of pressure-retaining components of Quality Group Classification A. The various ASME code case interpretations that were applied to components in the RCPB are listed in Table 5.2-1.

5.2.2 OVERPRESSURE PROTECTION

This section provides evaluation of the system that protects the RCPB from overpressurization.

5.2.2.1 Design Basis

Overpressure protection is provided in conformance with 10CFR50, Appendix A, General Design Criteria 15. Preoperational and startup instructions are given in Chapter 14.

5.2.2.1.1 Safety Design Bases

The nuclear pressure-relief system has been designed:

(1) To prevent overpressurization of the nuclear system that could lead to the failure of the reactor coolant pressure boundary.

(2) To provide automatic depressurization for small breaks in the nuclear system occurring with maloperation of the high pressure coolant injection (HPCI) system so that the low pressure coolant injection (LPCI) and the core spray (CS) systems can operate to protect the fuel barrier.

(3) To permit verification of its operability.

(4) To withstand adverse combinations of loadings and forces resulting from normal, upset, emergency and faulted conditions.

5.2.2.1.2 Power Generation Design Bases

The nuclear pressure relief system safety/relief valves have been designed to meet the following power generation bases:

(1) Discharge to the containment suppression pool.

(2) Correctly reclose following operation so that maximum operational continuity can be obtained.

5.2.2.1.3 Discussion

The ASME Boiler and Pressure Vessel Code requires that each vessel designed to meet Section III be protected from overpressure under upset conditions. The code allows a peak allowable pressure of 110% of vessel design pressure under upset conditions. The code specifications for safety valves require that: (1) the lowest safety valve set point be set at or below vessel design pressure and (2) the highest safety valve set point be set so that total accumulated pressure does not exceed 110% of the design pressure for upset conditions. The safety/relief valves are designed to open via either of two modes of operation as described in Section 5.2.2.4.1. The safety (spring lift) set points are listed in Table 5.2-2. These setpoints satisfy the ASME Code specifications for safety valves, because all valves open at less than the nuclear system design pressure of 1250 psig.

The automatic depressurization capability of the nuclear system pressure relief system is evaluated in Section 6.3 and in Section 7.3.

The following detailed criteria are used in selection of safety relief valves:

(1) Must meet requirements of ASME Code, Section III;

(2) Valves must qualify for 100% of nameplate capacity credit for the overpressure protection function;

(3) Must meet other performance requirements such as response time, etc., as necessary to provide relief functions.

The safety/relief valve discharge piping is designed, installed, and tested in accordance with the ASME Code, Section III.

5.2.2.1.4 Safety Valve Capacity

The safety valve capacity of this plant is adequate to limit the primary system pressure, including transients, to the requirements of the ASME Boiler and Pressure Vessel Code, Section III, Nuclear Vessels (up to and including Summer 1970 Addenda for Unit 1 and Unit 2).

It is recognized that the protection of vessels in a nuclear power plant is dependent upon many protective systems to relieve or terminate pressure transients. Installation of pressure relieving devices may not independently provide complete protection. The safety valve sizing evaluation assumes credit for operation of the scram protective system which may be tripped by one of two sources; i.e., a direct or high neutron flux trip signal. The direct scram trip signal is derived from position switches mounted on the main steam line isolation valves or the turbine stop valves or from pressure switches mounted on the dump valve of the turbine control valve hydraulic actuation system. The position switches are actuated when the respective valves are closing and following 10% travel of full stroke. The pressure switches are actuated when a fast closure of the turbine control valves is initiated. Further, no credit is taken for power operation of the pressure relieving devices. Credit is taken for the dual purpose safety/relief valves in their ASME Code qualified (spring lift) mode of safety operation.

The rated capacity of the pressure relieving devices are sufficient to prevent a rise in pressure within the protected vessel of more than 110% of the design pressure (1.10 x 1250 psig = 1375 psig) for events defined in Subsection 5.2.2.2.2.2.

Full account is taken of the pressure drop on both the inlet and discharge sides of the valves. All combination safety/relief valves discharge into the suppression pool through a discharge pipe from each valve which is designed to achieve sonic flow conditions through the valve; thus providing flow independence to discharge piping losses.

Table 5.2-6 lists the systems which could initiate during the design basis overpressure event.

5.2.2.2 Design Evaluation

The overpressure protection analysis is performed using cycle specific inputs. Hence, the overpressurization analysis must be evaluated for each reload cycle. This section contains the reload analysis results for both units.

5.2.2.2.1 Method of Analysis

To design the pressure protection for the nuclear boiler system, extensive analytical models representing all essential dynamic characteristics of the system are simulated. These models include the hydrodynamics of the flow loop, the reactor kinetics, the thermal characteristics of the fuel and its transfer of heat to the coolant, and all the principal controller features, such as feedwater flow, recirculation flow, reactor water level and pressure. These are represented with all their principal nonlinear features in models that have evolved through extensive experience and favorable comparison of analysis with actual BWR test data.

Unit 2 cycles starting with cycle 13 and Unit 1 Cycles starting with Cycle 15 use the method of analysis and models described in References 5.2-14 and 5.2-15. Safety/relief valves are simulated in the models.

In addition to determining the pressure in the pressure vessel, the model determines the pressure in the following components that comprise the reactor coolant pressure boundary.

	Design Pressure psig	Maximum Pressure psig
Recirculation Suction Pipe	1250	1375
Main Steam Piping	1250	1375
Recirculation Discharge Piping	1500	1650

The pressure at the bottom of the pressure vessel is explicitly calculated by the model as are the pressures in the other components listed above.

The safety/relief valve characteristic as modeled is shown in Figure 5.2-2 for the spring mode of operation. Typical valve characteristics are reflected in Figure 5.2-2A. The associated bypass, turbine control valve, and main steam isolation valve characteristics are also simulated in the model.

Closure time of the MSIVs is conservatively assumed to be less than or equal to the minimum closure time given in the Technical Specifications.

5.2.2.2.2 System Design

Reload specific evaluations are conducted to determine the required steam flow capacity of the safety/relief valves based on the following assumptions:

5.2.2.2.2.1 Operating Conditions

(1) operating power see Table 5.2-9

(2) vessel dome pressure $\leq$ 1050 psig, and

(3) core coolant flow = 108 million lbs/hr.

These conditions are the most severe because maximum stored energy exists at these conditions. At lower power conditions the transients would be less severe.

5.2.2.2.2.2 Transients

The overpressure protection system must accommodate the most severe pressurization transient. There are two major transients, the closure of all main steamline isolation valves and a turbine/generator trip with a coincident failure of the turbine steam bypass system valves that represent the most severe abnormal operational transient resulting in a nuclear system pressure rise. The evaluation of transient behavior with final plant configuration has shown that the isolation valve closure is slightly more severe when credit is taken only for indirect derived scrams. Therefore, it is used as the overpressure protection basis event. Typical results for Units 1 and 2 are shown in Figures 5.2-13 and 5.2-14 for Units 1 and 2, respectively. Cycle specific results may be found, by reference, in Chapter 15C and Chapter 15D for Unit 1 and Unit 2, respectively. The peak pressures are determined for each of the components listed in Section 5.2.2.2.1 and the minimum margin to their respective design limits can also be determined. Calculated pressures are all within the respective acceptance criteria of 110% of the design pressure for the reactor pressure vessel and the reactor pressure boundary components. Table 5.2-9 lists the sequence of

events of the various systems assumed to operate during the main steam line isolation closure with high neutron flux scram event for Units 1 and 2.

5.2.2.2.2.3 Scram

The scram times assumed for the overpressure protection analysis are based on the maximum allowable values given in the Technical Specifications.

5.2.2.2.2.4 Safety/Relief Valve Transient Analysis Specifications

(1) valve groups: spring-action safety mode - 3 groups

(2) pressure setpoints: see Table 5.2-2

The setpoints are assumed at a conservatively high level above the nominal setpoints. This is to account for initial setpoint errors and any instrument setpoint drift that might occur during operation. The assumed setpoints in the analysis are 3% above the actual nominal setpoints. Conservative safety/relief valve response characteristics as shown in figure 5.2-6 are assumed.

For the analysis, the safety valves that were assumed to be out of service were those that had the lowest pressure setpoints. The assumed minimum number of operable S/RVs is in accordance with the Technical Specifications.

5.2.2.2.2.5 Safety Valve Capacity

Sizing of the safety valve capacity and the number of valves allowed to be out-of-service was based on assuring that the peak vessel pressure is less than the vessel code limit (1375 psig) in response to the reference transients Subsection 5.2.2.2.2.2. In addition, the analyses that are performed under Subsection 5.2.2.2.2.2 are also used to confirm that the capacity of the safety valves is adequate to assure that the component peak pressures during the transient are less than the limits listed in Subsection 5.2.2.2.1.

5.2.2.2.3 Evaluation of Results

5.2.2.2.3.1 Safety Valve Capacity

The required safety valve capacity is determined by analyzing the pressure rise from a MSIV closure with flux scram transient. The plant is assumed to be operating at the turbine-generator design conditions at a maximum vessel dome pressure equal to the maximum dome pressure allowed by Technical Specifications. The reactor power assumed is given in Table 5.2-9. The analysis hypothetically assumes the failure of the direct MSIV position scram. The reactor is shut down by the backup, indirect, high neutron flux scram. The analysis indicates that the design valve capacity is capable of maintaining adequate margin below the peak ASME code allowable pressures in the reactor vessel and associated components as described above. Figure 5.2-13 shows typical curves produced by these analyses.

Under the General Requirements for Protection Against Overpressure as given in Section III of the ASME Boiler and Pressure Vessel Code, credit can be allowed for a scram from the reactor protection system. In addition, credit can also be taken for the protective circuits which are indirectly derived when determining the required safety valve capacity. However, only the backup

reactor high neutron flux scram is conservatively applied as a design basis in determining the required capacity of the pressure relieving safety valves. Application of the direct position scrams in the design basis could be used since they qualify as acceptable pressure protection devices when determining the required safety valve capacity of nuclear vessels under the provisions of the ASME code.

5.2.2.2.3.2 Pressure Drop in Inlet and Discharge

Pressure drop on the piping from the reactor vessel to the valves is taken into account in calculating the maximum vessel pressures. Pressure drop in the discharge piping to the suppression pool is limited by proper discharge line sizing to prevent backpressure on each safety/relief valve from exceeding 40% of the valve inlet pressure, thus assuring choked flow in the valve orifice and no reduction of valve capacity due to the discharge piping, (Reference 5.2-7). Each safety/relief valve has its own separate discharge line.

5.2.2.3 Piping & Instrument Diagrams

Dwgs. M-141, Sh. 1 and M-141, Sh. 2 are the P&ID for the Nuclear Boiler System including pressure-relieving devices.

5.2.2.4 Equipment and Component Description

5.2.2.4.1 Description

The nuclear pressure relief system consists of safety/relief valves located on the main steam lines between the reactor vessel and the first isolation valve within the drywell. These valves protect against over-pressure of the nuclear system.

The safety/relief valves provide three main protection functions:

(1) Overpressure relief operation. The valves open automatically to limit a pressure rise.

(2) Overpressure safety operation. The valves function as safety valves and open (self-actuated operation if not already automatically opened for relief operation) to prevent nuclear system overpressurization.

(3) Depressurization operation. The ADS valves open automatically as part of the emergency core cooling system (ECCS) for events involving small breaks in the nuclear system process barrier. The location and number of the ADS valves can be determined from Figure 5.1-2.

Chapter 15 discusses the events which are expected to activate the primary system safety/relief valves. The chapter also summarizes the number of valves expected to operate during the initial blowdown of the valves and the expected duration of this first blowdown. For several of the events it is expected that the lowest set safety/relief valve will reopen and reclose as generated heat drops into the decay heat characteristics. The pressure increase and relief cycle will continue with lower frequency and shorter relief discharges as the decay heat drops off and until such time as the RHR system can dissipate this heat. The duration of each relief discharge should in most cases be less than 30 seconds. Remote manual actuation of the valves from the

control room is recommended to minimize the total number of these discharges, with the intent of achieving extended valve seat life.

A schematic of the safety/relief valve is shown in Figure 5.2-7. It is opened by either of two modes of operation:

(1) The spring mode of operation which consists of direct action of the steam pressure against a spring-loaded disk that will pop open when the valve inlet pressure force exceeds the spring force. Figure 5.2-6 diagrams the valve lift vs time characteristic.

(2) The power actuated mode of operation which consists of using an auxiliary actuating device consisting of a pneumatic piston/cylinder and mechanical linkage assembly which opens the valve by overcoming the spring force, even with valve inlet pressure equal to zero psig.

The pneumatic operator is so arranged that if it malfunctions it will not prevent the valve disk from lifting if steam inlet pressure reaches the spring lift set pressure.

For overpressure safety/relief valve operation (self-actuated or spring lift mode), the spring load establishes the safety valve opening setpoint pressure and is set to open at setpoints designated in Table 5.2-2. The ASME code requires that full lift of this mode of operation should be attained at a pressure no greater than 3% above the setpoint.

To prevent backpressure, which results in the discharge line when the valve is open and discharging steam, from causing valve cycling and chatter with resulting set pressure variances, each valve contains an internal part/feature that has been factory adjusted by test to provide for proper valve blowdown/reclosure. The factory blowdown adjustment is compatible to the expected plant specific backpressure range to be realized under normal operating conditions.

The safety function of the safety/relief valve is a backup to the relief function described below. The spring-loaded valves are designed in accordance with ASME III, NB 7640 as safety valves with auxiliary actuating devices and manufactured in accordance with ASME Section III Class I component requirements.

For overpressure relief valve operation (power actuated mode), each valve is provided with a pressure sensing device which operates at the setpoints designated in Chapter 15. When the set pressure is reached, it operates a solenoid valve which in turn actuates the pneumatic piston/cylinder and linkage assembly to open the valve.

When the piston is actuated, the delay time, maximum elapsed time between receiving the overpressure signal at the valve actuator and the actual start of valve motion, will not exceed 0.1 seconds. The maximum full stroke opening time will not exceed 0.15 seconds.

The safety/relief valves can be operated in the power actuated mode by remote-manual controls from the main control room.

Each safety/relief valve is provided with its own pneumatic accumulator and inlet check valve. The accumulator capacity is sufficient to provide one safety/relief valve actuation, which is all that is required for overpressure protection. Subsequent actuations for an overpressure event can be spring actuations to limit reactor pressure to acceptable levels.

The safety/relief valves are designed to operate to the extent required for overpressure protection for the accident environment referenced in section 3.11.

The Automatic Depressurization System (ADS) utilizes selected safety/relief valves for depressurization of the reactor (See Section 7.3). Each of the safety/relief valves utilized for automatic depressurization is equipped with an air accumulator and check valve arrangement. These accumulators assure that the valves can be held open following failure of the air supply to the accumulators. They are sized to be capable of opening the valves and holding them open against a peak calculated drywell pressure of 48.6 psig with the reactor completely depressurized. The accumulator capacity is sufficient for each ADS valve to provide two actuations against 34.0 psig, which represents 70% of the peak calculated drywell pressure.

Each safety/relief valve discharges steam through a discharge line to a point below the minimum water level in the suppression pool. Safety relief valve discharge line piping from the safety relief valve to the suppression pool consists of two parts. The first part is attached at one end to the safety relief valve and attached at its other end to the containment diaphragm slab through a pipe anchor. The main steam piping, including this portion of the safety relief valve discharge piping, is analyzed as a complete system. This portion of the safety relief valve discharge lines is therefore classified as quality group C and Seismic Category I.

The second part of the safety relief valve discharge piping extends from the upstream anchor to the suppression pool. Because of the upstream anchor on this part of the line, it is physically decoupled from the main steam header and is therefore analyzed as a separate piping system. In analyzing this part of the discharge piping in accordance with the requirements of quality Group C and Seismic Category I, the following load combination will be considered as a minimum:

Pressure and temperature
Dead weight
Fluid dynamic loads due to S/R valve operation
Anchor relative seismic (SSE) movement

Movement of the safety relief valve discharge line will be monitored as a part of the preoperational and startup testing of the main steam lines, in accordance with the requirements of Chapter 14.

The safety/relief valve discharge piping is designed to limit valve outlet pressure to 40% of maximum valve inlet pressure with the valve wide open. Water in the line more than a few feet above suppression pool water level would cause excessive pressure at the valve discharge when the valve is again opened. For this reason, a vacuum relief valve is provided on each safety/relief valve discharge line to prevent drawing an excessive amount of water up into the line as a result of steam condensation following termination of relief operation. The safety/relief valves are located on the main steam line piping, rather than on the reactor vessel top head, primarily to simplify the discharge piping to the pool and to avoid the necessity of having to remove sections of this piping when the reactor head is removed for refueling. In addition, valves located on the steam lines are more accessible during a shutdown for valve maintenance.

The nuclear pressure relief system automatically depressurizes the nuclear system sufficiently to permit the LPCI or CS systems to operate as a backup for the HPCI system. Further descriptions of the operation of the automatic depressurization feature are found in Section 6.3, and in Subsection 7.3.1.1.1.

5.2.2.4.2 Design Parameters

Table 5.2-3 lists design temperature, pressure, and maximum test pressure for the RCPB components. The specified operating transients for components within the RCPB are given in Table 3.9-4. Refer to Section 3.7 for discussion of the input criteria for design of Seismic Category I structures, systems, and components.

The design requirements established to protect the principal components of the reactor coolant system against environmental effects are discussed in Section 3.11.

5.2.2.4.2.1 Safety/Relief Valve

The discharge area of the valve is 16.117 square inches and the coefficient of discharge K is equal to 0.966. The diameter and length of the discharge pipe from each valve to the discharge device in the suppression pool is defined in the Design Assessment Report (DAR), Table 1.3-2. The discharge pipe routing within the suppression chamber is shown in the DAR, Figures 1.3-2, 1.3-3 and 1.3-4. The design pressure and temperature of the valve inlet and outlet are 1250 psig @ 575°F and 550 psig @ 500°F, respectively.

Cyclic testing has demonstrated that the valves are capable of at least 60 actuation cycles between required maintenance.

See Figure 5.2-7 for a schematic cross section of the valve.

5.2.2.5 Mounting of Pressure Relief Devices

The pressure relief devices are located on the main steam piping header. The mounting consists of a special, contour nozzle and an over-sized flange connection. This provides a high integrity connection that accounts for the thrust, bending and torsional loadings which the main steam pipe and relief valve discharge pipe are subjected to. This includes:

(1) The thermal expansion effects of the connecting piping.

(2) The dynamic effects of the piping due to SSE.

(3) The reactions due to transient unbalanced wave forces exerted on the safety/relief valves during the first few seconds after the valve is opened and prior to the time steady-state flow has been established. (With steady-state flow, the dynamic flow reaction forces will be self-equilibrated by the valve discharge piping).

(4) The dynamic effects of the piping and branch connection due to the turbine stop valve closure.

In no case will allowable valve flange loads be exceeded nor will the stress at any point in the piping exceed code allowables for any specified combination of loads. The design criteria and analysis methods for considering loads due to SRV discharge is contained in Subsection 3.9.3.3.

5.2.2.6 Applicable Codes and Classification

The vessel overpressure protection system is designed to satisfy the requirements of Section III, Nuclear Vessels, of the ASME Boiler and Pressure Vessel Code. The general requirements for protection against overpressure as given in Article 9 of Section III of the Code recognize that reactor vessel overpressure protection is one function of the reactor protective systems and allows the integration of pressure relief devices with the protective systems of the nuclear reactor. Hence, credit is taken for the scram protective system as a complementary pressure protection device. The NRC has also adopted the ASME Codes as part of their requirements in the Code of Federal Regulations (10CFR50.55A).

5.2.2.7 Material Specification

Pressure retaining components of valves in Quality Group A are constructed only from ASME designated materials.

5.2.2.8 Process Instrumentation

Overpressure protection process instrumentation is shown on Figure 5.1-2.

5.2.2.9 System Reliability

This system is designed to satisfy the requirements of Section III of the ASME Boiler and Pressure Vessel code, therefore, it has high reliability. The consequences of failure are discussed in Subsections 15.1.4 and 15.6.1.

5.2.2.10 Inspection and Testing

The Main Steam Relief Valves were installed after certification from the valve manufacturer that design and performance requirements were met. This includes capacity and blowdown requirements. The set points are adjusted, verified, and indicated on the valves by the manufacturer. Specified manual and automatic actuation relief mode of each safety/relief valve is verified during the preoperational test program. Valve operability is verified during the preoperational test program in accordance with the requirements of Chapter 14.

The valves are mounted on 1500-lb primary service rating flanges. They can be removed for maintenance or bench checks and reinstalled during normal plant shutdowns.

The valves are tested in accordance with the ASME Code, requirements and, the approved request.

5.2.3 REACTOR COOLANT PRESSURE BOUNDARY MATERIALS

5.2.3.1 Material Specifications

Table 5.2-4 lists the principal pressure retaining materials and the appropriate material specifications for the reactor coolant pressure boundary components.

5.2.3.2 Compatibility with Reactor Coolant

5.2.3.2.1 PWR Chemistry of Reactor Coolant

Not applicable to BWRs.

5.2.3.2.2 BWR Chemistry of Reactor Coolant

The SSES Reactor Coolant System (RCS) Chemistry program is consistent with the EPRI BWR Water Chemistry Guidelines. The coolant chemistry requirements discussed in this subsection remain consistent with the requirements of Regulatory Guide 1.56 (6/73). The EPRI BWR Water Chemistry Guidelines are periodically revised by industry expert panels. Incorporated parameter limits are as or more conservative than those found in the original plant licensing documents. The EPRI BWR Water Chemistry Guideline document has been developed and routinely revised to reflect industry experience and research and have been shown to be effective over time with their widespread use as a water control document.

Materials in the primary system are primarily austenitic stainless steel and Zircaloy cladding. The reactor water chemistry limits are established to provide an environment favorable to these materials. Limits are placed on conductivity and chloride concentrations. Conductivity is limited because it can be continuously and reliably measured and gives an indication of abnormal conditions and the presence of impurities in the coolant. Chloride limits are specified to prevent stress corrosion cracking of stainless steel. For further information, see Reference 5.2-2. Reference Subsection 9.5.9 for a description of the Hydrogen Water Chemistry System.

Several investigations have shown that in neutral solutions some oxygen is required to cause stress corrosion cracking of stainless steel, while in the absence of oxygen no cracking occurs. One of these is the chloride-oxygen relationship of Williams (Reference 5.2-3), where it is shown that at high chloride concentration little oxygen is required to cause stress corrosion cracking of stainless steel, and at high oxygen concentration little chloride is required to cause cracking.

When BWR RCS conductivity is in its normal range, pH, chloride and other impurities affecting conductivity will also be within their normal range. When conductivity becomes abnormal, sampling and analysis measurements are made to determine whether or not EPRI recommended sample parameters are also out of their normal operating values. Conductivity could be high due to the presence of corrosive benign ions such as chromate, iron or zinc that would not have an adverse effect on pH. In such a case, high conductivity alone is not a cause for shutdown or other corrective action. In some types of water-cooled reactors, conductivities are high because of the purposeful use of additives. In BWRs, additives may be used and where near neutral pH is maintained, conductivity provides a good and prompt measure of the quality of the reactor water. Significant changes in conductivity provide the operator a warning so he can investigate and remedy the condition before reactor water limits are reached.

Methods available to the operator for correcting the off-standard condition include operation of the reactor water cleanup system and reduction of impurity source term(s). In some circumstances of abnormal chemistry, risk decisions may dictate it desirable to maintain power operation under HWC (maintain mitigation) during the transient and allow the impurity to cleanup. In other cases, it may be prudent to place the reactor in the cold shutdown condition. The major benefit of cold shutdown is to reduce the temperature dependent corrosion rates and provide time for the cleanup system to reestablish the purity of the reactor coolant.

The following is a summary and description of BWR water chemistry for various plant conditions.

(1) Normal Plant Operation

The SSES BWR Water Chemistry control program is continually improving based on the EPRI BWR Water Chemistry Guideline reviews and revisions. SSES adheres to the Guideline Control Parameter limits, unless justification for deviation is forwarded to the BWRVIP in accordance with approved BWRVIP reporting guidance.

For normal operation starting with the condenser-hotwell, condensate water is processed through a condensate treatment system. This process consists of condensate filtration to remove iron, followed by demineralization in the resin polisher system. When condensate system polisher resin becomes depleted, it is discarded and replaced. .

The effluent from the condensate treatment system is pumped through the feedwater heater train, and enters the reactor vessel at an elevated temperature. Feedwater (FW) zinc and iron injection are used as needed to control radiation field buildup on ex-core surfaces. FW hydrogen injection is used to mitigate reactor vessel and internals stress corrosion cracking. Approved modifications to supplemented HWC injection with noble metal injection may be implemented as necessary. Condensate system oxygen injection may be used to control balance of plant Flow Assisted Corrosion rates in the event air in leakage is not sufficient to maintain dissolved oxygen levels.

A reactor water cleanup system is provided for removal of impurities in the primary system. The cleanup process consists of filtration and ion exchange, and serves to maintain a high level of water purity in the reactor coolant.

Additional water input to the reactor vessel originates from the Control Rod Drive (CRD) cooling water. The CRD water is essentially feedwater quality. Separate filtration for purification and removal of insoluble corrosion products takes place within the CRD system prior to entering the drive mechanisms and reactor vessel.

No other inputs of water or sources of oxygen are routinely present during normal plant operation. During off normal plant conditions, additional inputs may result as outlined in the following section.

(2) Plant Conditions Outside Normal Operation

During periods of plant conditions other than normal power production, transients may take place, particularly with regards to the oxygen levels in the primary coolant.

Systems other than the reactor are not affected significantly enough to cause any impact on primary system components or subsequent operation. In essence, depending on what the plant condition is, i.e., hot standby with/without reactor vessel venting or plant shutdown, the hotwell condensate will absorb oxygen from the air when vacuum is broken in the condenser. Prior to startup and input of feedwater to the reactor, vacuum is established in the condenser and deaeration of the condensate takes place by means of mechanical vacuum pump and steam jet air ejector (SJAE) operation and condensate recirculation. During these plant conditions, continuous input of control rod drive (CRD) cooling water takes place as described previously.

a) Plant Depressurized and Reactor Vented

During certain periods such as during refueling and maintenance outages, the reactor is vented to the condenser or atmosphere. Under these circumstances the reactor cools and the oxygen concentration increases to a maximum value of about 8 ppm. Equilibrium between the atmosphere above the reactor water surface, the CRD cooling water input, any residual radiolytic effects, and the bulk reactor water will be established after some time. The specific conductivity of reactor water may increase to values of approximately 1 μS/cm due to the absorption of atmospheric carbon dioxide that is normally present in the atmosphere. No other changes in water chemistry of significance take place during this plant condition because no appreciable inputs take place.

b) Plant Transient Conditions - Plant Startup/Shutdown

During these conditions, no significant changes in water chemistry other than oxygen concentration take place.

(i) Plant Startup

Depending on the duration of the plant shutdown prior to startup and whether the reactor has been vented, the oxygen concentration could be that of air saturated water, i.e., ≈ 8 ppm oxygen.

Following nuclear heatup initiation, the oxygen level in the reactor water will decrease rapidly as a function of water temperature increase and corresponding oxygen solubility in water. The oxygen level will reach a minimum of about 20 ppb (0.02 ppm) at a coolant temperature of about 380°F, at which point an increase will take place due to significant radiolytic oxygen generation. For the elapsed process up to this point the oxygen is degassed from the water and is displaced to the steam dome above the water surface.

Further increase in power increases the oxygen generation as well as the temperature. The solubility of oxygen in the reactor water at the prevailing temperature controls the oxygen level in the coolant until rated temperature (540°F) is reached. Thus, a gradual increase from the minimum level of 20 ppb to a maximum value of about 200 ppb oxygen takes place. At, and after this point (540°F) steaming and the radiolytic process control the coolant oxygen concentration to a level of around 200

ppb. When in service, HWC injection will reduce the reactor coolant dissolved oxygen concentration.

(ii) Plant Shutdown

Upon plant shutdown following power operation, the radiolytic oxygen generation essentially ceases as the fission process is terminated. Because oxygen is no longer generated, while some steaming still will take place due to residual energy, the oxygen concentration in the coolant will decrease to a minimum value determined by steaming rate temperature. If venting is performed, a gradual increase to essentially oxygen saturation at the coolant temperature will take place.

(iii) Oxygen in Piping and Parts Other Than the Reactor Vessel Proper

As can be concluded from the preceding descriptions, the maximum possible oxygen concentration in the reactor coolant and therefore in contact with the reactor or any other directly related or associated parts is that of air saturation at ambient temperature. In the water phase, dissolved oxygen levels will be a nominal value of 8 ppm. As temperature is increased and hence, oxygen solubility decreased accordingly, the oxygen concentration will be maintained at this maximum value, or reduced below it depending on available removal mechanisms, i.e., diffusion, steam stripping, flow transfer or degassing.

Depending on the location, configuration, etc., such as dead legs or stagnant water, water inventory dissolved oxygen concentration may vary.

Primary coolant conductivity is continuously monitored with an instrument connected to redundant sources which include the reactor water recirculation loop and the reactor water cleanup system inlet. The effluent from the reactor water cleanup system is also monitored for conductivity on a continuous basis. These measurements provide adequate surveillance of the reactor coolant.

Grab sample capability is provided, for the locations shown on Table 5.2-7, for special and non-continuous measurements such as pH, oxygen, chloride and radiochemical measurements.

The relationship of chloride concentration to specific conductance measured at 25ºC for chloride compounds such as sodium chloride and hydrochloric acid can be calculated, see Figure 5.2-9. Values for these compounds essentially bracket values of other common chloride salts or mixtures at the same chloride concentration. Surveillance requirements are based on these relationships.

In addition to the reactor water chemistry program, limits, monitoring and sampling requirements are imposed on the condensate, condensate treatment system and feedwater by EPRI Water Chemistry Guideline

specifications. Thus, a total plant water quality surveillance program is established providing assurance that off specification conditions will quickly be detected and corrected.

The sampling frequency when reactor water has a low specific conductance is adequate for routine monitoring. When specific conductance increases, and higher chloride concentrations are possible, or when continuous conductivity monitoring is unavailable, increased sampling, analysis and monitoring is performed.

For the higher than normal limits of < 1μS/cm, more frequent sampling and analyses are invoked by the coolant chemistry surveillance program.

c. Water Purity During a Condenser Leakage

The condensate cleanup system was originally designed to maintain the reactor water chloride concentration below 200 ppb during a condenser tube leak of 23 gallons per minute indefinitely. The condensate cleanup system was originally designed to sustain an effluent conductivity of 0.15 micromho with a 46 gpm condenser leak when the circulating water contains 1000 ppm of TDS. Refer to Subsection 10.4.6.

To protect against a major condenser tube leak, sufficient instrumentation is provided to maintain a reserve of 50 percent of the theoretical ion exchange capacity during normal operation per Regulatory Guide 1.56.

5.2.3.2.3 Compatibility of Construction Materials with Reactor Coolant

The materials of construction exposed to the reactor coolant consist of the following:

(1) Solution annealed austenitic stainless steels (both wrought and cast) Types 304, 304L, 316 and 316L.

(2) Nickel base alloys - Inconel 600 and Inconel 750X.

(3) Carbon steel and low alloy steel.

(4) Some 400 series martensitic stainless steel (all tempered at a minimum of 1100ºF).

(5) Colmonoy and Stellite hardfacing material.

All of these materials of construction are resistant to stress corrosion in the BWR coolant. General corrosion on all materials, except carbon and low alloy steel, is negligible. Conservative corrosion allowances are provided for all exposed surfaces of carbon and low alloy steels.

5.2.3.2.4 Compatibility of Construction Materials with External Insulation and Reactor Coolant

The materials of construction exposed to external insulation are:

(1) Solution annealed austentitic stainless steels. Types 304, 304L and 316.

(2) Carbon and low alloy steel.

(3) Nickel alloy and austenitic stainless weld metal.

Two types of external insulation are employed on BWRs. Reflective metal insulation used does not contribute to any surface contamination and has no effect on construction materials. Nonmetallic insulation used on stainless steel piping and components complies with the requirements of either of the following industry standards:

(1) ASTM C692-71, Standard Methods for Evaluating Stress Corrosion Effects of Wicking Type Thermal Insulation on Stainless Steel (Dana Test).

(2) RDT-M12-1T, Test Requirements for Thermal Insulating Materials for Use on Austenitic Stainless Steel, Section 5 (KAPL Test).

Chemical analyses are required to verify that the leachable sodium, silicate, and chloride are within acceptable levels. Insulation is packaged in waterproof containers to avoid damage or contamination during shipment and storage.

Since there are no additives in the BWR coolant, leakage would expose materials to high purity, demineralized water. Exposure to demineralized water would cause no detrimental effects.

5.2.3.3 Fabrication and Processing of Ferritic Materials

5.2.3.3.1 Fracture Toughness

Fracture toughness requirements for the ferritic materials used for pumps, piping, and valves of the reactor coolant pressure boundary are as follows:

The pump components except for the bolting, are austenitic stainless steel. The bolting meets Section III of ASME B&PV Code, Summer 1971 Addenda which requires impact testing to be performed at 10ºF.

Safety/Relief Valves were exempted from fracture toughness requirements because Section III of the 1971 ASME Boiler and Pressure Vessel Code did not require impact testing on valves with inlet connections of 6 inches or less nominal pipe size.

Main Steam Isolation Valves were also exempted because the Code existing at the time of the purchase, ASME Section III Summer 1971 Addenda did not require brittle fracture testing on ferritic pressure boundary components when the system temperature was in excess of 250ºF at 20% of the design pressure.

Main Steam Piping was tested in accordance with and met the fracture toughness requirements of paragraph NB-2300 of the 1972 Summer Addenda to ASME Code, Section III, the applicable code at the time of the purchase order.

5.2.3.3.1.1 Compliance with Code Requirements

The ferritic pressure boundary material of the reactor pressure vessel was qualified by impact testing in accordance with the 1968 Edition of Section III ASME Code and Addenda to and including the Summer 1970 Addenda. From an operational standpoint, this Code would require that for any significant pressurization (taken to be more than 20% of Code hydrostatic test pressure = 312 psig) the minimum metal temperature of all vessel shell and head material be 100°F (NDTT +60°F).

5.2.3.3.1.2 Acceptable Fracture Energy Levels

Operating limits on reactor vessel pressure and temperature during normal heatup and cooldown, and during inservice hydrostatic testing, were established using as a guide Appendix G, Summer 1972 Addenda, of Section III of the ASME Boiler and Pressure Vessel Code, 1971 Edition.

These operating limits will assure that a large postulated surface flaw, having a depth of one-quarter of the material thickness, can be safely accommodated in regions of the vessel shell remote from discontinuities. In addition the specific additional margins required by 10CFR50, Appendix G, paragraph IV.A.2.c are included in the operating limits for core operations.

For the purpose of setting these operating limits, the reference temperature, RT_{NDT}, was determined from the impact test data taken in accordance with requirements of the Code to which this vessel is designed and manufactured. The dropweight NDT temperature was used as the reference temperature.

The highest reference temperature of any part of the reactor pressure vessel pressure boundary material was used as the reference temperature for calculating one set of operating temperature and pressure limits for the shell remote from the core beltline region. A second set of temperature and pressure limits for the core beltline region was calculated based on the core beltline region material reference temperature.

The requirements of the Code to which the vessel was designed and manufactured results in a third set of vessel shell temperature pressure limits; namely, NDTT +60°F or CVN +60°F at pressure greater than 20% of preoperational system hydrostatic test pressure. The more conservative of the above three limits was used to set pressure and temperature limits for the vessel shell.

5.2.3.3.1.3 Operating Limits During Heatup, Cooldown, and Core Operation

Since 100°F/hour is the maximum average normal heatup or cooldown rate for which the reactor vessel is designed, a conservative fracture toughness analysis was done for this assumed rate.

The maximum temperature gradient through the wall corresponding to this rate was considered. The results of this analysis are a set of operating limits for non-nuclear heatup or cooldown following nuclear shutdown, and another set for operating limits for operation whenever the core is critical (except for low level physics tests).

5.2.3.3.1.4 Temperature Limits for ISI Hydrostatic or Leak Pressure Tests

The fracture toughness analysis for pressure tests resulted in the curves shown on Figure 5.3-4A, 5.3-4B, 5.3-4C, and 5.3-4D of minimum vessel shell and head temperatures versus vessel pressure as measured in vessel top head. The dashed line curve, beltline region, is based on an assumed initial RT_{NDT} of +10°F, the predicted shift in the RT from Figure 5.3-5 based on neutron fluence at 1/4 of vessel wall thickness must be added to the beltline curve to account for the effect of fast neutrons on the beltline material properties. The curve for areas remote from the beltline (upper curve) is based on an assumed RT_{NDT} of +40°F. The controlling minimum temperature for a desired pressure is then selected as the greater of the solid curve or the dashed curve plus the shift.

5.2.3.3.1.5 Temperature Limits for Boltup

Minimum closure flange and closure stud temperatures of 70°F (NDTT + 60°F) are required whenever the closure studs are under preload or are being tensioned.

5.2.3.3.1.6 Reactor Vessel Annealing

In-place annealing of the reactor vessel because of radiation embrittlement is unnecessary since the predicted value in transition of adjusted reference temperature will not exceed 200°F - see 10CFR50, Appendix G, Paragraph IV.C.

5.2.3.3.2 Control of Welding

5.2.3.3.2.1 Control of Preheat Temperature Employed for Welding of Low Alloy Steel Regulatory Guide 1.50.(Rev. 0)

The use of low alloy steel is restricted to the reactor pressure vessel. Other ferritic components in the reactor coolant pressure boundary are fabricated from carbon steel materials.

Preheat temperatures employed for welding of low alloy steel meet or exceed the recommendations of ASME Section III, Subsection NA. Components were either held for an extended time at preheat temperature to assure removal of hydrogen, or preheat was maintained until post weld heat treatment. The minimum preheat and maximum interpass temperatures were specified and monitored.

All welds were nondestructively examined by radiographic methods. In addition, a supplemental ultrasonic examination was performed.

For repair welding utilizing the ASME Section XI temperbead welding methods, the preheat temperatures and supplemental nondestructive examination shall be in accordance with the temperbead welding rules as provided in Section XI and applicable Code Cases.

5.2.3.3.2.2 Control of Electroslag Weld Properties Regulatory Guide 1.34. (Rev. 0)

No electroslag welding was performed on BWR components.

5.2.3.3.2.3 Welder Qualification for Areas of Limited Accessibility. Regulatory Guide 1.71.(Rev. 0)

For non-NSSS items, refer to response to Regulatory Guide 1.71 in Section 3.13.

There are few restricted access welds involved in the fabrication of NSSS reactor coolant pressure boundary components. Welder qualification for welds with the most restricted access was accomplished by mock-up welding. Mock-ups were examined with radiography or sectioning.

5.2.3.3.3 Nondestructive Examination of Ferritic Tubular Products

For non-NSSS items, refer to response to Regulatory Guide 1.66 in Section 3.13.

Wrought tubular products were supplied in accordance with applicable ASTM/ASME material specifications. These specifications require a hydrostatic test on each length of tubing or pipe.

These components met the requirements of the ASME Codes existing at the time of placement of order which predate Regulatory Guide 1.66 (Rev.0).

5.2.3.4 Fabrication and Processing of Austenitic Stainless Steels

For non-NSSS items, refer to response to Regulatory Guide 1.44 in Section 3.13

5.2.3.4.1 Avoidance of Stress Corrosion Cracking

5.2.3.4.1.1 Avoidance of Significant Sensitization

All austenitic stainless steel was purchased in the solution heat treated condition in accordance with applicable ASME and ASTM specifications. Carbon content was limited to 0.08% maximum, and cooling rates from solution heat treating temperatures were required to be rapid enough to prevent sensitization.

Welding heat input was restricted to 110,000 joules per inch maximum, and interpass temperature to 350^{o}F. High heat welding processes such as block welding and electroslag welding were not permitted. All weld filler metal and castings were required by specification to have a minimum of 5% ferrite.

Whenever any wrought austenitic stainless steel was heated to temperatures over 800^{o}F, by means other than welding or thermal cutting, the material was re-solution heat treated.

These controls were used to avoid severe sensitization. Compliance with Regulatory Guide 1.44 (5/73) is discussed in Section 3.13.

5.2.3.4.1.2 Process Controls to Minimize Exposure to Contaminants

Exposure to contaminants capable of promoting stress corrosion cracking of austenitic stainless steel components was avoided by carefully controlling all cleaning and processing materials which contact the stainless steel during manufacture and construction.

Special care was exercised to insure removal of surface contaminants prior to any heating operations. Water quality for cleaning, rinsing, flushing, and testing was controlled and monitored. Suitable packaging and protection was provided for components to maintain cleanliness during shipping and storage.

The degree of surface cleanliness obtained by these procedures meets the requirements of Regulatory Guides 1.44 (5/73) and 1.37 (3/73).

5.2.3.4.1.3 Cold Worked Austenitic Stainless Steels

Austenitic stainless steels with a yield strength greater than 90,000 psi are not used.

5.2.3.4.2 Control of Welding

For non-NSSS items, refer to response to Regulatory Guide 1.31 in Section 3.13.

5.2.3.4.2.1 Avoidance of Hot Cracking

All austenitic stainless steel filler materials were required by specification to have a minimum of 5% ferrite. This amount of ferrite is considered adequate to prevent hot cracking in austenitic stainless steel welds.

An extensive test program performed by General Electric Company, with the concurrence of the Regulatory Staff, has demonstrated that controlling weld filler metal ferrite at 5% minimum produces production welds which meet the requirements of Regulatory Guide 1.31, (Rev. 1). A total of approximately 400 production welds in five BWR plants were measured and all welds met the requirements of the Interim Regulatory Position to Regulatory Guide 1.31.

5.2.3.4.2.2 Electroslag Welds

Electroslag welding was not employed for reactor coolant pressure boundary components.

5.2.3.4.2.3 Welder Qualification for Areas of Limited Accessibility Regulatory Guide 1.71 (Rev. 0)

For non-NSSS items, refer to response to Regulatory Guide 1.71 in Section 3.13.

There are few restrictive welds involved in the fabrication of NSSS reactor coolant pressure boundary components. Welder qualification for welds with the most restrictive access was accomplished by mock-up welding. Mock-ups were examined with radiography or sectioning.

5.2.3.4.3 Nondestructive Examination of Tubular Products Regulatory Guide 1.66 (Rev. 0)

For non-NSSS items, refer to response to Regulatory Guide 1.66 in Section 3.13.

Wrought tubular products were supplied in accordance with applicable ASTM/ASME material specifications. These specifications require a hydrostatic test on each length of tubing. Additionally, the specification for the tubular product used for CRD housings specified ultrasonic examination to paragraph NB-2550 of ASME Code Section III.

These components met the requirements of ASME Codes existing at time of placement of order.

5.2.4 IN-SERVICE INSPECTION AND TESTING OF REACTOR COOLANT PRESSURE BOUNDARY

The construction permits for the Susquehanna SES were issued in November, 1973. Relating this date to the requirements of 10CFR50.55a(g), the preservice examination program with provisions for design and access should comply, as a minimum, with the 1971 Edition of the ASME B&PV Code Section XI including the Summer 1972 Addenda. The Susquehanna SES preservice examination program will not be conducted to the minimum requirements of 10CFR50.55a(g) but rather to the more current 1974 Edition of Section XI including the Winter 1975 Addenda for the RPV and the Summer 1975 addenda as modified by Appendix III from the Winter 1975 addenda and IWA-2232 from the Summer 1976 addenda for the piping systems to the extent practical within the limitations of design, geometry, and materials of construction of the component. Preservice examination of piping integrally welded supports, category B-K-1, and piping pressure retaining bolting categories B-G-1 and B-G-2, will be in accordance with ASME Section XI, 1977 edition including addenda through Summer 1978.

Throughout the service life of the Susquehanna SES, components and their supports classified as ASME Code Class 1, Class 2 or Class 3, except for components excluded under IWB-1220, IWC-1220, and IWD-1220, and IWF-1230, will meet the requirements, except design and access provisions, set forth in Editions of Section XI of the ASME B&PV Code and Addenda that became effective subsequent to the editions specified above and are incorporated by reference in 10CFR50.55 a(g), and to the extent practical within the limitations of design, geometry, and materials of construction of the component.

The initial in-service examinations conducted during the first 120 months will comply, to the extent practical, with the requirements of the ASME B&PV Code Section XI Edition and Addenda incorporated by reference in 10CFR50.55a (b) on the date 12 months prior to the date of issuance of the operating license, subject to modifications listed by the reference sections.

The in-service examinations conducted during successive 120-month periods throughout the service life of the Susquehanna SES will comply, to the extent practical with the requirements of the ASME B&PV Code Section XI Edition and Addenda incorporated by reference in 10CFR50.55a(b) 12 months prior to the start of the 120 month inspection interval, subject to limitations listed by the reference sections.

Details of the inservice inspection program for the inspection interval are contained in the "Inservice Inspection Program Plan."

This document will be updated, as a minimum, every ten (10) years to reflect program commitments for subsequent ten (10) year intervals.

5.2.4.1 System Boundary Subject to Inspection

The inspection requirements of Section XI of the Code are met for all Class 1 pressure-containing components (and their supports) except for components excluded under IWB-1220 of Section XI. Note that the EPRI Topical Report TR-112657, Rev. B-A methodology, which was supplemented by Code Case N-578-1, will be utilized for implementing the risk-informed inservice inspection program. The risk-informed program scope will be implemented as an alternative to the ASME Section XI examination program for Class 1 Examination Categories B-F and B-J welds in accordance with 10CFR50.55a(a)(3)(I). The risk-informed inservice inspection program has been expanded to include welds in the break exclusion region piping, also referred to as the high energy line break region, which includes several non-class welds that fall within the break exclusion region augmented inspection program. Additional guidance for adaptation of the risk-informed inservice inspection evaluation process to break exclusion region piping is given in EPRI TR-1006937 Rev. 0-A. The system boundary includes all pressure vessels, piping, pumps, and valves that are part of the reactor coolant system, or connected to the reactor coolant system, up to and including:

a) The outermost containment isolation valve in system piping that penetrates the primary reactor containment

b) The second of two valves normally closed during normal reactor operation in system piping that does not penetrate primary reactor containment

c) The reactor coolant system safety and relief valves.

5.2.4.2 Accessibility

The design and arrangement of system components are in accordance with IWA-1500, "Accessibility", of the 1971 Edition of Section XI. Adequate clearances for general access are provided as follows:

a) Sufficient space is provided for personnel and equipment to perform inspections.

b) Provisions are made for the removal and storage of structural members, shielding components, and insulating materials, to permit access to the components being inspected.

c) Provisions are made for hoists and other handling machinery needed to handle items in (b), above.

d) Provisions are made for alternative examinations if structural defects or indications reveal that such examinations are required.

e) Provisions are made for the necessary operations associated with repair or replacement of system components and piping.

Piping systems requiring volumetric ultrasonic inspection are designed so that welds requiring inspection are physically accessible for inspection and ultrasonic equipment. Access is provided by leaving adequate space around pipes at these welds and by removing insulation and shielding as required.

The surfaces of welds requiring ultrasonic examination have been ground and contoured to permit effective use of ultrasonic transducers, and to minimize geometric reflectors that could be misinterpreted as flaws.

Piping systems requiring surface or visual examination are designed to allow access and visibility adequate for performance of such examinations.

Access is provided to reactor vessel components to meet, as a minimum, the examination requirements of ASME Section XI as outlined above.

Because high potential radiation levels in the vicinity of the reactor vessel limit access to the vessel, considerations for meeting ASME Section XI have been incorporated into the plant design as follows:

a) An annular space (8-in. minimum) sufficient to accommodate remotely operated inspection equipment is provided between the reactor vessel shell and the thermal insulation for areas behind the reactor shield wall.

b) Removable sections of thermal insulation and openings in the reactor shield with hinged shield plugs are provided to allow access for remote or manual examination of the reactor vessel nozzle-to-shell, nozzle-to-safe-end, and safe-end-to-pipe welds.

c) Access to full penetration vessel welds, nozzle welds above the reactor shield, and all top head welds is provided by removable, freestanding thermal insulation.

d) Openings in the reactor shield and removable insulation are provided to allow access to the reactor skirt-to-bottom head welds.

e) Openings in the reactor skirt, removable insulation panels, and walk-on grating are provided to allow access to the bottom head welds inside the support skirt.

f) The reactor vessel closure head is stored dry in an accessible area to provide direct access for inspection.

g) Reactor vessel studs, nuts, and washers are removed to dry storage for inspection.

In-service inspection access to other major reactor coolant system components is provided as follows:

a) Working platforms are provided to facilitate access to inspection areas.

b) The insulation covering component and piping welds and adjacent base metal is designed for easy removal and reinstallation in areas where inspection is required.

c) The physical arrangement of pipe, pumps, valves, and other components allows personnel access to welds requiring in-service inspection in accordance with ASME Section XI.

5.2.4.3 Examination Techniques and Procedures

The methods, techniques, and procedures used in the Susquehanna SES in-service inspection program comply with the requirements of ASME Section XI,. Subarticle IWA-2200

The visual, surface, and volumetric examination techniques are in compliance with IWA-2210, 2220, and 2230, respectively. In accordance with IWA-2240, if any alternative examination methods, combination of methods, or newly developed techniques are substituted for the above-described methods, results will be provided that demonstrate that the alternative methods are equivalent to or superior to those methods specified in Section XI.

If, as a result of the preservice or in-service examinations, flaw indications are found to have developed and/or propagated beyond the acceptance standards of IWB-3000, then further examinations will be conducted, as needed, to determine the exact condition. Following evaluation of this evidence, a decision will be made regarding repair requirements related to plant safety. Any repairs, if needed, will be performed to the rules of IWB-4000.

5.2.4.4 Inspection Intervals

In-service inspections will primarily be performed during plant outages such as refueling shutdowns or maintenance shutdowns. With the exception of the examinations that may be deferred until the end of the inspection interval, the required examinations will be completed in accordance with Table IWB-2412-1 (Inspection Program B).

A combination of manual and mechanized techniques will be used for in-service examinations. Preservice (or baseline) data will be generated accordingly using the methods/techniques similar to those which will be used for in-service examinations.

Specific details of the inservice inspection program are contained in the "Inservice Inspection Program Plan.”

5.2.4.5 Evaluation of Examination Results

a) Examination evaluation shall be performed in accordance with the requirements of Section XI, IWB-3000, "Acceptance Standards” Acceptance of components for continued service shall be in accordance with IWB-3122 IWB-3123, and IWB-3124.

b) The program regarding repairs of unacceptable indications or replacement of components containing unacceptable indications is in accordance with the requirements of Section XI, IWA-4000, "Repair/Replacement Activities."

5.2.4.6 System Leakage and Hydrostatic Pressure Tests

The pressure retaining Code Class 1 component leakage and hydrostatic pressure test program is in accordance with the requirements of Section XI, IWB-5000, "System Pressure Tests".

5.2.4.7 Augmented Inservice Inspection To Protect Against Postulated Piping Failures

The augmented inservice inspection program to provide 100 percent volumetric examination of circumferential and longitudinal pipe welds in high energy systems between containment isolation valves will be reviewed and implemented as described in Subsection 6.6.8. Commencing with the Third Ten Year Inspection Interval, the risk-informed break exclusion region program methodology, described in EPRI TR-1006937, Rev. 0-A, will be used to define the inspection scope in lieu of the 100% examination of all piping welds in the previous break exclusion region augmented program. Therefore, all welds in the original augmented program for the break exclusion region will be evaluated under the risk-informed inservice inspection program using an integrated risk-informed approach.

5.2.5 DETECTION OF LEAKAGE THROUGH REACTOR COOLANT PRESSURE BOUNDARY

5.2.5.1 Leakage Detection Methods

The nuclear boiler leak detection system consists of temperature, pressure, and flow sensors with associated instrumentation and alarms. This system detects, annunciates, and isolates (in certain cases) leakages in the following systems:

(1) Main steam lines

(2) Reactor water cleanup (RWCU) system

(3) Residual heat removal (RHR) system

(4) Reactor core isolation cooling (RCIC) system

(5) Feedwater system

(6) High Pressure Coolant Injection (HPCI) System

Isolation and/or alarm of affected systems and the detection methods used are summarized in Table 5.2-8.

Small leaks into the drywell (5 gpm and less) are detected by temperature and pressure changes and floor drain sump levels. Large leaks are also detected by changes in reactor water level and changes in flow rates in process lines.

The 5 gpm leakage rate is a technical specification limit on unidentified leakage into the drywell. The containment floor drain sump collection system, which is part of the leak detection system, is capable of monitoring flowrates with an accuracy of one gpm. Thus, the SSES design is in compliance with Paragraph C.2 of Regulatory Guide 1.45.

5.2.5.1.1 Detection of Abnormal Leakage Within the Primary Containment (NSS-Systems)

Abnormal leakage may result in a decrease of reactor water level. Low reactor water level will cause isolation of the RHR shutdown cooling suction line (level 3), the RWCU suction line (level 2) and the main steam lines (level 1). Reactor water level monitoring is also described in FSAR Section 7.3.1.1a.2.4.1.1.

The RCIC and HPCI steam lines are monitored for large leaks inside containment by pressure switches installed on each leg of the elbow tap flow elements. Low pressure will close the isolation valves in the respective line and initiate an alarm in the control room. Closing the isolation valves will not isolate the leak, but the alarm will alert the operators. This leakage will also be detected by the drywell leak detection systems described in Section 5.2.5.1.2. Leakage downstream of the elbow tap flow elements (inside or outside containment) will be detected by high flow switches, causing an alarm and closing the isolation valves. RCIC and HPCI leak detection is also discussed in FSAR Sections 7.6.1a.4.3.3 and 7.6.1a.4.3.8.

Abnormal leakage from the core spray discharge line will result in a pressure differential between the discharge line and the vessel shroud. High differential pressure will initiate an alarm in the control room in the event of a leak.

5.2.5.1.2 Detection of Abnormal Leakage Within the Primary Containment (Non-NSSS)

Leakage through the reactor coolant pressure boundary within the primary containment is detected by monitoring temperatures, pressures, airborne gaseous and particulate radioactivity, and changes of levels in the floor drain sumps. These monitors and their respective locations are listed in Table 5.2-14.

The following systems are used to monitor these variables:

a) Primary containment and suppression pool temperature monitoring system.

b) Primary containment and suppression chamber pressure monitoring system

c) Primary containment atmosphere monitoring system (containment radiation detection)

d) Drywell floor drain sump level monitoring and drywell equipment drain tank level monitoring system.

The above-mentioned leak detection systems are designed in accordance with recommendations of Regulatory Guide 1.45 except as noted in Subsection 5.2.5.1.2.4.6.

The drywell leak detection system is not intended to be qualified as a post LOCA system; it is designed for use during power operation as implied by the Technical Specifications. There would be no practical way of recalibrating the system after the LOCA transient.

For the post accident condition, separate monitoring systems are provided for both the primary containment drywell and suppression chamber pressure and the primary containment and suppression pool temperatures. Details of the design are discussed in Subsections 7.6.1b.1.1.2, 7.6.1b.2.2, 7.6.1b.2.3 and 7.6.1b.1.2.4.2.

5.2.5.1.2.1 Primary Containment Temperature Monitoring System

Temperatures within the drywell are monitored at various elevations. A drywell ambient temperature rise will indicate the presence of reactor coolant or steam leakage. Temperature monitoring of the containment provides an indirect indication of leakage as defined in regulatory position (3) of Regulatory Guide 1.45.

A detailed description of the system, sensitivity and response time, and the system reliability is discussed in Subsection 7.6.1b.1.2.

Limiting temperature conditions are included in the Technical Specifications.

Provisions for testing and calibration are described in Subsection 7.6.2b.

5.2.5.1.2.2 Primary Containment Pressure Monitoring System

Pressure monitoring within the containment provides an indirect method of detecting leakage.

The drywell pressure fluctuates slightly during reactor operation as a result of barometric pressure changes and out-leakage. A pressure increase above normal values indicates a RCS leak in the primary containment.

The primary containment monitoring system and instrumentation is described in Subsection 7.6.1b.

Subsection 7.5.1b identifies safety-related display instrumentation.

5.2.5.1.2.3 Primary Containment Atmosphere Monitoring - Airborne Radioactivity Monitoring

The primary containment is continuously monitored for airborne radioactivity. A sample is drawn from the primary containment and a sudden increase of activity indicates a steam or reactor water leakage.

5.2.5.1.2.3.1 Sensitivity and Response Time

The objective of the drywell leak detection monitors as indicated in Regulatory Guide 1.45 is to be able to detect less than 1 gpm of unidentified primary coolant pressure boundary leakage in 1 hour. Several detection systems supplied to accomplish this are the drywell sump level monitor (see Subsection 5.2.5.1.2.4), a noble gas radiation monitor, and a particulates radiation monitor. A radioiodine collector is also provided. The two radiation monitors sample drywell for the activity levels on the assumption that flashing coolant leakage will result in radioactivity in the atmosphere. The radioiodine collector provides a means for laboratory analysis of a containment air sample for radioiodine activity.

The reliability, sensitivity and response times of radiation monitors to detect 1 gpm in 1 hour of Reactor Coolant Pressure Boundary leakage will depend on many complex factors. The major factors are discussed below:

A. Source of Leakage

1) Location of Leakage

The amount of activity which would become airborne following a 1 gpm leak from the RCPB will vary depending upon the leak location and the coolant temperature and pressure. For example, a feedwater pipe leak will have concentration factors of 100 to 1000 lower than a recirculation line leak. A steam line leak will be a factor of 10 to 100 lower in iodine and particulate concentrations than the recirculation line leak, but the noble gas concentrations may be comparable. A RWCU leak upstream of the demineralizers and heat exchangers will be a factor of 10 to 100 higher than downstream, except for noble gases. Differing coolant temperatures and pressures will affect the flashing fraction and partition factor for iodines and particulates. Thus, an airborne concentration cannot be correlated to a quantity of leakage without knowing the source of the leakage.

2) Coolant Concentrations

Variations in coolant concentrations during operation can be as much as several orders of magnitude within a time frame of several hours. These effects are mainly due to spiking during power transients or changes in the use of the RWCU system. Examples of these transients for I-131 can be found in NEDO-10585 (8/72), Behavior of Iodine in Reactor Water During Plant Shutdown and Startup. Thus, an increase in the coolant concentrations could give increased containment concentrations when no increase in unidentified leakage occurs.

3) Other Sources of Leakage

Since the unidentified leakage is not the sole source of activity in the containment, changes in other sources will result in changes in the containment airborne concentrations. For example, identified leakage is piped to the equipment drain tank in the drywell, but the tank is vented to the drywell atmosphere allowing the release of noble gases and some small quantities of iodines and particulates from the drain tank.

B. Drywell Conditions Affecting Monitor Performance

1) Equilibrium Activity Levels

During normal operation the activity release from acceptable quantities of identified and unidentified leakage will build up to significant amounts in the drywell air. Conversations with several operating plants indicate that levels as high as .1 to 10 times MPC are not uncommon for noble gases and iodines. (MPC refers to "maximum permissible concentration" as defined by 10CFR20, MPC is used here only as a convenient reference.) Due to these high equilibrium

activity levels the small increases due to a 1 gpm increase in leakage may be difficult to see within an hour. Typical MPC ranges are:

	1 MPC to 10 MPC
Noble Gases	$1x10^{-6}$ - $1x10^{-4}$ μ Ci/cc
Particulates	$1x10^{-6}$ - $1x10^{-4}$ μ Ci/cc
Iodines	$5x10^{-7}$ - $5x10^{-5}$ μ Ci/cc

Fresh fuel backgrounds were not considered because no fission products are available at that point in time. The numbers given above include amounts of failed and/or irradiated fuel. These numbers also include normal expected leakage rates.

2) Purge and Pressure Release Effects

Changes in the detected activity levels have occurred during periodic drywell purges to lower the drywell pressure. These changes are of the same order of magnitude as approximately a 1 gpm leak, and are sufficient to invalidate the results from particulate monitors and iodine collector analysis.

3) Plateout, Mixing, Fan Cooler Depletion

Plateout effects on iodines and particulates will vary with the distance from the coolant release point to the detector. Larger travel distances would result in more plateout. In addition the pathway of the leakage will influence the plateout effects. For example, a leak from a pipe with insulation will have greater plateout than a leak from an uninsulated pipe. Although the drywell air will be mixed by the fan coolers, it may be possible for a leak to develop in the vicinity of the radiation detector sample lines. In addition, condensation in the coolers will remove iodines and particulates from the air. Variations in the flow, temperature and number of coolers will affect the plateout fractions. Plateout within the detector sample tube will also add to the reduction of the iodine and iodine and particulate activity levels. The uncertainties in any estimate of plateout effects could be as much as one or two orders of magnitude.

C. Physical Properties and Capabilities of the Detectors

1) Detector Ranges

The detectors were chosen to ensure that the operating ranges covered the concentrations expected in the drywell. The operating ranges are:

Noble Gases	$1x10^{-6}$ to $1x10^{-2}$ μ Ci/cc
Particulates	$1x10^{-9}$ to $1x10^{-4}$ μ Ci/cc

2) Sensitivity

In the absence of background radiation and equilibrium drywell activity levels, the detectors have the following minimum sensitivity.

Noble Gas	1×10^{-6} μ Ci/cc
Particulates	1×10^{-9} μ Ci/cc

3) Counting Statistics and Monitor Uncertainties

In theory these radioactivity monitors are statistically able to detect increases in concentration as small as 2 or 3 times the square root of the count rate, i.e., at 10^6 cpm an increase of 2×10^3, or 0.2%, is detectable; at 10^2 cpm an increase of 20, or 20% is detectable. In addition at high count rates the monitors have dead-time uncertainties and the potential for saturating the monitor or the electronics. Uncertainties in calibration ($\pm$5%) sample flow ($\pm$10%) and other instrument design parameters tend to make the uncertainty in a count rate closer to 20% to 40% of the equilibrium drywell activity.

4) Monitor Setpoints

Due to the uncertainty and extreme variability of the concentrations to be measured in the containment the use of alarm setpoints on the radioactivity monitors would not be practical or useful. As indicated in the following section the setpoints which would be required to alarm at 1 gpm would be well within the bounds of uncertainty of the measurements. The use of such setpoints would result in many unnecessary alarms and the frequent resetting of setpoints. A setpoint alarm on the sump level monitor alone is used; the radioactivity monitors are for supporting information to confirm that the leak is radioactive. The alarm setpoints for the radiation monitors will be set significantly above background to prevent nuisance alarms. The actual setpoint will be changed as background increases. At these levels, the radiation monitors will provide no warning of a 1 gpm leak in one hour.

5) Estimated Monitor Responses

Table 5.2-13 estimates the expected monitor responses for several types of leaks and several types of monitors. As indicated in column 3, the added activity in containment from a 1 gpm leak for 1 hour is less than the nominal 20% increase which could be meaningfully detected. The final columns estimate the detectable leakage in 1 hour.

6) Operator Action

There is no direct correlation or known relationship between the detector count rate and the leakage rate, because the coolant activity levels, source of leakage, and background radiation levels (from leakage alone) are not known and cannot be cost-effectively determined in existing reactors. There are also several other sources of containment airborne activity (e.g. safety relief valve leakage) which further complicate the correlation.

Thus, the recommended procedure for the control room operator is to set an alarm setpoint at 1 gpm in 1 hour on the sump level monitor (measuring water collected in the sump which may not exactly correspond to water leaking from an unidentified source). When the alarm is actuated, the operator will review all other monitors (e.g., noble gas, particulates, temperature, pressure, fan cooler drains, etc.) to determine if the leakage is from the primary coolant pressure boundary and not from an SRV or cooling water system, etc. Appropriate actions will then be taken in accordance with Technical Specifications. The review of other monitors will consist of comparisons of the increases and rates of increase in the values previously recorded on the strip chart recorders. Increases in all parameters except sump level will not be correlated to a RCPB leakage rate. Instead, the increases will be compared to normal operating limits and limitations (e.g., 2 psi maximum pressure for ECCS initiation) and abnormal increases will be investigated.

Since the Technical Specification limit for leakage is allowed to be averaged over 24 hours, quick and accurate responses are not necessary unless the leakage is very large and indicative of a pipe break. In this case, the containment pressure and reactor vessel water level monitors will alarm within seconds, and the sump level monitor would alarm within minutes or tens of minutes.

The radiation monitor alarms will not be set to levels that correspond to RCPB leakage levels since the correlations can't be made. Also, since the containment airborne activity levels vary by orders of magnitude during operation due to power transients, spiking, steam leaks, and outgassing from sumps, etc., an appropriate alarm setpoint, if one is used, should be determined by the operator based on experience with the specific plant. A setpoint level of 2 to 3 times the background level during full power steady state operation may be useful for alarming large leaks and pipe breaks, but it would not always alarm for 1 gpm in 1 hour.

7) Conclusion

Due to the sum total of the uncertainties identified in the previous paragraphs the iodine collector analysis, gaseous, and particulate monitors will not be relied upon for leak detection purposes but only as supporting instrumentation. These monitors will be used to give supporting information to that supplied by the sump level monitors and would be able to give an early warning of a major leak especially if equilibrium containment activity levels are low. However, the uncertainties and variations in noble gas leaks and concentrations would preclude the setting of a meaningful set point on the monitors.

5.2.5.1.2.4 Drywell Floor Drain Sump Monitoring System

The drywell floor drain sump monitoring system is designed to permit leak detection in accordance with Regulatory Guide 1.45.

5.2.5.1.2.4.1 System Description

Two drywell floor drain sumps are located in the primary containment for collection of leakage from vent coolers, control rod drive flange leakage, chilled water drains, cooling water drains, and overflow from the equipment drain sump.

The drywell floor drain sump is located at the drywell diaphragm slab low point. Unidentified leakages will, by gravity, flow down the slab surface into the floor drain sump. No floor drain piping system is employed for this purpose. Piped inputs to the drywell floor drain sump are from clean system drains. No surveillance program is planned to detect piped equipment drain system blockage.

Small, unidentified leakages of concern flowing into the drywell floor drain sump will not be masked by larger, acceptable, identified leakages overflowing from the drywell equipment drain tank. The drywell equipment drain tank drains by gravity. During conditions of acceptable identified leakage rates, the gravity flow from the drywell equipment drain tank will be capable of preventing the drywell equipment drain tank from overflowing to the drywell floor drain sump.

Water flow rate greater than 0.5 gpm can be detected by monitoring changes of level over a time period. The following method of flow rate measurement was selected to comply with the requirements of Regulatory Guide 1.45. The necessary sensitivity is obtained by measuring the changes of level during a fixed time interval. For this purpose, a stepped level measurement system is installed in each of the sumps. The level of each sump is recorded by stepped pen recorders located in the Main Control Room. The change in sump level per unit of time determines the leak rate.

There is no reliable quantitative relationship between the sump level and the leakage rate from any source. The quantity is dependent upon the temperature and pressure of the containment and the leak and the location of the leak. Part of the leak will flash to steam; it may be partially trapped between insulation layers. Presumably the leakage will get to an equilibrium level where most of it ends up in the sump, unless the drywell is vented to relieve the pressure buildup. Since the Technical Specification allows 24-hour averaged leak limits, short term variations in the ability to relate the sump quantity to the leaked quantity are ignored, and it is assumed that all leakage reaches the sump. The errors introduced will not impair the ability to detect larger leaks which could rapidly result in severe accidents.

The upper drywell area, above the refueling bellows and seal plate, is capable of accumulating a quantity of water not monitored by the drywell sump leak detection system.

The presence of a leakage liquid accumulation reservoir in the upper drywell is not a safety concern for leaks that are possible in any significant quantity from the reactor coolant pressure boundary. Any leak of significance from the vessel head area would be in the form of steam, condensing in some relatively small quantities, with the rest pressurizing the area and flowing out to the lower drywell area via the drywell cooling system head area returns.

Once in the lower drywell, the chillers will condense the steam and the leakage would be identified and quantified in the conventional manner. The reservoir may contain, and may eventually fill with, water, but the detection and mitigation of the leak is not directly affected by the unmonitored accumulation of leakage.

Less quantifiable leak detection means are also available from the temperature, pressure, particulate and Noble Gas monitoring instrumentation.

Liquid leakage in the vessel head area would be expected only under shutdown/hydro conditions or under unusual or EOP conditions (vessel flooding accident response) when leakage is of relatively minor consequences or concern. During normal operation, relatively small amounts of liquid water could come from an instrument line leak. Such a leak would quickly develop into a steam leak, to be detected by normal means. Head spray lines are normally shut off from sources of water, and are normally open only briefly when preparing for vessel disassembly.

Some leakage will no doubt be trapped in insulation etc., but no other large reservoirs for leakage have been found.

Each sump is equipped with two pumps which operate in an alternating mode. High sump level starts the pump automatically. Remote manual control of the pump is provided in the control room. Both pumps will be operating as soon as an abnormal, high level is detected. The capability of each pump is such that normal expected flow rates can be easily accommodated.

As discussed above for the drywell sump monitoring system and as discussed in section 5.2.5.1.2.3 for Primary Containment Atmosphere – Airborne Radiation Monitoring system ability to detect leakage is reduced during containment inerting and purging. These systems are not credited in any FSAR safety analysis. Therefore, the reduction in detection capability during containment inerting and purging evolutions is not safety significant. Note that FSAR Section 7.3.1.1b.1.1 identifies that the containment purge line will isolate on high radiation at the SGTS exhaust stack. This isolation feature provides additional protection of the public from any significant radiological release during inerting and purging activities.

5.2.5.1.2.4.2 Instrumentation

Magnetic float type continuous level probes are used to measure the fluid level and provide the signal for the recording of the actual sump level in the control room and for the starting and stopping of the drywell sump pumps. Excessive sump level is alarmed in the control room. The sump level can be observed by the control room operator.

5.2.5.1.2.4.3 Drywell Equipment Drain Tank Level Monitoring System

The drywell equipment drain tank collects identified leakage within the primary containment from reactor head seal leak off, bulkhead drain, refueling bellows drain, RPV head vent, recirculation pump seals, reactor recirculation pump cooler drains, and RPV bottom drain(Unit 1 only).

All identified leakages which may have temperatures of 212°F or above are hard-piped directly to the drywell equipment drain tank. These leakages will tend to partially flash into steam and then condense in the drain pipe. This approach minimizes the possibility that leakage will escape as steam into the containment atmosphere prior to measurement in the equipment drain tank.

The drywell equipment drain tank drains by gravity. The drain tank's discharge valves automatically open when a predetermined high level in the tank is reached. The discharge valves close at a predetermined low level.

Water flow rate better than 0.5 gpm can be obtained by monitoring changes of level over a time period. The following method of flow rate measurement was selected to comply with the requirements of Regulatory Guide 1.45. The necessary sensitivity is obtained by measuring the changes of level during a fixed time interval. For this purpose a continuous level measurement system is installed in the tank. An electronic signal directly proportional to the actual tank level is applied to one pen of a multi-pen recorder, to an electronic sample and hold device, and to an electronic differential switch. The sample and hold device, upon command from a timer, applies its output signal to the second pen of the multi-pen recorder and to the second input of the electronic differential switch. The sample and hold unit's output signal level is regularly updated to the reference tank level signal.

The actual level signal of the tank and the reference level signal are continuously displayed on the multi-pen recorder. The same signals are being monitored by the electronic differential switch. When the level signals differ by 216 gallons or more during a 50 minute period (equal to 4.3 gpm) an alarm is actuated on the local panel and on the control board in the main control room. The change in tank level per unit of time determines the leak rate and is available from the recorder.

5.2.5.1.2.4.4 Sensitivity and Response Time of Measurement

The method for liquid leak detection in the primary containment is designed to meet the recommended water leak rate changes of 0.5 to 1.0 gpm as defined in Regulatory Guide 1.45.

The following assumptions and design considerations were incorporated:

a) Leak rate is directly proportional to the associated change in sump level.

b) The selected measurement period T for the average change in level is 50 minutes (not available on Unit 1 drywell floor drain sump).

c) The drywell floor drain sumps have a capacity of 150 gal with a depth of 5 in. The drywell equipment drain tank useful capacity is 842 gal with a useful depth of 36 in.

d) Recorder response is equal to 1 second for full range.

e) The electronic differential switch setpoint can alarm at rates less than or equal to one gpm (not available on Unit 1 drywell floor drain sump).

These design factors allow a detection of 1 gpm flow rate within a 50-minute time period (not available on Unit 1 drywell floor drain sump).

The operator can verify this leak rate on the recorder in the control room by observation of the average change of level (not available on Unit 1 drywell floor drain sump).

5.2.5.1.2.4.5 Signal Correlation and Calibration

Drywell Floor Drain Sump

The sump depth of 0-5 in. is displayed on a 0-100 percent recorder chart, which relates to the sump nominal capacity of 0-150 gal.

Drywell Equipment Drain Tank

The measured tank depth of 36 in. is displayed on a 0-100 percent recorder chart. This relates directly to the measured tank capacity of 842 gal.

5.2.5.1.2.4.6 Seismic Qualifications

The drywell floor drain sump, all drywell drain piping, and all instrumentation used to monitor drywell floor drain sump are qualified to operate following an OBE. The drywell equipment drain tank, drywell equipment drain tank level instrumentation, and drywell floor drain sump pumps are not qualified to operate following an OBE.

Credit will be taken for monitoring unidentified leakage following an OBE thru the use of the drywell floor drain sump level monitoring system. The proper functioning of at least one leakage detection system following an SSE is provided by the design of the air borne radioactivity monitoring system. Refer to Subsection 7.6.1b for description.

5.2.5.1.2.4.7 Testing and Calibration

Calibration of level sensors is possible by observing the change in level during the periodic pump down operations of the drywell floor drain sump, and periodic draining of the drywell equipment drain tank.

For the drywell floor drain sump, the pumps are automatically started and stopped by electronic level switches which receive their signals from the level probes in the sumps, but can also be operated manually, at any time, to check the calibration of the level sensors. In the event that the high-high level is reached, two pumps will operate. The drain tank discharge valves are opened automatically on high level and can be operated manually at any time, to check the calibration of the level sensors.

5.2.5.1.3 Detection of Abnormal Leakage Outside the Primary Containment

The method used to monitor for leakage for each reactor coolant pressure boundary component is listed in Table 5.2-8. Leak detection systems are also described in FSAR Section 7.3.1.1a.2 and 7.6.1a.4.

(1) Ambient and Differential Room Ventilation Temperature

Outside drywell, the piping within each system which interfaces with the reactor coolant pressure boundary is installed in compartments or rooms, separate form other systems where feasible, so that leakage may be detected by area temperature measurement. Ambient and differential temperature sensors are installed in the HPCI, RCIC, RHR and RWCU equipment rooms, the HPCI/RCIC Piping Area and the Reactor Building Main Steam Tunnel. Ambient temperature sensors are installed in the Turbine Building Main Steam Tunnel.

Ambient temperature switches, connected to the respective sensors, initiate an alarm or system isolation when the temperature rises to a preset value. Differential temperature switches initiated a system isolation and an associated isolation alarm if differential temperature reached the isolation setpoint. This leak detection isolation feature has

been disabled and the instrumentation remains in place. Differential temperature switches, connected to respective to respective sensors, initiate a pre-isolation alarm to alert operators of a potential reactor coolant release. The high ambient set points include sufficient margin above the post LOCA or normal design maximum temperature to preclude inadvertent isolation signals. The setpoints isolation and alarm are designed to detect a leakage rate below the leak rate corresponding to critical crack size for the smallest high energy line in the room which is part of the respective system. The HPCI, RCIC, and RWCU equipment rooms, HPCI/RCIC Piping Area and Main Steam Tunnel ambient and differential temperature switch set points are based on the temperature rise resulting from a leak at system conditions corresponding to full reactor power. The RHR Pump Room ambient and differential temperature switch set points are based on the temperature rise resulting from a leak at system conditions corresponding to Hot Shutdown (Reactor Condition 3). The RHR Pump Room ambient and differential temperature instruments have been removed from the Technical Specifications. The RHR Pump Room ambient and differential temperature instruments initiate an alarm only when temperature rises to a preset value.

(2) Visual and Audible Inspection

Accessible areas are inspected periodically and the temperature indicators discussed above are monitored regularly as required by Chapter 16. Alarms provide visual and audible indication in the control room of leakage. Any indication of abnormal leakage will be investigated.

(3) Differential Flow Measurement (Reactor Water Cleanup System Only)

Because of the arrangement of the reactor water cleanup system, differential flow measurement provides an accurate leakage detection method. The flow from the reactor vessel is compared with the flow back to the vessel. An alarm in the control room and an isolation signal are initiated when higher flow out of the reactor vessel indicates a leak may exist.

(4) Equipment Room Flood (Water Level) Detection

The HPCI, RCIC Core Spray and RHR Equipment Rooms are monitored for water level on the floor of the room. When the water level reaches a preset value, an alarm is initiated in the control room. The level switches are designed to detect water accumulation resulting from a leak. For a water leak, the accumulation will account for nearly 100% of the leaking fluid. For a steam leak, the water accumulation will consist of the percentage of leaking steam which condenses to water within the room.

(5) Detection of Large Leaks (High Flow and Low Reactor Water Level)

The main steam line, HPCI steam line, RCIC steam line, RHR Shutdown Cooling suction line and RWCU suction line are all monitored for high flow. A high flow signal will initiate an alarm in the control room and isolation of the system. Low reactor water level will isolate the main steam, RHR and RWCU lines. High flow or low reactor water level can indicate a break or large leak in the reactor coolant piping.

5.2.5.2 Leak Detection Devices for NSS-System

(1) Reactor Vessel Head Closure

The reactor vessel head closure is provided with double seals with a leak off connection between seals that is piped through the normally closed manual valves to the equipment drain tank. Leakage through the first seal is indicated locally in the reactor building. The second seal then operates to contain the vessel pressure.

(2) Reactor Water Recirculation Pump Seal

As discussed in Subsection 5.4.1.3, the reactor recirculation pump shaft is provided with two seals. Leakage past each seal is piped to the Drywell Equipment Drain Tank. Leakage past the first stage seal is designed to flow at approximately 0.75 gpm normally. The first stage seal leakoff line is provided with a high/low flow alarm which actuates at 0.9 gpm increasing or 0.5 gpm decreasing. The second stage pump seal is designed for zero leakage normally. The second stage seal leakoff line is provided with a high flow alarm which actuates at 0.1 gpm.

(3) Safety/Relief Valves

Temperature sensors connected to a multipoint recorder are provided to detect safety/ relief valve leakage during reactor operation. Safety/relief valve temperature elements are mounted, using a thermowell, in the safety/relief valve discharge piping several feet from the valve body. Temperature rise above ambient is annunciated in the main control room. See the nuclear boiler system P&ID, Dwgs. M-141, Sh. 1 and M-141, Sh. 2.

(4) Valve Packing Leakage

Power-operated valves in the nuclear boiler system and recirculation system were originally provided with valve stem packing leakoff connections. These leakoffs were either plugged or provided with normally closed isolation valves and capped. It was thought by keeping these leakoff connections isolated and thereby providing two sets of valve packing, that stem leakage would be limited. Recent research and testing has, in fact, shown that one set of graphite packing provides a more effective seal than two independent sets of packing. As a result, a programmatic replacement of the two sets of packing with one set of graphite packing has been undertaken. As part of this effort leakoff isolation valves will be removed and the leakoff lines will be permanently capped.

5.2.5.3 Limits for Reactor Coolant Leakage

5.2.5.3.1 Total Leakage Rate

The total leakage rate consists of all leakage, identified and unidentified, that flows to the drywell floor drain sumps and the equipment drain tank. The criterion for establishing the total leakage rate limit is based on the makeup capability of the RCIC system. The total leakage rate limit is established at 25 gpm. The total leakage rate limit is also set low enough to prevent overflow of the drywell sumps.

5.2.5.3.2 Normally Expected Leakage Rate

The pump packing glands, valve stems, and other seals in systems that are part of the reactor coolant pressure boundary and from which normal design leakage is expected are provided with drains or auxiliary sealing systems. Nuclear system valves and pumps inside the drywell are equipped with double seals. The double seals on valves are systematically being replaced with single sets of graphite packing. A single set of graphite packing has been shown to produce a more effective seal than a double set of standard packing. Leakage from the primary recirculation pump seals is piped to the drywell equipment drain tank as described in Subsections 5.2.5.2(2) and 5.4.1.3. Leakage from the safety/relief valves is identified by temperature sensors in the discharge line that transmit to the control room. Any temperature increase above the drywell ambient temperature detected by these sensors indicates valve leakage.

Except for the leakoffs from the reactor recirculation pumps, all drains routed to the Drywell Equipment Drain Tank are normally isolated by closed valves. Therefore, any leakage measured during normal plant operation in the Equipment Drain Tank is attributable to the recirculation pumps.

The leakage rates from the recirculation pumps, plus any other leakage rates measured while the drywell is open, are defined as identified leakage rates. Table 5.2-11 lists normal and maximum identified leakage rates directed into the Drywell Equipment Drain Tank, and the associated activity concentrations.

5.2.5.4 Unidentified Leakage Inside the Drywell

5.2.5.4.1 Unidentified Leakage Rate

The unidentified leakage rate is the portion of the total leakage rate received in the drywell sumps that is not identified as previously described. A threat of significant compromise to the nuclear system process barrier exists if the barrier contains a crack that is large enough to propagate rapidly (critical crack length). The unidentified leakage rate limit must be low because of the possibility that most of the unidentified leakage rate might be emitted from a single crack in the nuclear system process barrier.

An allowance for leakage that does not compromise barrier integrity and is not identifiable is made for normal plant operation.

The unidentified leakage rate limit is established at 5 gpm rate to allow time for corrective action before the process barrier could be significantly compromised. This 5 gpm unidentified leakage rate is a small fraction of the calculated flow from a critical crack in a primary system pipe (Figure 5.2-10). Safety limits and safety limit settings are discussed in Chapter 16.
Table 5.2-12 lists unidentified leakage rates directed into the Drywell Floor Drain Sump, and the associated Activity Concentrations.

5.2.5.4.2 Sensitivity and Response Times

Sensitivity, including sensitivity tests and response time of the leak detection system are covered in Subsection 7.6.1.

5.2.5.4.3 Length of Through-Wall Flaw

Experiments conducted by GE and Battelle Memorial Institute, (BMI), permit an analysis of critical crack size and crack opening displacement (Reference 5.2-4). This analysis relates to axially oriented through-wall cracks.

(1) Critical Crack Length

Satisfactory empirical expressions have been developed to fit test results. A simple equation which fits the data in the range of normal design stresses (for carbon steel pipe) is

$$\ell_c = \frac{15000D}{\sigma_h}$$

where

ℓ_c = critical crack length (in.)
D = mean pipe diameter (in.)

σ_h = nominal hoop stress (psi).

(2) Crack Opening Displacement

The theory of elasticity predicts a crack opening displacement of

$$W = \frac{2 \ell \sigma}{E} \qquad \text{(Eq. 5.2-1)}$$

where

ℓ = crack length
σ = applied nominal stress
E = Young's Modulus

Measurements of crack opening displacement made by BMI show that local yielding greatly increases the crack opening displacement as the applied stress s approaches the failure stress sf. A suitable correction factor for plasticity effects is:

$$c = \text{SEC} \; \frac{\pi\sigma}{2\sigma_f}$$

The crack opening area is given by

$$A = C\ \frac{\pi}{4}\,wl\ \frac{\pi\, l^2\,\sigma}{2E}\ \ SEC\ \ \frac{\pi\sigma}{2\,\sigma_f}$$

For a given crack length ℓ, σ_f = 15,000 D/ℓ.

(3) Leakage Flow Rate

The maximum flow rate for blowdown of saturated water at 1000 psi is 55 lb/sec-in^2. and for saturated steam the rate is 14.6 lb/sec-in^2, (Reference 5.2-5). Friction in the flow passage reduces this rate, but for cracks leaking at 5 gpm (0.7 lb/sec) the effect of friction is small. The required leak size for 5 gpm flow is

A = 0.0126 in^2. (saturated water)
A = 0.0475 (saturated steam)

From this mathematical model, the critical crack length and the 5 gpm crack length have been calculated for representative BWR pipe size (Schedule 80) and pressure (1050 psi).

The lengths of through-wall cracks that would leak at the rate of 5 gpm given as a function of wall thickness and nominal pipe size are:

Nominal Pipe Size (Sch 80), in.	**Average Wall Thickness, in.**	**Crack Length ℓ, in.**	
		Steam Line	**Water Line**
4	0.337	7.2	4.9
12	0.687	8.5	4.8
24	1.218	8.6	4.6

The ratios of crack length, ℓ, to the critical crack length, ℓ_c, as a function of nominal pipe size are:

Nominal Pipe Size (Sch 80), in.	Ratio ℓ/ℓ_c	
	Steam Line	Water Line
4	0.745	0.510
12	0.432	0.243
24	0.247	0.132

It is important to recognize that the failure of ductile piping with a long, through-wall crack is characterized by large crack opening displacements which precede unstable rupture. Judging from observed crack behavior in the GE and BMI experimental programs, involving both circumferential and axial cracks, it is estimated that leak rates of hundreds of gpm will precede crack instability. Measured crack opening displacements for the BMI experiments were in the range of 0.1 to 0.2 in. at the time of incipient rupture, corresponding to leaks of the order of 1 sq in. in size for plain carbon steel piping. For austenitic stainless steel piping, even larger leaks are expected to precede crack instability, although there are insufficient data to permit quantitative prediction.

The results given are for a longitudinally oriented flaw at normal operating hoop stress. A circumferentially oriented flaw could be subjected to stress as high as the 550°F yield stress, assuming high thermal expansion stresses exist. It is assumed that the longitudinal crack, subject to a stress as high as 30,000 psi, constitutes a "worst case" with regard to leak rate versus critical size relationships. Given the same stress level, differences between the circumferential and longitudinal orientations are not expected to be significant in this comparison.

Figure 5.2-10 shows general relationships between crack length, leak rate, stress, and line size, using the mathematical model described previously. The asterisks denote conditions at which the crack opening displacement is 0.1 in., at which time instability is imminent as noted previously under "Leakage Flow Rate". This provides a realistic estimate of the leak rate to be expected from a crack of critical size. In every case, the leak rate from a crack of critical size is significantly greater than the 5 gpm criterion.

If either the total or unidentified leak rate limits are exceeded, an orderly shutdown would be initiated and the reactor would be placed in a cold shutdown condition within 24 hours.

5.2.5.4.4 Margins of Safety

The margins of safety for a detectable flaw to reach critical size are presented in Subsection 5.2.5.4.3. Figure 5.2-10 shows general relationships between crack length, leak rate, stress and line size using the mathematical model.

5.2.5.4.5 Criteria to Evaluate the Adequacy and Margin of the Leak Detection System

For process lines that are normally open, there are at least two different methods of detecting abnormal leakage from each system within the nuclear system process barrier located in the drywell and reactor building as shown in Table 5.2-8. The instrumentation is designed so it can be set to provide alarms at established leakage rate limits and isolate the affected system, if necessary. The alarm points are determined analytically or based on measurements of appropriate parameters made during startup and preoperational tests.

The unidentified leakage rate limit is based, with an adequate margin for contingencies, on the crack size large enough to propagate rapidly. The established limit is sufficiently low so that, even if the entire unidentified leakage rate were coming from a single crack in the nuclear system process barrier, corrective action could be taken before the integrity of the barrier would be threatened with significant compromise.

The leak detection system will satisfactorily detect unidentified leakage of 5 gpm.

Sensitivity, including sensitivity testing and response time of the leak detection system, and the criteria for shutdown if leakage limits are exceeded, are covered in Section 7.6.1.

5.2.5.5 Differentiation Between Identified and Unidentified Leaks

Subsection 5.2.5.1 describes the systems that are monitored by the leak detection system. The ability of the leak detection system to differentiate between identified and unidentified leakage is discussed in Subsections 5.2.5.1, 5.2.5.4 and 7.6.1.

5.2.5.6 Sensitivity and Operability Tests

Testability of the leakage detection system is contained in Subsection 7.6.1.

5.2.5.7 Safety Interfaces

The Balance of Plant-GE Nuclear Steam Supply System safety interfaces for the Leak Detection system are the signals from the monitored balance of plant equipment and systems which are part of the nuclear system process barrier, and associated wiring and cable lying outside the Nuclear Steam Supply System Equipment. These balance of plant systems and equipment include the main steam line tunnel, the safety/relief valves, and the drywell sumps and equipment drain tank.

5.2.5.8 Testing and Calibration

Provisions for Testing and Calibration of the leak detection system is covered in Chapter 14.

5.2.6 References

5.2-1 R. Linford, "Analytical Methods of Plant Transient Evaluation for the General Electric Boiling Water Reactor," NEDO-10802, April, 1973.

5.2-2 J. M. Skarpelos and J. W. Bagg, "Chloride Control in BWR Coolants," June, 1973, NEDO-10899.

5.2-3 W. L. Williams, Corrosion, Vol. 13, 1957, p. 539t.

5.2-4 GEAP-5620, Failure Behavior in ASTM A106B Pipes Containing Axial Through-Wall Flows, by M. B. Reynolds, April, 1968.

5.2-5 "Investigation and Evaluation of Cracking in Austenitic Stainless Steel Piping of Boiling Water Reactor Plants," NUREG-76/067, NRC/PCSG, dated October, 1975.

5.2-6 F. Odar, "Safety Evaluation for General Electric Topical Report: Qualification of the One-Dimensional Core Transient Model for Boiling Water Reactors," NEDO-24154, NEDE-24154-P Vol. I, II, III, dated June, 1980.

5.2-7 Faynshtein, K., and D. R. Pankratz, "Power Uprate Engineering Report for Susquehanna Steam Electric Station, Units 1 and 2," General Electric Report NEDC-32161P, as revised by PP&L Calculation EC-PUPC-1001, Revision 0, March, 1994.

5.2-8 Not Used

5.2-9 Not Used

5.2-10 Not Used

5.2-11 Not Used

5.2-12 Not Used

5.2-13 Not Used

5.2-14 XN-NF-80-19(PA)(A) Volume 4, Revision 1, Exxon Nuclear Methodology for Boiling Water Reactors: Application of the ENC Methodology to BWR Reloads, Exxon Nuclear Company, June 1986.

5.2-15 ANF-913(P)(A) Volume 1 Revision 1 and Volume 1 Supplements 2,3, and 4 COTRANSA2: A Computer Program for Boiling Water Reactor Transient Analyses, Advanced Nuclear Fuels Corporation, August 1990.

TABLE 5.2-1

REACTOR COOLANT PRESSURE BOUNDARY COMPONENTS

CODE CASE INTERPRETATIONS

For GE supplied equipment, the following applicable Code Case Interpretations are listed below:

A. Vessel Fabrication: Section III Cases

1.	1141-1	Foreign Produced Steel
2.	1332-5	Requirements for Steel Forgings
3.	1420	Nickel-Chromium-Iron Alloy Pipe or Tube
4.	1441-1	Waiving of 3S Limit for Section III Construction
		As the need arises, we intend to permit the use of the following code cases:
5.	1492	Postweld Heat Treatment, Section I, III and VIII, Div. 1 and 2

B. System Assembly Cases That May Be Applied: Section III Cases

1.	1424	Requirements for Stamping of Field Installation for Section III, Class A, Vessels
2.	1428	Field Welding of Joints Between Components
3.	1429	Completion of Components in the Field
4.	1430	Owner's Certificate of Authorization
5.	1442	Pressure Tests of Nuclear Components

C. Pressure Integrity of Piping and Equipment

1. Code Case 78, ANSI B31, Documentation for pipe and fittings 3/4 inch nominal pipe size and smaller.
2. Code CASE 1388, Requirements for stainless steel - precipitation hardening. Control valves.
3. Code Case 1555, Certification of Liquid Pressure Safety Relief Valves on Liquids for which NV Code Symbol Stamp applies.
4. Code Case N-242, Materials Certification, Alternate rules used for Section III, Division 1, Classes 1,2,3,MC and CS Component Construction.

TABLE 5.2-2

NUCLEAR SYSTEM SAFETY /RELIEF SET POINTS*

UNITS 1 AND 2

No. of Valves	Spring Set Pressure (psig)	ASME Rate Capacity at 103% Spring Set Pressure (lbs/hr)
2	1175	883,950
6	1195	898,800
8	1205	906,250

* Six of the Safety Relief Valves Serve in the Automatic Depressurization Function.

TABLE 5.2-3

DESIGN TEMPERATURE, PRESSURE AND MAXIMUM TEST

PRESSURE FOR RCPB COMPONENTS

Component	Design Temperature (F°)	Design Pressure (psig)	Maximum Test Pressure (psig)
REACTOR VESSEL	575	1250	1563
RECIRCULATION SYSTEM			
Pump Discharge Piping	575	1500	(1)
Pump Suction Piping	575	1250	(1)
Pump	575	1500	(2)
Discharge Valves	575	1525	(2)
Suction Valves	575	1275	(2)
MAIN STEAM SYSTEM			
Reactor Vessel to Second Isolation	575	1250	(1)
Valve Piping	575	1250	(3)
Valves			
RESIDUAL HEAT REMOVAL SYSTEM			
Shutdown Suction			
Recirculation Suction Piping through Second Isolation Valve			
Piping	565	1250	
Valves	565	1250	(1)
Recirculation Discharge Piping Through Second Isolation Valves			(2)
Piping	565	1500	
Valves	565	1500	(1)
			(2)
REACTOR FEEDWATER			
Reactor Vessel to Maintenance Valve (F011)	575	1250	(1)
Piping		1250	
Maintenance Valve through Outboard Isolation Valve			
Piping	546	1350	
Valves	546	1350	

TABLE 5.2-3

DESIGN TEMPERATURE, PRESSURE AND MAXIMUM TEST

PRESSURE FOR RCPB COMPONENTS

Component	Design Temperature (F°)	Design Pressure (psig)	Maximum Test Pressure (psig)
REACTOR CORE ISOLATION COOLING SYSTEM			
Main Steam Line Through Second Isolation Valve			
Piping	585	1250	(1)
Valves	585	1250	(2)
RCIC Steam Supply Break Detection Instrumentation Line			
Piping	585	1350	(1)
Valves	585	1350	(2)
HIGH-PRESSURE COOLANT INJECTION SYSTEM			
Main Steam Line Through Second Isolation Valve			
Piping	585	1250	(1)
Valves	585	1250	(2)
HPCI Steam Supply Break Detection Instrument Line			
Piping	585	1350	(1)
Valves	585	1350	(2)
CORE SPRAY SYSTEM			
DCA-107 Injection Lines to Reactor Vessel			
Piping	575	1250	(1)
Valves	575	1250	(2)
DCA-109 Injection Lines to Second Isolation Valve			
Piping	400	1250	(1)
Valves	400	1250	(2)
STANDBY LIQUID CONTROL SYSTEM			
Reactor Vessel to Second Isolation Valve			
Piping	575	1250	(1)
Valves	575	1250	(2)

TABLE 5.2-3

DESIGN TEMPERATURE, PRESSURE AND MAXIMUM TEST PRESSURE FOR RCPB COMPONENTS

Component	Design Temperature (F°)	Design Pressure (psig)	Maximum Test Pressure (psig)
REACTOR WATER CLEANUP SYSTEM			
Pump Suction			
Recirculation Suction Piping to Isolation Valve Outside Drywell	565	1250	(1)
Piping	565	1250	(2)
Valves			
Vessel Drain to Pump Suction	565	1250	(1)
Piping	565	1250	(2)
Valves			

(1) Test pressure is calculated at 1.25 x design pressure.

(2) Test pressure is calculated at 1.50 x design pressure.

(3) Test pressure is based on interpolation of ANSI B16.5 tables for design pressure and temperature designated.

TABLE 5.2-4

REACTOR COOLANT PRESSURE BOUNDARY MATERIALS

Component	Form	Material	Specification (ASTM/ASME)
Reactor Vessel Heads, Shells	Rolled Plate	Low Alloy Steel	SA-533 Gr. B
	Welds	Low Alloy Steel	SFA-5.5
RPV Top Head and Shell Closure Flange	Forged Ring	Low Alloy Steel	SA-508 Cl.2
	Welds	Low Alloy Steel	SFA-5.5
RPV Closure Flange Studs	Bar	Alloy Steel	SA-540 Grade B24
RPV Closure Flange Nuts and Washers	Smls Tubing	Alloy Steel	SA-540 Grade B23
RPV Nozzles (N1 through N9, N15)	Forged Shapes	Low Alloy Steel	SA-508 Cl.2
(N10, N11, N12, N13, N16)	Forgings	Ni-Cr-Fe Alloy	SB-166
	Welds	Low Alloy Steel	SFA-5.5
RPV Nozzle Safe Ends (N1, N2, N5 Safe End Ext., N8, N10)	Forgings or Plate	Stainless Steel	SA-182, F 316L SA-336, F8 SA-240, 304 or 316
	Welds	Stainless Steel	SFA-5.9 TP.308L or 316L SFA-5.4 TP.308L or 316L
RPV Nozzle Safe Ends/Cap (N5, N9 Cap)	Forgings	Ni-Cr-Fe	SB-166
	Welds	Ni-CR-Fe	SFA-5.14 TP. ERNiCr-3 or SFA-5.11 TP. ENiCrFe-3
RPV Nozzle Safe Ends (N3, N4, N11, N12, N16)	Forgings	Carbon Steel	SA-105 Gr. 2, SA-106 Gr. B or SA-508 Cl.1
	Welds	Carbon Steel	SA-508 Cl. 1 w/309, 308L overlay SFA-5.1, SFA-5.18 Gr. A,
RPV Cladding	Weld Overlay	Austenitic Stainless Steel	SFA-5.9 or SFA-5.4
Control Rod Drive Housings	Pipe	Austenitic Stainless Steel	SA-312, Type 304
	Welds	Austenitic Stainless Steel	SFA-5.9 or SFA-5.4
	Forgings	Stainless Steel	SA-182, F304
In-Core Housings	Tube	Austenitic Stainless Steel Ni-Cr-Fe	SA-213, Type 304 SB-167
	Welds	Inconel	SFA-5.11, Type ENiCrFe-3 or SFA-5.14, Type ERNiCr-3
		Austenitic Stainless Steel	SFA-5.9 or SFA-5.4
	Forgings	Stainless Steel	SA-182, F304

TABLE 5.2-4

REACTOR COOLANT PRESSURE BOUNDARY MATERIALS

Component	Form	Material	Specification (ASTM/ASME)
Nozzle Weld Overlay (N1B)	Welds	Inconel	SFA-5.11, Type ENiCrFe-7 or SFA-5.14, Type ERNiCrFe-7
Nozzle Weld Overlay (U1 N2J)	Welds	Inconel	SFA-5.11, Type ENiCrFe-7 or SFA-5.14, Type ERNiCrFe-7

Additional RCPB component materials and specifications to be used are specified below.

Depending on whether impact tests are required and, depending on the lowest service metal temperature when impact tests are required, the following ferritic materials and specifications are to be used:

Pipe	SA-106 Grade B; SA-333 Grade 6 and SA-155 Grade KCF-70
Valves	SA-105 Grade II; SA-350 Grade LF2 and SA-216 Grade WCB
Fittings	SA-105 Grade II; SA-350 Grade LF1 or LF2; SA-234 or WPB; and SA-420 Grade WPL6
Bolting	SA-193 Grade B7; SA-194 Grades 7 and 2H; SA-194 Grades 4 and 7 impact tested per NB-2300
Welding	SFA 5.1 (E 7015, 7016 and E 7018 only),
Material	SFA 5.4, SFA 5.9, SFA 5.18

For those systems or portions of systems, such as the reactor recirculation system, which require austenitic stainless steel, the following materials and specifications are to be used:

Pipe	SA-376 Type 304; SA-312 Type 304, Type 304 (0.030 Carbon Max.) and 304L; SA-358 Type 304, Type 304 (0.030 Carbon max.)
Valves	SA-182 Grade F-304, F304L, and F-316; SA-351 Grades CF-8, CF-8M; CF3, and CF3M; SA-240 Type 316
Pump	SA-351 Grade CF-8M
Flanges	SA-182 Grade F-316 and F-316L
Bolting	SA-193 Grade B7; SA-194 Grades 7 and 2H; SA-194 Grades 4 and 7 impact tested per NB-2300
Welding	SFA-5.4 (E308-15, E308L-15, E316-15, E308L-16); SFA-5.9 (ER-308, ER 308L, ER-316)
Fittings	SA-182 Grade F-304 and F-304L; SA-403 Grades WP-304, 304W and WP-304L; SA-479 Type 316

TABLE 5.2-5

BWR WATER CHEMISTRY

	Concentrations – Parts Per Billion (ppb)				Conductivity	
	IRON	COPPER	CHLORIDE	OXYGEN	µmho/cm @ 25°C	pH @ 25°C
Condensate (1)* (1) (2)	15-30	3-5	≤20	20-50	≈0.1	≈7
Condensate Treatment Effluent (2)*	5-15	<1	≈0.2	20-50	<0.1	≈7
Feedwater (3)*	5-15	<1	≈0.2	20-50	<0.1	≈7
Reactor Water (4)*						
(a) Normal Operation	10-50	<20	<20	100-300	0.2-0.5	≈7
(b) Shutdown	-	-	<20		<1	≈7
(c) Hot Standby	-	-	<20	See Outline	<1	≈7
(d) Depressurized	-	-	<20	8000	<2	6-6.5
Steam (5)*	0	0	0	10000-30000	≈0.1	--
Control Rod Drive Cooling Water (6)*	-	-	<20	≤50	≤0.1	≈7

*Numerals in parenthesis refer to location delineated on Figure 5.2-8

(1) Typical radioactivity concentrations range from 10^{-4} to 10^{-3} uCi/ml.
(2) Typical nickel concentrations are 5 ppb for normal operation.

TABLE 5.2-5

BWR WATER CHEMISTRY

COOLANT CHEMISTRY LIMITING CONDITION FOR OPERATION	COOLANT CHEMISTRY SURVEILLANCE REQUIREMENTS
a. Prior to startup and during reactor operation, when the reactor is pressurized, or above 212°F, and at less than 1% of rated steam flow, including hot standby, the reactor coolant shall not exceed the following limits: Conductivity at 25°C 2 μmho/cm Chloride 0.1 ppm	a. Reactor coolant shall be continuously monitored to conductivity 1. Whenever the continuous conductivity monitor is inoperable an in-line conductivity measurement shall be obtained at least once every 4 hours. 2. Once a week the continuous monitor shall be checked with an in-line flow cell. This in-line conductivity calibration shall be performed every 24 hours whenever the reactor coolant conductivity is >1.0 μmho/cm at 25°C.
b. During reactor operation in excess of 1% of rated steam flow, the reactor coolant shall not exceed the following limits: Conductivity at 25°C 1 μmho/cm Chloride 0.2 ppm	b. During startup prior to pressuring the reactor above atmospheric pressure, measurements of reactor water quality shall be performed to show conformance with Paragraph a. of limiting conditions.
c. During reactor operation in excess of 1% of rated steam flow, the reactor coolant may exceed the limits of Paragraph b only for the time limits specified here. Exceeding these time limits or the following maximum limits shall be cause for immediately shutting down and placing the reactor in the cold shutdown condition. Conductivity Time above 1 μmho/cm at 25°C 2 weeks/year. Maximum limit – 10 μmho/cm at 25°C Chloride Time above 0.2 ppm 2 weeks/year Maximum limit – 0.5 ppm The reactor shall be shutdown if pH is <5.6 or >8.6 for a 24-hour period.	c. Whenever the reactor is operating (including hot standby conditions), measurements of reactor water quality shall be performed according to the following schedule: 1. Chloride ion content shall be measured at least once every 96 hours. 2. Chloride ion content shall be measured at least every 8 hours whenever reactor conductivity is >1.0 μmho/cm at 25°C.
d. When the reactor is not pressurized (i.e., at or below 212°F), reactor coolant shall be maintained below the following limits: Conductivity at 25°C 10 μmho/cm Chloride 0.5 ppm And pH shall be between 5.3 and 8.6.	d. Whenever the reactor is not pressurized, a sample of the reactor coolant shall be analyzed at least every 96 hours for chloride ion content and pH.
e. When the time limits or maximum conductivity or chloride concentration limits are exceeded, an orderly shutdown shall be initiated immediately. The reactor shall be brought to the cold shutdown condition as rapidly as cooldown rate permits.	

TABLE 5.2-6

SYSTEMS WHICH MAY INITIATE DURING OVERPRESSURE EVENT

SYSTEM	INITIATING/TRIP SIGNAL(S)
Reactor Protection System	Reactor trips "OFF" on High Flux
RCIC	"ON" when Reactor Water Level <L2
	"OFF" when Reactor Water Level >L8
HPCI	"ON" when Reactor Water Level <L2
	"OFF" when Reactor Water Level >L8
Recirculation System	"OFF" when Reactor Water Level <L2
	"OFF" when Reactor Pressure >1125 psi
RWCU	"OFF" when Reactor Water Level <L2

TABLE 5.2-7

WATER SAMPLE LOCATIONS

Sample Origin	Sensor Location	Indicator Location	Recorder** Location	Range µmho/cm	Alarm	Minimum*** Accuracy
Reactor Water Recirculation Loop	Sample Line	Sample Station	Control Room	0-10*	1.0	0.5%
Reactor Water Cleanup Inlet	Sample Line	Sample Station	Control Room	0-10*	1.0	0.5%
Reactor Water Cleanup Outlet	Sample Line	Sample Station		0.5	0.1	0.5%
Control Rod Drive Supply Water	Sample Line	Sample Station		0.1*	0.1	0.5%

* The instrument has auto output scaling change : when selected shifts the output scaling by a factor of 10 when the measured value is less than 10% of the selected output scaling range high limit.

** The output of each sample analyzer is recorded by the Water Chemistry Data Acquisition System.

*** The accuracy is expressed as percent of full scale range. The instruments are sensitive to within or less than the accuracy, and are periodically (1/week) calibrated against laboratory calibration instruments.

Table 5.2-8
SUMMARY OF ISOLATION/ALARM OF SYSTEM MONITORED
AND THE LEAK DETECTION METHODS USED

Security-Related Information
Table Withheld Under 10 CFR 2.390

Table 5.2-9

SEQUENCE OF EVENTS FOR MSIV ISOLATION CLOSURE (TYPICAL)

Security-Related Information

Table Withheld Under 10 CFR 2.390

TABLE 5.2-10

RCPB COMPONENTS IN COMPLIANCE WITH 10CFR50.55(a) (2) (ii)

COMPONENT	CODE APPLIED	CODE REQUIRED BY 10CFR50.55 (a)
*RPV	70S	71S
Recirculation Piping	71S	72S
Recirculation Pumps	71S	71W
MSIV	71S	71W
MSRV	71S	71W

*RPV 10CFR50.55 (a) (2) (ii) compliance is based on the following:

(a) Meeting the requirements of Section NB-2152 of the 1971 edition including the Summer 1971 Addenda of the ASME Code Section III for the work at the SSES site.

(b) Meeting the requirements of Section NB-2400 of the 1971 edition including the Summer 1971 Addenda of the ASME Code Section III for work at the SSES site.

(c) Performing on ASME site audit as required by the 1971 edition of the ASME Code Section III.

Table 5.2-11

IDENTIFIED LEAKAGES INTO THE DRYWELL EQUIPMENT DRAIN TANK

Security-Related Information
Table Withheld Under 10 CFR 2.390

Table 5.2-12
UNIDENTIFIED LEAKAGES INTO THE DRYWELL FLOOR DRAIN SUMP

Security-Related Information
Table Withheld Under 10 CFR 2.390

Table 5.2-13
ESTIMATED MONITOR RESPONSES

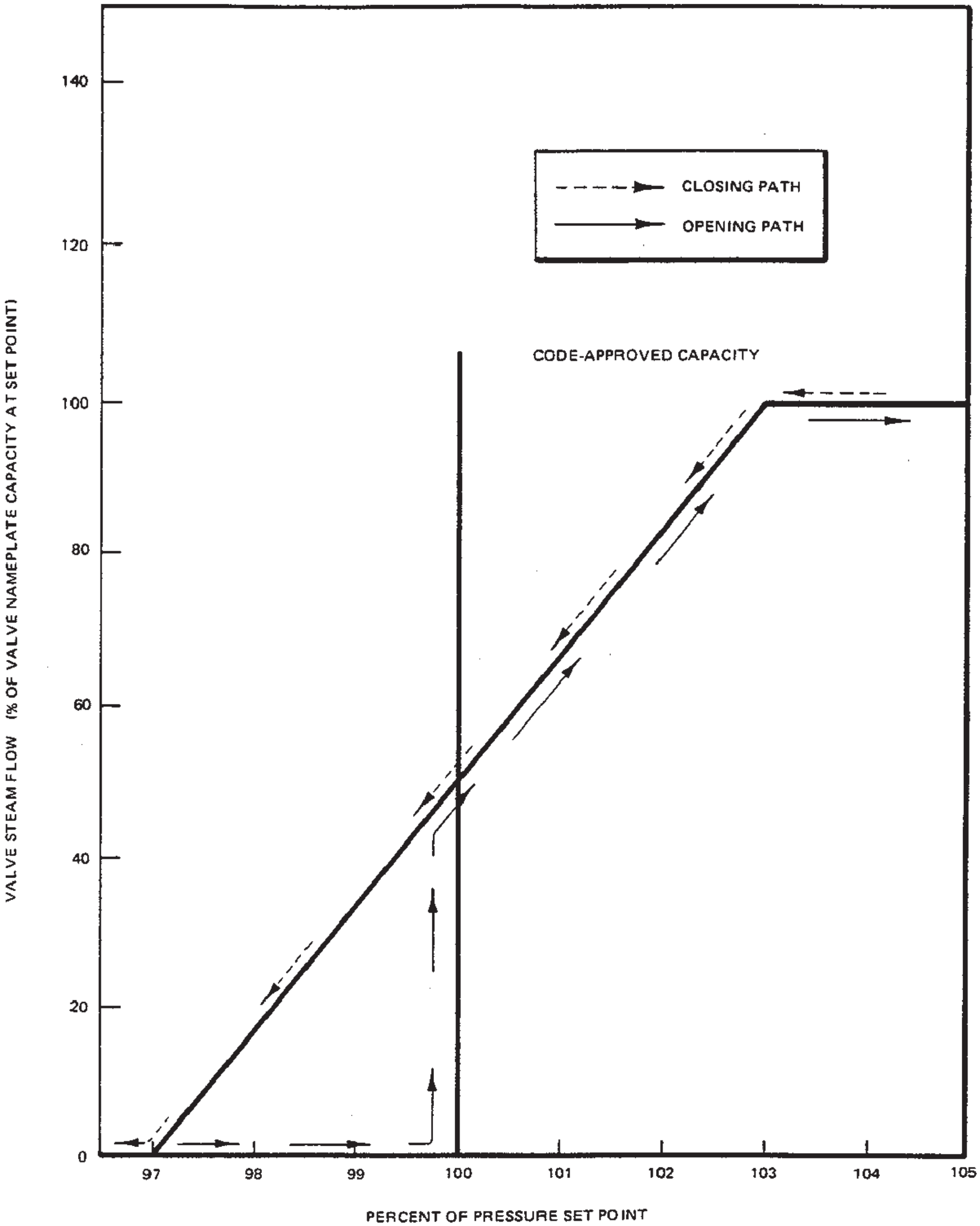
Security-Related Information
Table Withheld Under 10 CFR 2.390

Table 5.2-14

RCPB LEAK DETECTION MONITORS INSIDE

PRIMARY CONTAINMENT DRYWELL

Security-Related Information
Table Withheld Under 10 CFR 2.390



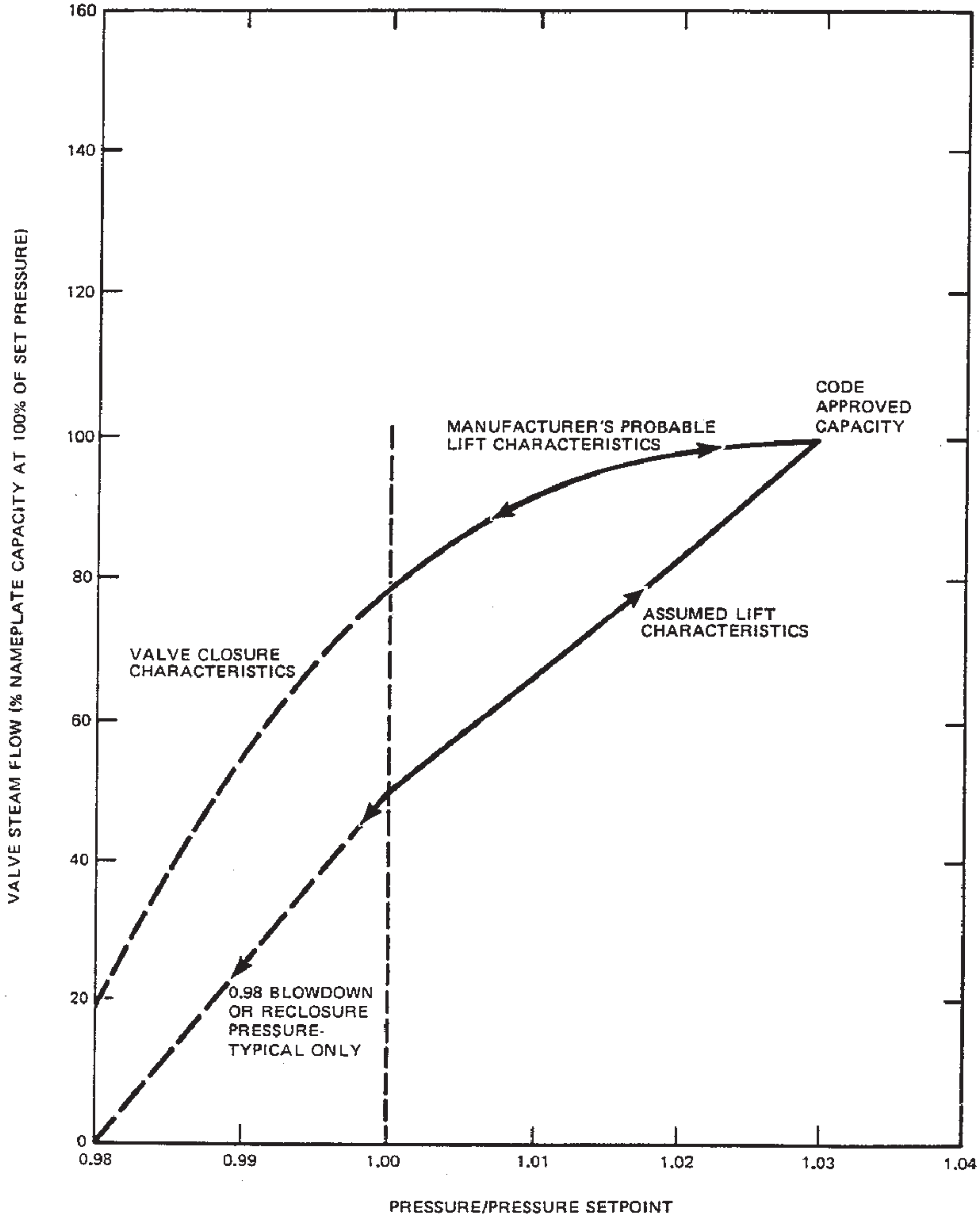
FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 AND 2
FINAL SAFETY ANALYSIS REPORT

SIMULATED SAFETY/RELIEF VALVE
SPRING MODE CHARACTERISTIC
USED FOR CAPACITY SIZING
ANALYSIS

FIGURE 5.2-2, Rev. 49

Atuo Cad: Figure Fsar 5_2_2.dwg



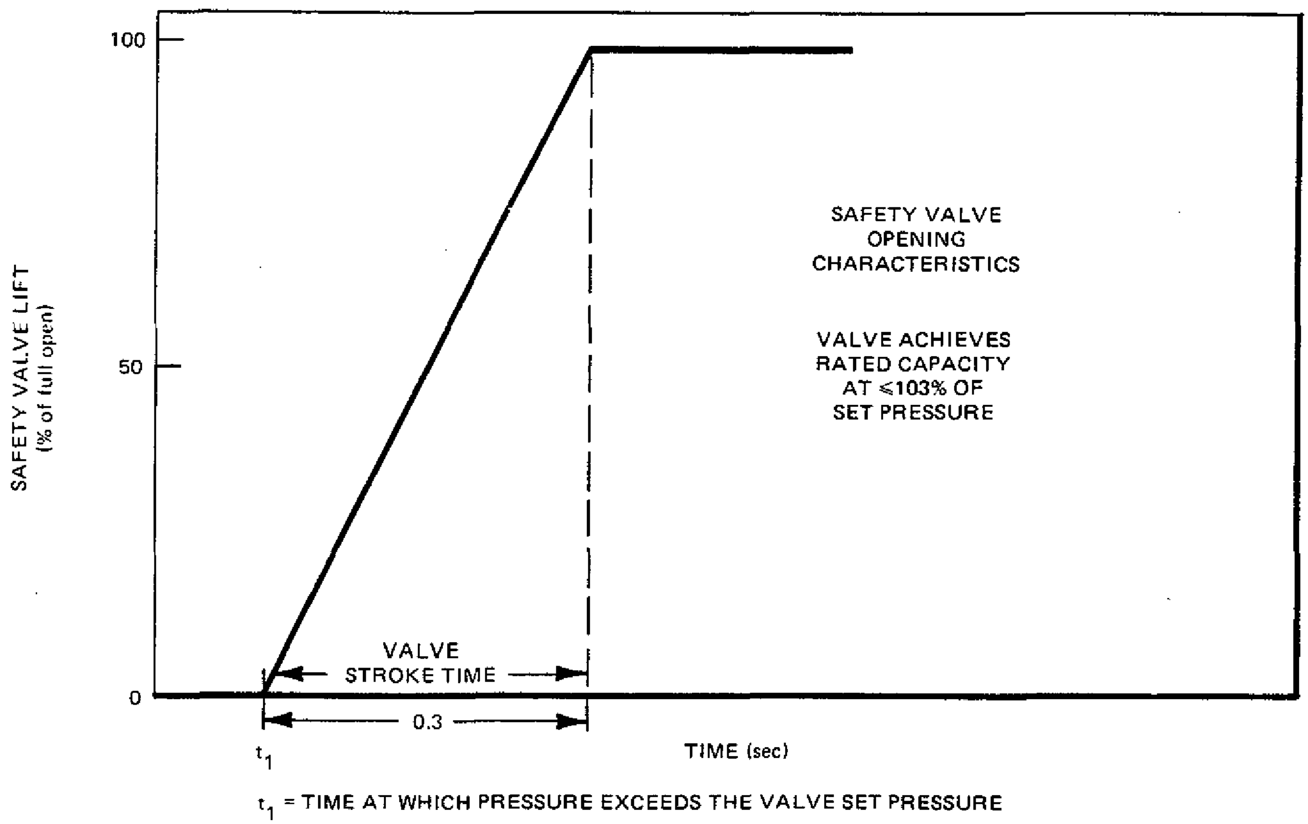
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

SIMULATED SAFETY/RELIEF VALVE
SPRING MODE
CHARACTERISTIC

FIGURE 5.2-2A, Rev.49

AutoCAD: Figure Fsar 5_2_2A.dwg



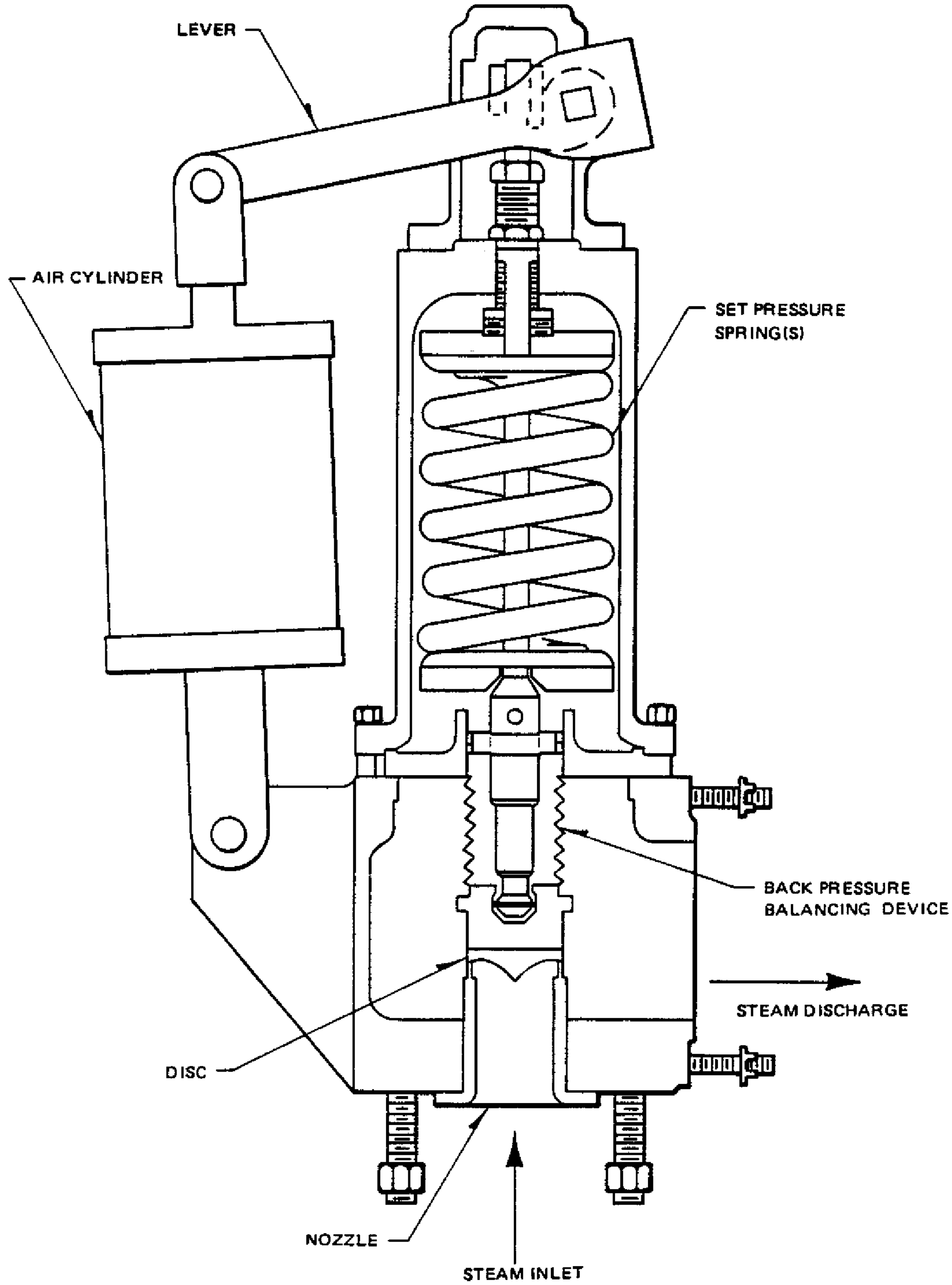
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

SAFETY VALVE LIFT
VERSUS
TIME CHARACTERISTIC

FIGURE 5.2-6, Rev.49

AutoCAD: Figure Fsar 5_2_6.dwg



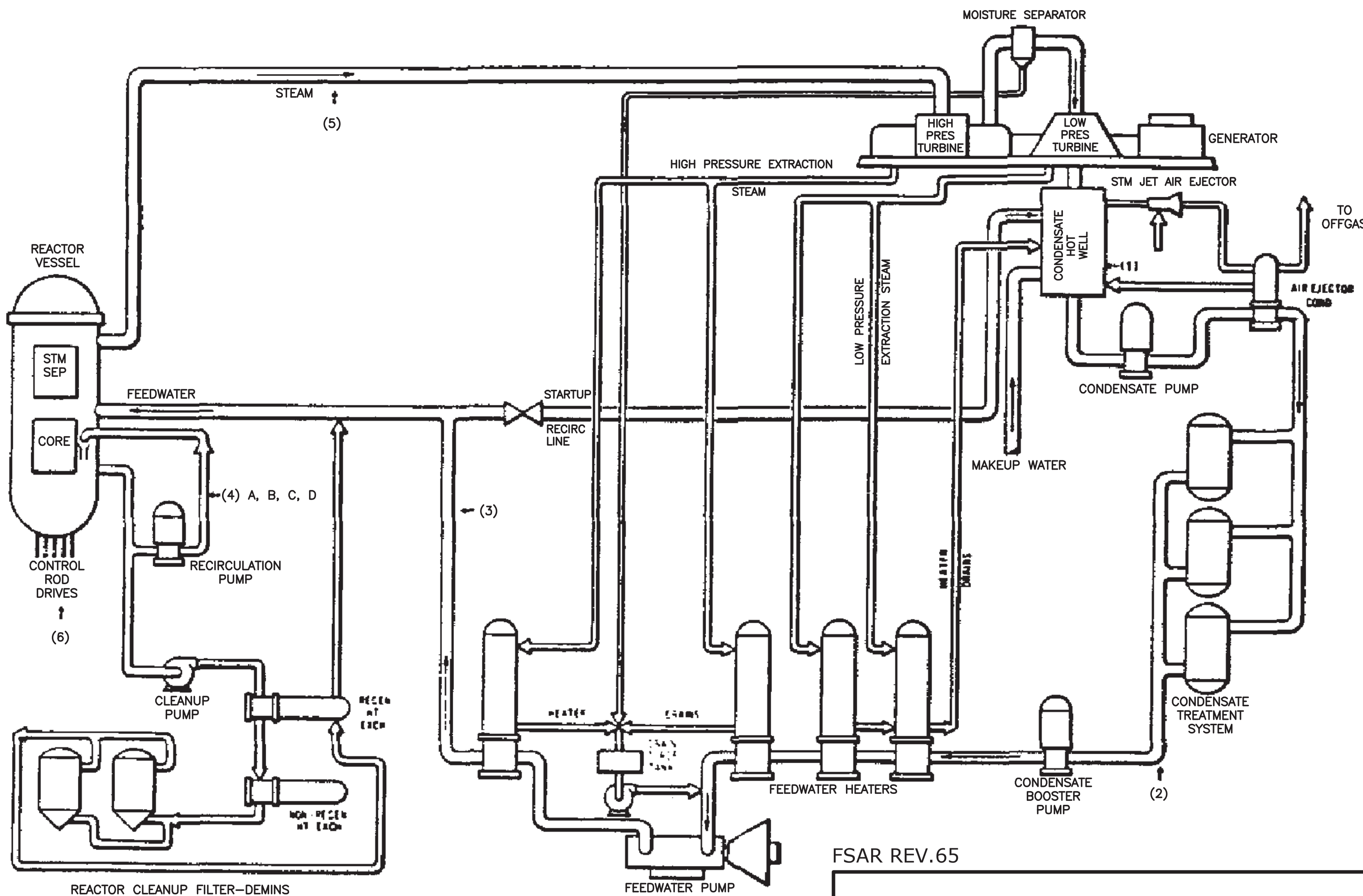
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

SCHEMATIC OF SAFETY VALVE
WITH
AUXILIARY ACTUATING DEVICE

FIGURE 5.2-7, Rev.49

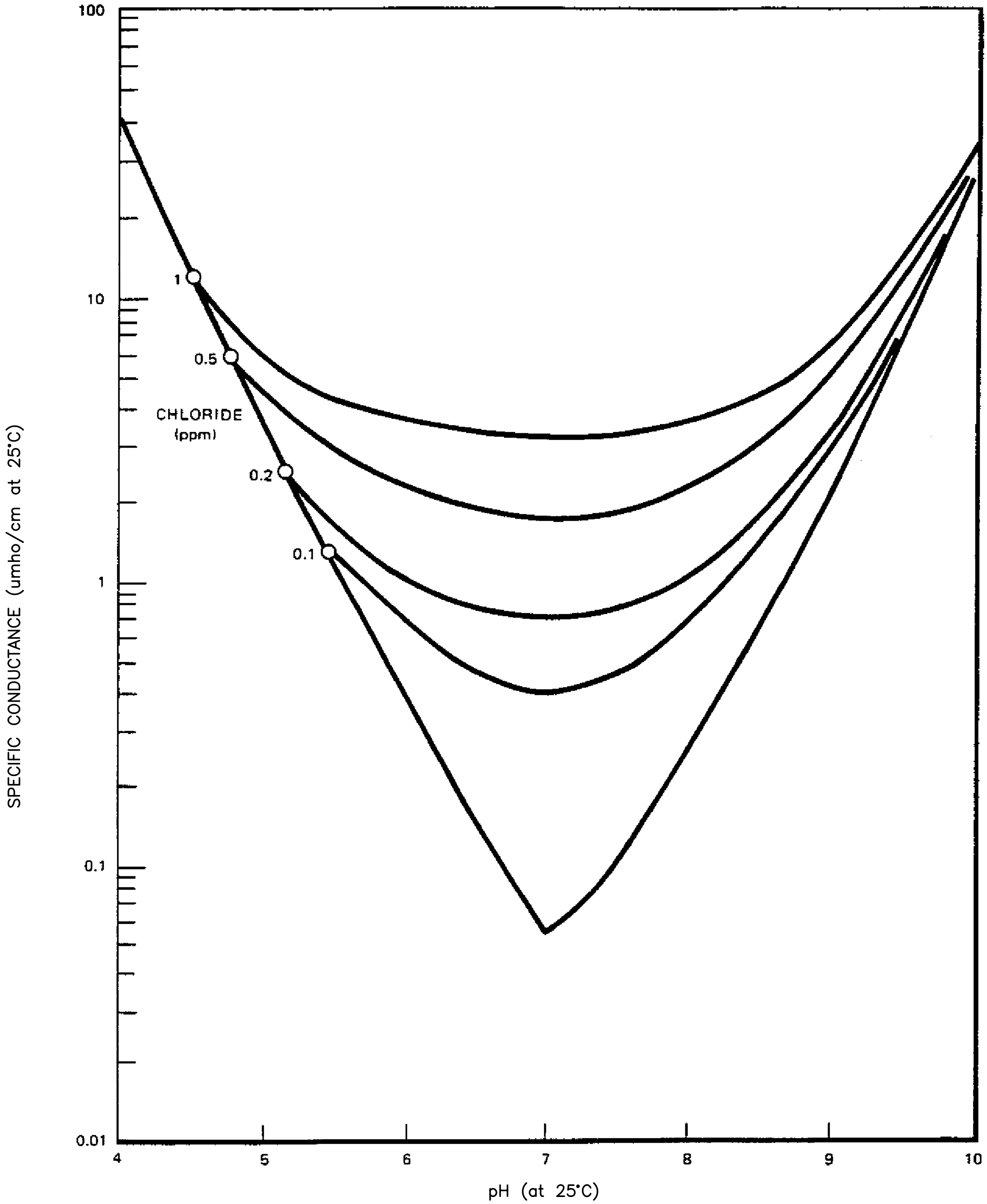
AutoCAD: Figure Fsar 5_2_7.dwg



FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
TYPICAL BWR FEEDWATER CYCLE FIGURE 5.2-8, Rev.49

AutoCAD: Figure Fsar 5_2_8.dwg



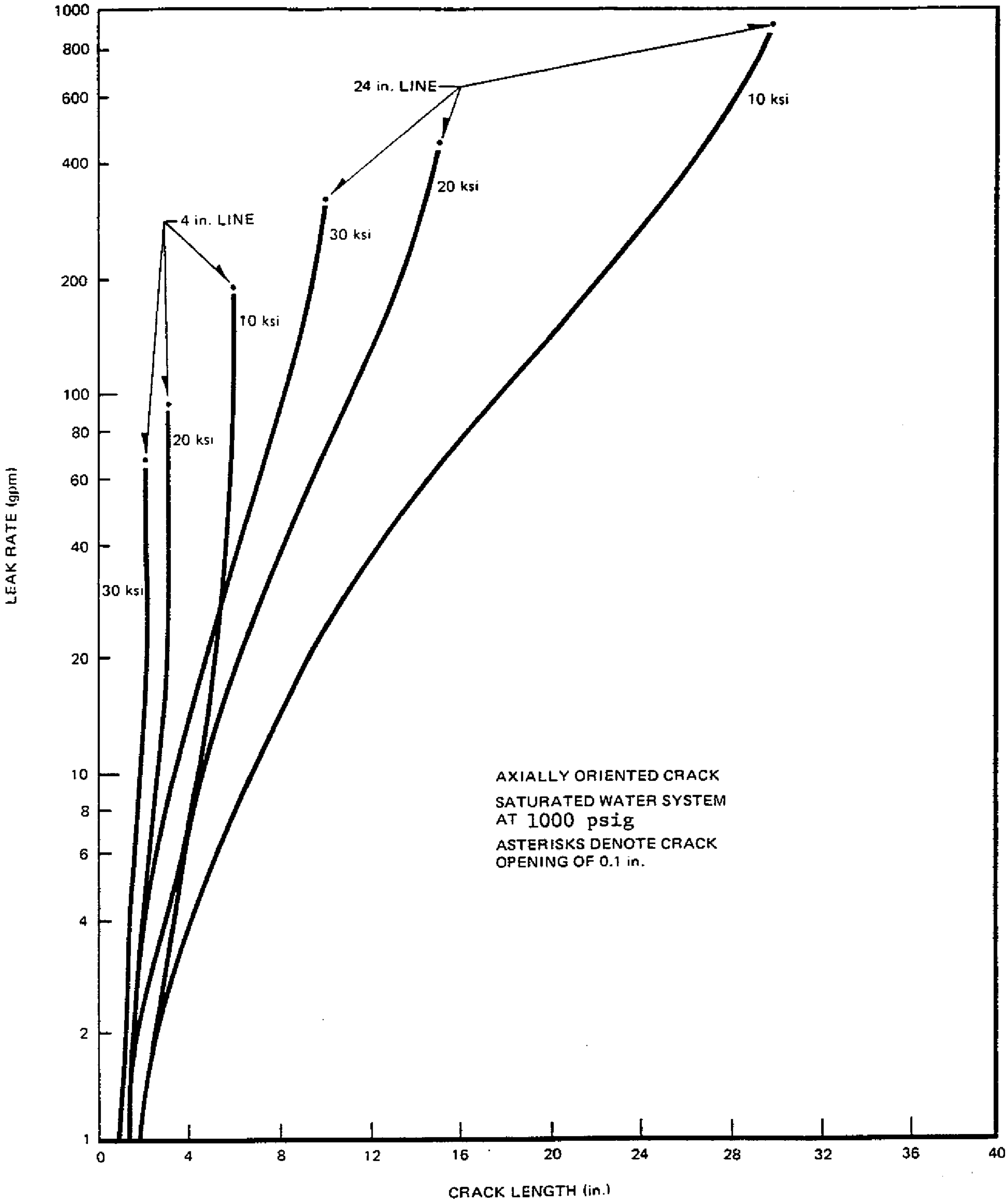
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

CONDUCTANCE VS. PH AS A
FUNCTION OF CHLORIDE
CONCENTRATION OF AQUEOUS
SOLUTION AT 25° C

FIGURE 5.2-9, Rev.49

AutoCAD: Figure Fsar 5_2_9.dwg



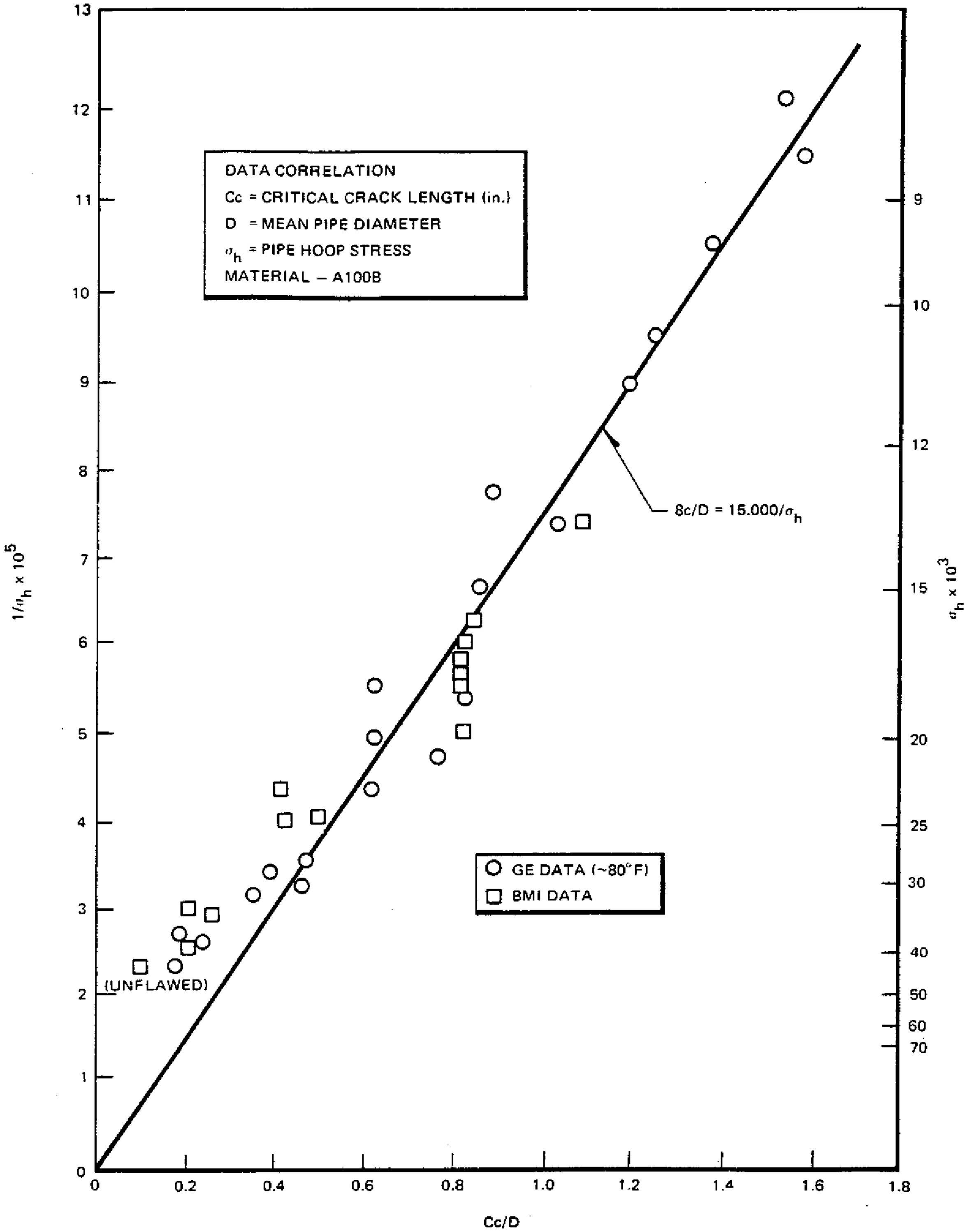
FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 AND 2
FINAL SAFETY ANALYSIS REPORT

CALCULATED LEAK RATE VERSUS
CRACK LENGTH AS A FUNCTION
OF APPLIED HOOP STRESS

FIGURE 5.2-10, Rev. 49

Atuo Cad: Figure Fsar 5_2_10.dwg



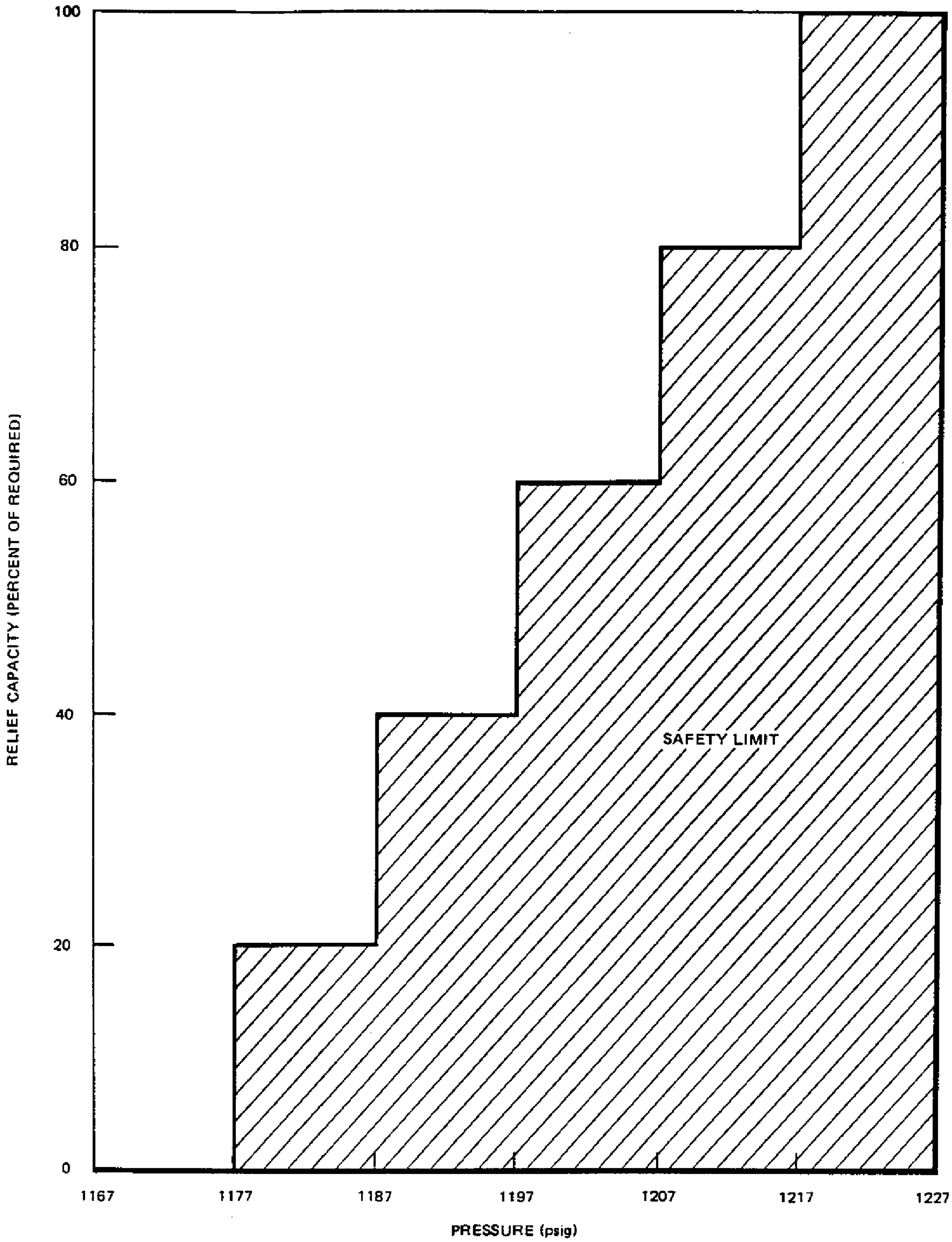
FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 AND 2
FINAL SAFETY ANALYSIS REPORT

AXIAL THROUGH-WALL
CRACK LENGTH
DATA CORRELATION

FIGURE 5.2-11, Rev. 49

Atuo Cad: Figure Fsar 5_2_11.dwg



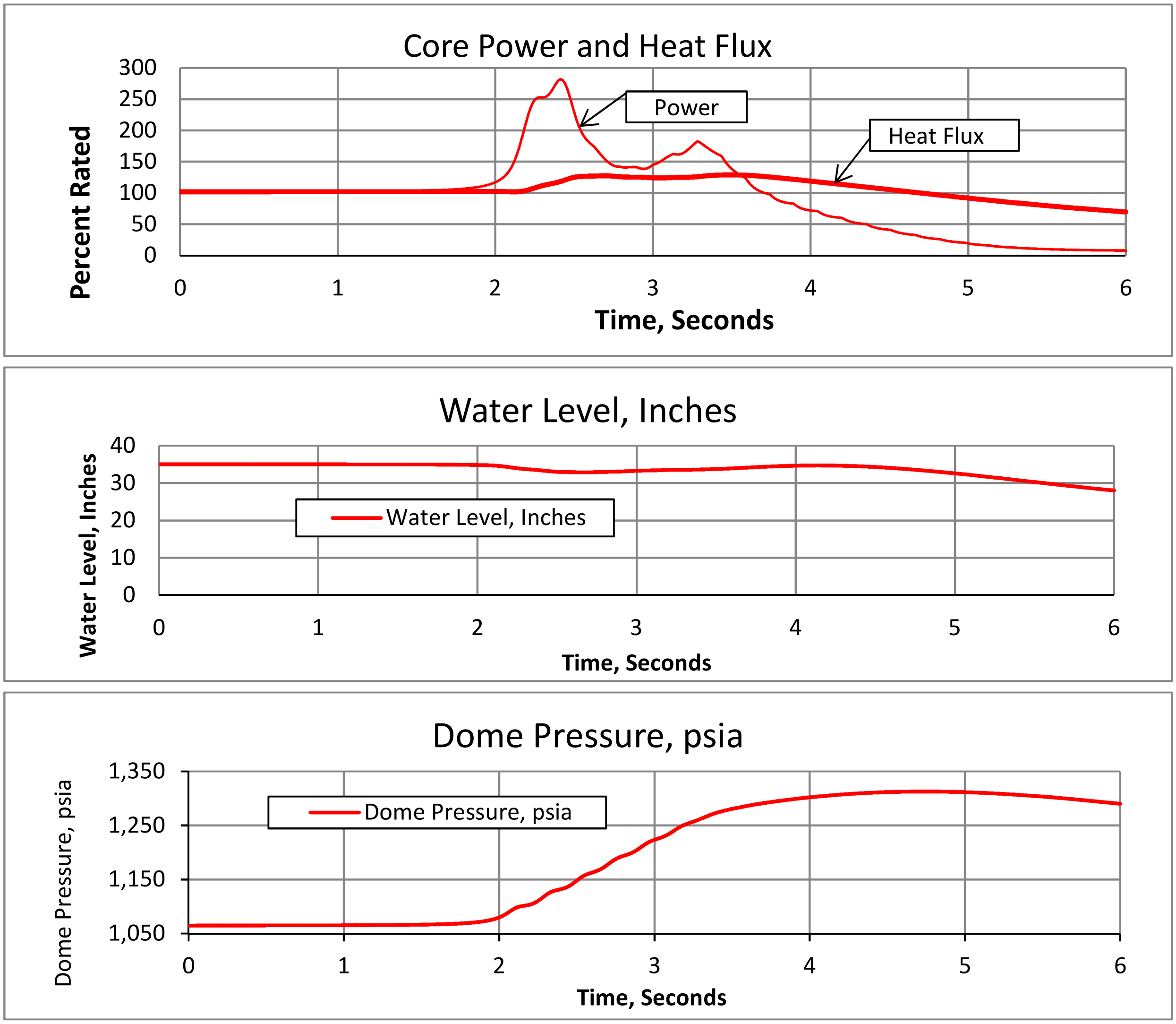
FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 AND 2
FINAL SAFETY ANALYSIS REPORT

REGION FOR SPRING SAFETY
MODE NOMINAL SETPOINT

FIGURE 5.2-12, Rev. 49

Atuo Cad: Figure Fsar 5_2_12.dwg

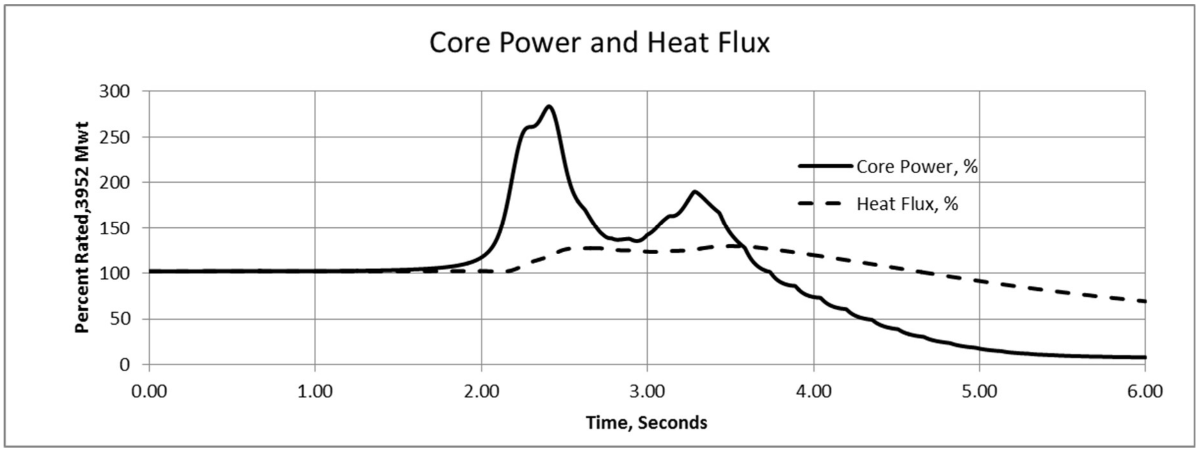


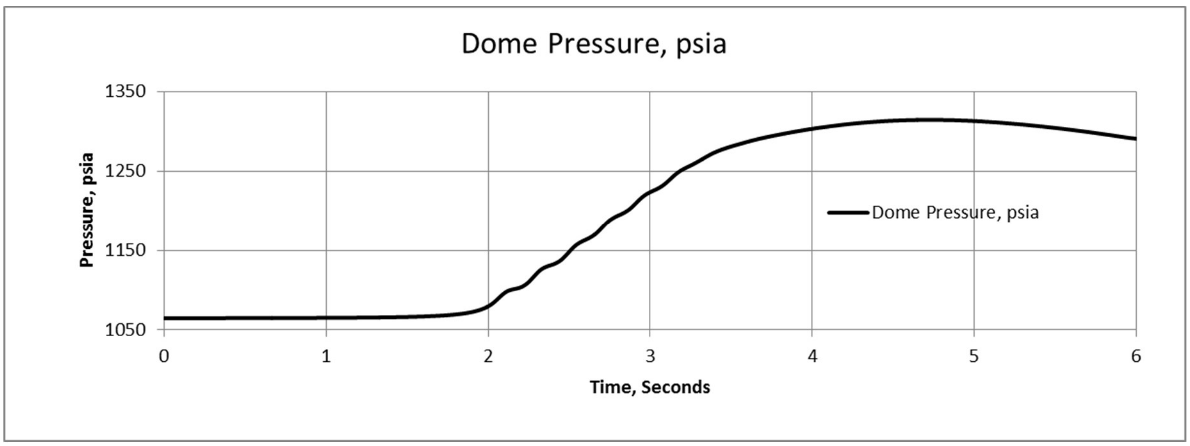
FSAR REV. 69

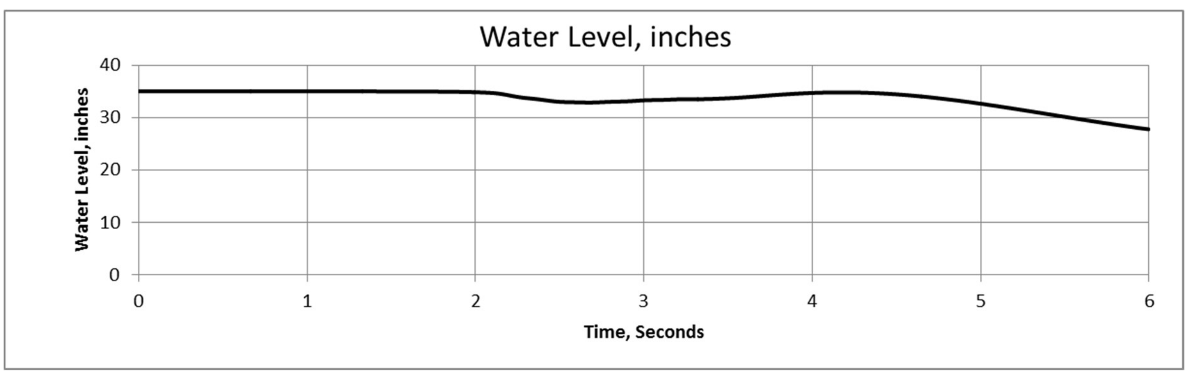
SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 AND 2
FINAL SAFETY ANALYSIS REPORT

OVERPRESSURE PROTECTION ANALYSIS
(MSIV CLOSURE WITH HIGH FLUX SCRAM TRIP)
TYPICAL OF UNIT 1

Figure 5.2-13, Rev 64







FSAR REV. 69

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 AND 2
FINAL SAFETY ANALYSIS REPORT

OVERPRESSURE PROTECTION ANALYSIS
(MSIV CLOSURE WITH HIGH FLUX SCRAM TRIP)
TYPICAL OF UNIT 2

FIGURE 5.2-14, Rev 66

5.3 REACTOR VESSEL

5.3.1 Reactor Vessel Materials

5.3.1.1 Materials Specifications

The materials used in the reactor pressure vessel and appurtenances are shown in Table 5.2-4 together with the applicable specifications.

5.3.1.2 Special Processes Used for Manufacturing and Fabrication

The reactor pressure vessel is primarily constructed from low alloy, high strength steel plates and forgings. Plates are ordered to ASME SA 533 Grade B, Class 1, and forgings to ASME SA 508, Class 2. These materials are melted to fine grain practice and are supplied in the quenched and tempered condition. Further restrictions include a requirement for vacuum degassing to lower the hydrogen level and improve the cleanliness of the low alloy steels.

Studs, nuts, and washers for the main closure flange are ordered to the requirements on Table 5.2-4. Welding electrodes are low hydrogen type ordered to ASME SA 316.

All plates, forgings, and boltings are 100% ultrasonically tested and surface examined by magnetic particle methods or liquid penetrant methods in accordance with ASME Section III, Subsection NB standards. Fracture toughness properties are also measured and controlled in accordance with subsection NB requirements.

All fabrication of the reactor pressure vessel is performed in accordance with GE approved drawings, fabrication procedures, and test procedures. The shells and vessel heads are made from formed plates, and the flanges and nozzles from forgings. Welding performed to join these vessel components is in accordance with procedures qualified per ASME Section III and IX requirements. Weld test samples are required for each procedure for major vessel full penetration welds. Tensile and impact tests are performed to determine the properties of the base metal, heat affected zone and weld metal.

Submerged arc and manual stick electrode welding processes are employed. Electroslag welding is not permitted. Preheat and interpass temperatures employed for welding of low alloy steel meet or exceed the requirements of ASME Section III, Subsection NA. Post weld heat treatment at 1100°F minimum is applied to all low alloy steel welds.

Radiographic examination is performed on all pressure containing welds in accordance with Paragraph N624 of the 1968 ASME Section III Code including 1970 addenda. In addition, all welds are given a supplemental ultrasonic examination.

The materials, fabrication procedures, and testing methods used in the construction of BWR reactor pressure vessels meet the requirements of ASME Section III Class I vessels, 1968 Edition with Summer 1970 Addenda. Paragraph NB-338.2(d)(4) of the Winter 1971 Addenda shall supercede Paragraph I-613(d) of the 1968 Edition, and Paragraph NB-2400 of the 1971 Edition shall apply for all fabrication performed at the Susquehanna site.

5.3.1.3 Special Methods for Nondestructive Examination

The materials and welds on the reactor pressure vessel were examined in accordance with methods prescribed and met the acceptance requirements specified by ASME Boiler and Pressure Vessel Code, Section III. In addition, the pressure retaining welds were ultrasonically examined using manual techniques. The ultrasonic examination method, including calibration, instrumentation, scanning sensitivity, and coverage was based on the requirements imposed by ASME Code, Section XI in Appendix I. Acceptance standards were equivalent or more restrictive than required by ASME Code, Section XI.

5.3.1.4 Special Controls For Ferritic and Austenitic Stainless Steels

5.3.1.4.1 Compliance With Regulatory Guides

5.3.1.4.1.1 Regulatory Guide 1.31, (Rev 1) Control of Stainless Steel Welding

Controls on stainless steel welding are discussed in Subsection 5.2.3.4.2.1

5.3.1.4.1.2 Regulatory Guide 1.34, (12/72) Control of Electroslag Weld Properties

Electroslag welding was not employed for the reactor pressure vessel fabrication.

5.3.1.4.1.3 Regulatory Guide 1.43, (5/73) Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components

Reactor pressure vessel specifications require that all low alloy steel be produced to fine grain practice. Regulatory Guide 1.43 applies to RPV steels that have been manufactured to coarse grain steel making practice. The SSES vessels were manufactured to fine grain steel making practice; therefore, this regulatory guide does not apply to the SSES vessels.

5.3.1.4.1.4 Regulatory Guide 1.44, (5/73) Control of the Use of Sensitized Stainless Steel

Controls to avoid severe sensitization are discussed in Subsection 5.2.3.4.1.1.

5.3.1.4.1.5 Regulatory Guide 1.50 (5/73), Control of Preheat Temperature for Welding Low-Alloy Steel

Preheat controls are discussed in Subsection 5.2.3.3.2.1.

5.3.1.4.1.6 Regulatory Guide 1.71, (12/73) Welder Qualification for Areas of Limited Accessibility

Qualification for areas of limited accessibility is discussed in Subsection 5.2.3.3.2.3.

5.3.1.4.1.7 Regulatory Guide 1.99, (Rev. 2) Radiation Embrittlement of Reactor Vessel Materials

Predictions for changes in transition temperature and upper shelf energy were made in accordance with the requirements of Regulatory Guide 1.99, Revision 2.

5.3.1.5 Fracture Toughness

5.3.1.5.1 Compliance with 10CFR50 Appendix G

A major condition necessary for full compliance to Appendix G is satisfaction of the requirements of the Summer 1972 Addenda to Section III. This is not possible with components which were purchased to earlier Code requirements. For the extent of compliance, see Tables 5.3-1a and 5.3-2a.

Ferritic material complying with 10 CFR 50, Appendix G, must have both drop weight tests and Charpy V-notch (CVN) tests with the CVN specimens oriented transverse to the principal material working direction to establish the RT_{NDT}. The CVN tests must be evaluated against both an absorbed energy and lateral expansion criteria. The maximum acceptable RT must be determined in accordance with the analytical procedures of ASME Code Section III, Appendix G. Appendix G of 10 CFR 50 requires a minimum of 75 ft-lbs upper shelf CVN energy for beltline material. It also requires at least 45 ft-lbs CVN energy and 25 mils lateral expansion for bolting material at the lower of the preload or lowest service temperature.

By comparison, material for the Susquehanna SES reactor vessels was qualified by either drop weight tests and/or longitudinally oriented CVN tests (both not required), confirming that the material nil-ductility transition temperature (NDT) is at least 60°F below the lowest service temperature. When the CVN test was applied, a 30 ft-lbs energy level was used in defining the NDT. There was no upper shelf CVN energy requirement on the beltline material. The bolting material was qualified to a 30 ft-lbs energy requirement at 60°F below the minimum preload temperature.

From the previous comparison it can be seen that the fracture toughness testing performed on the SSES reactor vessel material in some cases cannot be shown to comply with 10 CFR 50, Appendix G. However, to determine operating limits in accordance with 10 CFR 50, Appendix G, estimates of the beltline material RT_{NDT} and the highest RT_{NDT} of all other material were made, as explained in Subsection 5.3.1.5.1.2. The method for developing these operating limits is also described therein.

On the basis of the last paragraph on page 19013 of the July 17, 1973, Federal Register, the following is considered an appropriate method of compliance.

5.3.1.5.1.1 Intent of Proposed Approach

The intent of the proposed special method of compliance with Appendix G for this vessel is to provide operating limitations on pressure and temperature based on fracture toughness. These operating limits assure that a margin of safety against a nonductile failure of this vessel is nearly the same as that for a vessel built to the Summer 1972 Addenda.

The specific temperature limits for operation when the core is critical are based on 10 CFR 50, Appendix G, January 1998.

5.3.1.5.1.2 Operating Limits Based on Fracture Toughness

Operating limits which define minimum reactor vessel metal temperatures vs reactor pressure during normal heatup and cooldown and, during in-service hydrostatic testing, were established using the methods of Appendix G of Section XI of the ASME Boiler and Pressure Vessel Code, 1992 Edition. The results are shown in Technical Specification Figures 3.4.10-1, 3.4.10-2, and 3.4.10-3 for each Unit. These figures have been modified and updated based on surveillance capsule tests that were performed after the first specimens were removed from the reactors and tested after 6 EFPY and calculations based on ASME Code Cases N-588 and N-640.

Estimated RT_{NDT} values and temperature limits are given in this section for the limiting locations in the reactor vessel.

All the vessel shell and head areas remote from discontinuities, all other shell and head areas, flanges, and the feedwater nozzles were evaluated: the operating limit curves are based on the limiting locations. The boltup limits for the flange and adjacent shell region are based on a minimum metal temperature of RT_{NDT} + 60°. The maximum through-wall temperature gradient from continuous heating or cooling at 100°F per hour was considered. The safety factors applied were as specified in ASME Code, Section XI, Appendix G.

For the purpose of setting these operating limits, the reference temperature, RT_{NDT}, is determined from the toughness test data taken in accordance with requirements of the Code to which this vessel is designed and manufactured. This toughness test data, Charpy V-notch (CVN) and/or drop-weight nil-ductility transition temperature (NDT) is analyzed to permit compliance with the intent of 10 CFR 50, Appendix G. Because all toughness testing needed for strict compliance with Appendix G was not required at the time of vessel procurement, some toughness results are not available. For example, longitudinal CVN's, instead of transverse, were tested, usually at a single test temperature of +10°F or +40°F, for absorbed energy. Also, at the time either CVN or NDT testing was permitted; therefore, in some cases both tests were not performed as is currently required. To substitute for this absence of certain data, toughness property correlations were derived for the vessel materials in order to operate upon the available data to give a conservative estimate of RT_{NDT}, compliant with the intent of Appendix G criteria.

These toughness correlations vary, depending upon the specific material analyzed, and were derived from the results of WRC Bulletin 217, "Properties of Heavy Section Nuclear Reactor Steels," and from toughness data from the Susquehanna SES vessels and other reactors. In the case of vessel plate material (SA-533 Grade B, Class 1), the predicted limiting toughness property is either NDT or transverse CVN 50 ft-lbs temperature minus 60°F. CVN and NDT data are available for all of the beltline plates. Where NDT results are missing, NDT is estimated as the longitudinal CVN 35 ft-lbs transition temperature. The transverse CVN 50 ft-lbs transition temperature is estimated from longitudinal CVN data in the following manner. The lowest longitudinal CVN ft-lb value is adjusted to derive a longitudinal CVN 50 ft-lbs transition temperature by adding 2°F per ft-lb to the test temperature. If the actual data equal or exceed 50 ft-lbs, the test temperature is derived by interpolation or conservatively taken as the transition temperature. Once the longitudinal 50 ft-lbs temperature is derived, an additional

30°F is added to account for orientation effects and to estimate the transverse CVN 50 ft-lbs temperature minus 60°F, as described above.

For forgings (SA-508 Class 2), the predicted limiting property is the same as for vessel plates. CVN and NDT values are available for the vessel flange, closure head flange, and feedwater nozzle materials for Susquehanna SES. RT_{NDT} is estimated in the same way as for vessel plates.

For the vessel weld metal, the predicted limiting property is the CVN 50 ft-lbs transition temperature minus 60°F, as the NDT values are -50°F or lower for these materials. This temperature is derived in the same way as for the vessel plate material, except the 30°F addition for orientation effects is omitted since there is no principal working direction. When NDT values are available, they are also considered and the RT_{NDT} is taken as the higher of NDT or the 50 ft-lbs temperature minus 60°F. When NDT is not available, the RT_{NDT} shall not be less than -50°F, since lower values are not supported by the correlation data.

For vessel weld heat affected zone (HAZ) material, the RT_{NDT} is assumed the same as for the base material as ASME Code weld procedure qualification test requirements, and post weld heat treatment indicates this assumption is valid.

Closure bolting material (SA-540 Grade B24) toughness test requirements for Units 1 and 2 were for 30 ft-lbs at 60°F below the bolt-up temperature. Current Code requirements are for 45 ft-lbs and 25 mils lateral expansion at the preload or lowest service temperature, including bolt-up. The reactor vessel closure studs for Unit 1 have a minimum Charpy impact energy of 40 ft-lbs and 25 mils lateral expansion at 10°F. The lowest service temperature for the closure studs is 70°F for Unit 1. For Unit 2, the closure studs have a minimum Charpy impact energy of 48 ft-lbs and 27 mils lateral expansion at 10°F; therefore, the lowest service temperature for the Unit 2 closure studs is +10°F.

Using the above general approach, an initial RT_{NDT} of +18°F was established for the core beltline region for Unit 1 and +10°F for Unit 2.

The effect of the main closure discontinuity was considered by adding 60°F to the RT_{NDT} to establish the minimum temperature for boltup and pressurization. The minimum bolt-up temperature of +70°F for Units 1 and 2, which is required in Technical Bases 3.4.10 and is shown on Figures 3.4.10-1 through 3.4.10-3 in the Technical Specification for each Unit, is based on an initial RT_{NDT} of +10°F for the closure flange forgings.

The effect of the vessel nozzle and bottom head discontinuities is considered by developing separate curves for the bottom head and non-beltline regions. These separate curves utilized the appropriate Susquehanna SES nozzle forging and bottom head RT_{NDT}'s. For Unit 1, the controlling discontinuity limits are based on the recirculation inlet nozzles (RT_{NDT}=40°F) for feedwater nozzle limits and the bottom head penetrations (RT_{NDT}=34°F) for the CRD penetration limits. For Unit 2, the controlling discontinuity limits are based on the steam outlet nozzles (RT_{NDT}=30°F) for the feedwater nozzle limits and the recirculation outlet nozzle (RT_{NDT}=24°F) for the CRD penetration limits.

5.3.1.5.1.3 Operating Limits During Heatup, Cooldown and Core Operation

The fracture toughness analysis was done for the normal heatup or cooldown rate of 100°F/ hour. The temperature gradients and thermal stress effects corresponding to this rate were included. The results of the analyses are a set of operating limits for non-nuclear heatup or cooldown shown as curves labeled B on Figures 3.4.10-1 in Technical Specifications for each Unit (Reference 5.3-1, Task A-15). Curves labeled C on Technical Specification Figures 3.4.10-3 apply whenever the core is critical.

5.3.1.5.1.4 Temperature Limits For ISI Hydrostatic or Leak Pressure Tests

The fracture toughness analysis for in-service inspection or leak pressure tests resulted in the curves in Technical Specification Figure 3.4.10-1 for each Unit.

5.3.1.5.1.5 Adjusted Reference Temperature for Limiting Core Beltline Material

Adjusted reference temperature (ART) is a prediction of the effect of fast neutron fluence on RT_{NDT} at 1/4 of the vessel wall thickness. The applicable shift in RT_{NDT} including applicable effects of power uprate, is added to the pressure-temperature curve of each of the limiting beltline materials to produce the Unit 1 and 2 "Core Beltline" curves in Technical Specification Figures 3.4.10-1, 3.4.10-2, and 3.4.10-3.

The limiting beltline material for Unit 1 is plate heat C2433-1, which has an initial RT_{NDT} of +18°F and a 35.7-EFPY ART of 61.4°F (References 5.3-2 and 5.3-6).

The limiting beltline material for Unit 2 is plate heat C2421-3, which has an initial RT_{NDT} of -10°F and a 30.2-EFPY ART of 46.7°F (References 5.3-3 and 5.3-6).

5.3.1.5.1.6 Temperature Limits for Boltup

A minimum temperature of 70° for Unit 1 and of 10° for Unit 2 is required for closure studs. A sufficient number of studs may be tensioned at 70°F to seal the closure flange O-rings for the purpose of raising reactor water level above the closure flanges in order to assist in warming them. The flanges and adjacent shell are required to be warmed to a minimum temperature of 70°F before they are stressed by the full intended bolt preload. The fully preloaded bolt-up limits are shown on Technical Specification Figures 3.4.10-1, 3.4.10-2, and 3.4.10-3 for Unit 1 and Unit 2.

5.3.1.5.1.7 Reactor Vessel Annealing

In-place annealing of the reactor vessel because of radiation embrittlement is unnecessary because the predicted end of life value of adjusted reference temperature will not exceed 200°F (see 10 CFR 50, Appendix G, Paragraph IV.C).

5.3.1.6 Material Surveillance

5.3.1.6.1 Compliance with "Reactor Vessel Material Surveillance Program Requirements"

The materials surveillance program monitors changes in the fracture toughness properties of ferritic materials in the reactor vessel beltline region resulting from their exposure to neutron irradiation and thermal environment.

Materials for the program are selected to represent materials used in the reactor beltline region. The specimens are manufactured from a plate actually used in the beltline region and a weld typical of those in the beltline region and thus represent base metal, weld metal, and the transition zone between base metal and weld. The plate and weld are heat treated in a manner which simulates the actual heat treatment performed on the core region shell plates of the completed vessel.

The surveillance program includes three capsule holders per reactor vessel. Charpy impact specimens for the reactor specimens for the reactor vessel surveillance programs are of the longitudinal orientation consistent with the ASME requirements prior to the issuance of the Summer 1972 Addenda and ASTM-E-185-82. Based on GE experience, the amount of shift measured by these irradiated logitudinal test specimens will be essentially the same as the shift in an equivalent transverse specimen.

The program for implementation of the scheduling and testing of the surveillance specimen is governed and controlled by the BWR Integrated Surveillance Program (ISP) [References 5.3-7 through 5.3-10]. The Unit 1 second holder (131C7717(G2) will be pulled in accordance with the schedule in the ISP. For Unit 2, all the information will come from the other plants in the Integrated Surveillance Program. No capsules are scheduled to be withdrawn from Unit 2. Other plants will remove and test specimens in accordance with the ISP. The results from these test will provide the necessary data to monitor embrittlement for Unit 2. Since the predicted adjusted reference temperature of the reactor vessel beltline steel is less than 100°F at end of life, the use of the capsules per the ISP meets the requirements of 10 CFR 50, Appendix H, and ASTM-E-185-82. The withdrawl schedule and other requirements are provided in the ISP.

For the extent of compliance to 10 CFR 50, Appendix H, see Tables 5.3-1 b and 5.3-2b.

Each holder is loaded with capsules which contain the following surveillance specimens and dosimeter wires:

First holder (131C7717G3):

36 Charpy impact specimens including 12 base metal, 12 weld metal, and 12 heat affected zone metal specimens; 10 tensile specimens including 3 base metal, 4 weld metal, and 3 weld heat affected zone metal specimens; 9 metal wire dosimeters including 3 iron, 3 nickel, and 3 copper.

After the first capsule holders (for both Units 1 and 2) were withdrawn and the specimens tested (see references 5.3-4 and 5.3-5), the broken specimens were remachined as miniature specimens and reloaded in the vessels during the next refueling outages. The contents of the new “reconstituted” capsules (for both Units 1 and 2) are as follows:

2 Charpy specimen packets each containing 12 Charpy specimens – 1 packet for base metal specimens and 1 for weld metal specimens. (EXCEPTION: The Unit 1 weld metal capsule only has 11 specimens).

Copper, Iron and Niobium flux wires are included in the capsules with the Charpy specimens.

2 tensile specimen tubes – 1 containing one tensile capsule with four 0.113 inch diameter miniature tensile specimens, the other containing 1 capsule with one 0.113 inch diameter miniature tensile specimen and one 0.250 inch diameter original weld metal tensile specimen.

The new holders have the same geometry as the original capsule holders.

Second holder (131C7717G2):

24 Charpy impact specimens including 8 base metal, 8 weld metal, and 8 weld heat affected zone metal specimens; 8 tensile specimens including 3 base metal, 3 weld metal, and 2 weld heat affected zone metal specimens; 6 metal wire dosimeters including 2 iron, 2 nickel, and 2 copper.

Third holder (131C7717G1):

24 Charpy impact specimens including 8 base metal, 8 weld metal and 8 weld heat affected zone metal specimens; 6 tensile specimens including 2 base metal, 2 weld metal, and 2 weld heat affected zone metal specimens; 6 metal wire dosimeters including 2 iron, 2 nickel, and 2 copper.

A set of out-of-reactor baseline Charpy V-notch specimens is provided with the surveillance test specimens.

5.3.1.6.2 Neutron Flux and Fluence Calculations

A description of the methods of analysis is contained in Subsections 4.1.4.5 and 4.3.2.8.

5.3.1.6.3 Positioning of Surveillance Capsules and Method of Attachment

Surveillance specimen capsules are located at three azimuths at a common elevation in the core beltline region. The sealed capsules are not attached to the vessel but are in welded capsule holders. The capsule holders are mechanically retained by capsule holder brackets welded to the vessel cladding as shown in Figure 5.3-3. The capsule holder brackets allow the capsule holder to be removed at any desired time in the life of the plant for specimen testing. These brackets are designed, fabricated and analyzed to the requirements of Section III of the ASME Code. A positive spring-loaded locking device is provided to retain the capsules in position throughout any anticipated event during the lifetime of the vessel.

5.3.1.6.4 Time and Number of Dosimetry Measurements

GE has provided neutron dosimetry wires in each of the specimen holders. In addition, one holder in each vessel is designed with a separately removable dosimeter, to be removed after one fuel cycle. The first cycle dosimeter was removed from Unit 1 in 1986 and analyzed. A first cycle dosimeter was not available for removal from Unit 2. However, the first cycle dosimetry

for Unit 1 provides a good estimate of flux for Unit 2, because vessel geometries and core power shapes are very similar.

The first cycle dosimetry provides a means of calibrating the flux distribution calculations to actual vessel conditions. Dosimetry will be updated as holders are removed and tested. The holder withdrawal schedule is listed in Table 5.3-3.

5.3.1.7 Reactor Vessel Fasteners

The reactor vessel closure head (flange) is fastened to the reactor vessel shell flange by multiple sets of threaded studs and nuts. The lower end of each stud is installed in a threaded hole in its vessel shell flange. A nut and washer are installed on the upper end of each stud. The proper amount of preload can be applied to the studs by a sequential tensioning using hydraulic tensioners. The design and analysis of this area of the vessel is in full compliance with all Section III Class I Code requirements. The material for studs, nuts and washers is SA-540 Grade B23 or B24. The maximum reported ultimate tensile stress for the bolting material was 163,500 psi, which is less than the 170,000 psi limitation in Regulatory Guide 1.65 (10/73). Also the Charpy impact test results for the closure studs are given in Subsection 5.3.1.5.1.2. Hardness tests are performed on all main closure bolting to demonstrate that heat treatment has been properly performed. Studs, nuts, and washers are ultrasonically examined in accordance with Section III, N-322 and the following additional requirements:

(1) Examination is performed after heat treatment and prior to machining threads.

(2) Straight beam examination is performed on 100 percent of each stud. Reference standard for the radial scan is a 1/2-inch diameter flat bottom hole having a depth equal to 10 percent of the material thickness. For the end scan, the reference is a 1/4-inch flat bottom hole having a standard depth of 1/2-inch.

(3) Nuts and washers are examined by angle beam from the outside circumference in both the axial and circumferential directions.

There are no metal platings applied to closure studs, nuts, or washers. A phosphate coating is applied to threaded areas of studs and nuts and bearing areas of nuts and washers to act as a rust inhibitor and to assist in retaining lubricant on these surfaces.

5.3.2 PRESSURE-TEMPERATURE LIMITS

5.3.2.1 Limit Curves

Limits on pressure and temperature for in-service leak and hydrostatic tests, normal operation (including heatings and cooldown), and reactor core operation are shown in Technical Specification Figures 3.4.10-1, 3.4.10-2, and 3.4.10-3. The basis used to determine these limits is described in Subsection 5.3.1.5.1.2.

5.3.2.2 Operating Procedures

By comparison of the pressure vs. temperature limits in Subsection 5.3.2.1 with intended normal operating procedures for the most severe upset transient, it is shown that the limits will not be

exceeded during any foreseeable upset condition. Reactor operating procedures have been established such that actual transients will not be more severe than those for which the vessel design adequacy has been demonstrated. Of the design transients, the upset condition producing the most adverse temperature and pressure condition anywhere in the vessel head and/or shell areas (loss of AC power) yields a minimum fluid temperature of 250°F and a maximum pressure peak of 1218 psig. Scram automatically occurs with initiation of this upset condition, so the applicable operating limits are given by the curves labeled A in Figure 3.4.10-1 in the Technical Specification for each Unit. For a temperature of 250°F, the maximum allowable pressure exceeds 1218 psig for the intended margin against nonductile failure. The maximum transient pressure of 1218 psig is, therefore, within the specified allowable limits.

5.3.3 REACTOR VESSEL INTEGRITY

The reactor vessels were fabricated for General Electric's Nuclear Energy Division by Chicago Bridge and Iron (CB&I), and were subject to the requirements of General Electric's Quality Assurance program.

Assurance was made that measures were established requiring that purchased material, equipment, and services associated with the reactor vessels and appurtenances conform to the requirements of the subject purchase documents. These measures included provisions, as appropriate, for source evaluation and selection, objective evidence of quality furnished, inspection at the vendor source, and examination of the completed reactor vessels.

General Electric provided inspection surveillance of the reactor vessel fabricator's in process manufacturing, fabrication, and testing operations in accordance with GE's Quality Assurance program and approved inspection procedures. The reactor vessel fabricator was responsible for the first level inspection of his manufacturing, fabrication, and testing activities and General Electric is responsible for the first level of audit and surveillance inspection.

Adequate documentary evidence that the reactor vessel material, manufacture, testing, and inspection conforms to the specified quality assurance requirements contained in the procurement specification is available at the fabricator's office.

5.3.3.1 Design

5.3.3.1.1 Description

5.3.3.1.1.1 Reactor Vessel

The reactor vessel shown in Figure 5.3-1 is a vertical, cylindrical pressure vessel of welded construction. The vessels for Units 1 and 2 are designed, fabricated, tested, inspected, and stamped in accordance with the ASME Code Section III, Class A including the Summer Addenda 1970 except that: Paragraph NB-338.2(d)(4) of the Winter 1971 Addenda shall supercede Paragraph I-613(d) of the 1968 Edition, and paragraph NB-2400 of the 1971 Edition shall apply for all fabrication performed at the Susquehanna site. Design of the reactor vessel and its support system meets Seismic Category I equipment requirements. Quality control methods used during the fabrication and assembly of the reactor vessel and appurtenances assure that design specifications are met.

The materials used in the reactor pressure vessel are shown in Table 5.2-4. The cylindrical and bottom head sections of the reactor vessel are fabricated of low alloy steel, the interior of which is clad with stainless steel weld overlay. Nozzle and nozzle weld zones are unclad except for those mating to stainless steel piping systems.

In place annealing of the reactor vessel is unnecessary because shifts in transition temperature caused by irradiation during the 40-year life can be accommodated by raising the minimum pressurization temperature. Radiation embrittlement is not a problem outside of the vessel beltline region because the neutron fluence in those areas is less than 1×10^{17} n/cm^2 with neutron energies in excess of 1 MeV.

The vessel top head is secured to the reactor vessel by studs and nuts. These nuts are tightened with a stud tensioner. The vessel flanges are sealed with two concentric metal seal-rings designed to permit no detectable leakage through the inner or outer seal at any operating condition, including heating to operating pressure and temperature at a maximum rate of 100°F/hr. in any one hour period. To detect seal failure, a vent tap is located between the two seal-rings. A monitor line is attached to the tap to provide an indication of leakage from the inner seal-ring seal.

5.3.3.1.1.2 Shroud Support

The shroud support is a circular plate welded to the vessel wall. This support is designed to carry the weight of the shroud, shroud head, peripheral fuel elements, neutron sources, core plate, top guide, the steam separators, the jet pump diffusers, and to laterally support the fuel assemblies. Design of the shroud support also accounts for pressure differentials across the shroud support plate, for the restraining effect of components attached to the support, and for earthquake loadings. The shroud support design is specified to meet appropriate ASME code stress limits.

5.3.3.1.1.3 Protection of Closure Studs

Under normal operating conditions, the Boiling Water Reactor does not use borated water for reactivity control.

This subsection is therefore not applicable.

5.3.3.1.2 Safety Design Bases

Design of the reactor vessel and appurtenances meet the following safety design bases:

(1) The reactor vessel and appurtenances will withstand adverse combinations of loading and forces resulting from operation under abnormal and accident conditions.

(2) To minimize the possibility of brittle fracture of the nuclear system process barrier, the following are required:

   a. Impact properties at temperatures related to vessel operation have been specified for materials used in the reactor vessel.

b. Expected shifts in transition temperature during design life as a result of environmental conditions, such as neutron flux, are considered in the design. Operational limitations assure that NDT temperature shifts are accounted for in reactor operation.

c. Operational margins to be observed with regard to the transition temperature are specified for each mode of operation.

5.3.3.1.3 Power Generation Design Basis

The design of the reactor vessel and appurtenances meets the following power generation design basis:

(1) The reactor vessel has been designed for a useful life of 40 years. Operation of the vessel for the period of extended operation was reviewed for license renewal and found to be acceptable.

(2) External and internal supports that are integral parts of the reactor vessel are located and designed so that stresses in the vessel and supports that result from reactions at these supports are within ASME Code limits.

(3) Design of the reactor vessel and appurtenances allow for a suitable program of inspection and surveillance.

5.3.3.1.4 Reactor Vessel Design Data

Reactor vessel design data are contained in Tables 5.2-3 and 5.2-4.

5.3.3.1.4.1 Vessel Support

The reactor vessel support assembly consists of a ring girder and the various bolts and shims necessary to position and secure the assembly between the reactor vessel support skirt and the support pedestal. The concrete and steel support pedestal is constructed as an integral part of the building foundation. Steel anchor bolts are set in the concrete with their threads extending above the surface. The anchor bolts extend through the ring girder bottom flange. High strength bolts are used to secure the flange of the reactor vessel support skirt to the top flange of the ring girder. The ring girder is fabricated of ASTM A-36 Structural Steel according to AISC specifications.

5.3.3.1.4.2 Control Rod Drive Housings

The control rod drive housings are inserted through the control rod drive penetrations in the reactor vessel bottom head and are welded to the reactor vessel. Each housing transmits loads to the bottom head of the reactor. These loads include the weights of a control rod, a control rod drive, a control rod guide tube, a four-lobed fuel support piece, and the four fuel assemblies that rest on the fuel support piece. The housings are fabricated of Type 304 austenitic stainless steel.

5.3.3.1.4.3 In-Core Neutron Flux Monitor Housings

Each in-core neutron flux monitor housing is inserted through the in-core penetrations in the bottom head and is welded to the inner surface of the bottom head.
An in-core flux monitor guide tube is welded to the top of each housing and either a source range monitor/intermediate range monitor (SRM/IRM) drive unit or a local power range monitor (LPRM) is bolted to the seal/ring flange at the bottom of the housing (Section 7.6).

5.3.3.1.4.4 Reactor Vessel Insulation

The reactor vessel top head insulation is designed to permit complete submersion in water during shutdown without loss of insulating material, contamination of the water, or adverse effect on the insulation efficiency after draining. All reactor vessel insulation is the stainless steel, reflective type.

The top head insulation framework is designed to seismic category I requirements and is used as a structural support point for reactor vessel head spray and vent piping.
The insulation above the reactor vessel stabilizer brackets is close fitting, free-standing insulation designed to be 100% removable for inservice inspection of the reactor vessel.

The insulation below the stabilizer brackets is suspended from the brackets to allow a minimum of 8 inches annular clearance between the reactor vessel and the insulation for remote inservice inspection of the reactor vessel. The suspended insulation is also equipped with removable access ports.

Reactor vessel bottom head insulation includes horizontal flat panels connected to a cylindrical shell covering the inside of the reactor support skirt. The top row of the cylindrical shell panels are removable to expose the bottom head for inservice inspection.

Quick removable insulation is provided around all reactor vessel nozzles to allow manual or remote automatic examination of nozzle-to-vessel and nozzle-to-piping welds.

5.3.3.1.4.5 Reactor Vessel Nozzles

All piping connecting to the reactor vessel nozzles has been designed so as not to exceed the allowable loads on any nozzle.

The vessel top head nozzle is provided with a flange with small groove facing. The drain nozzle is of the full penetration weld design. The recirculation inlet nozzles (located as shown in Figure 5.3-1), feedwater inlet nozzles, and core spray inlet nozzles all have thermal sleeves.

Nozzles connecting to stainless steel piping have safe ends, made of stainless steel. These safe ends are welded to the nozzles after the pressure vessel has been heat treated to avoid furnace sensitization of the stainless steel safe ends. The material used is compatible with the material of the mating pipe. The nozzle for the standby liquid control pipe is designed to minimize thermal shock effects on the reactor vessel, in the event that use of the standby liquid control system is required.

5.3.3.1.4.6 Materials and Inspections

The reactor vessels were designed and fabricated in accordance with the appropriate ASME Boiler and Pressure Vessel Code as defined in Subsection 5.2.1. Table 5.2-4 defines the materials and specifications. Subsection 5.2.4 defines the compliance with reactor vessel Inservice Inspection program requirements.

5.3.3.1.4.7 Reactor Vessel Schematic

The reactor vessel schematic is illustrated in Figure 5.3-1.

5.3.3.2 Materials of Construction

All materials used in the construction of the reactor pressure vessel conform to the requirements of ASME Code Section II materials. The vessel heads, shells, flanges, and nozzles are fabricated from low alloy steel plate and forgings purchased in accordance with ASME specifications SA533 Grade B Class I and SA508 Class 2. Special requirements for the low alloy steel plate and forgings are discussed in Subsection 5.3.1.2. Cladding employed on the interior surfaces of the vessel consists of austenitic stainless steel weld overlay.

These materials of construction were selected because they provide adequate strength, fracture toughness, fabricability, and compatibility with the BWR environment. Their suitability has been demonstrated by long term successful operating experience in reactor service.

5.3.3.3 Fabrication Methods

The reactor pressure vessel is a vertical cylindrical pressure vessel of welded construction fabricated in accordance with ASME Code, Section III Class I requirements. All fabrication of the reactor pressure vessel was performed in accordance with GE approved drawings, fabrication procedures, and test procedures. The shells and vessel heads were made from formed low alloy steel plates, and the flanges and nozzles from low alloy steel forgings. Welding performed to join these vessel components was in accordance with procedures qualified per ASME Section III and IX requirements. Weld test samples were required for each procedure for major vessel full penetration welds.

Submerged arc, gas tungsten arc and gas metal arc welding processes were employed. Electroslag welding was not permitted. Preheat and interpass temperatures employed for welding of low alloy steel met or exceeded the requirements of ASME Section III, subsection NA. Post weld heat treatment of 1100°F minimum was applied to all low alloy steel welds.

All previous BWR pressure vessels have employed similar fabrication methods. These vessels have operated for periods up to 30 years and their service history is excellent. The vessel fabricator, CBI Nuclear Co., has had extensive experience with GE Co. reactor vessels and has been the primary supplier for GE domestic reactor vessels and some foreign vessels since the company was formed in 1972 from a merger agreement between Chicago Bridge and Iron Co. and General Electric Co. Prior experience by the Chicago Bridge and Iron Co. with GE Co. reactor vessels dates back to 1966.

5.3.3.4 Inspection Requirements

All plates, forgings, and boltings were 100% ultrasonically tested and surface examined by magnetic particle methods or liquid penetrant methods in accordance with ASME Section III requirements. Welds on the reactor pressure vessel were examined in accordance with methods prescribed and met the acceptance requirements specified by ASME Section III. In addition, the pressure retaining welds were ultrasonically examined using acceptance standards which were equivalent or more restrictive than required by ASME Section XI.

5.3.3.5 Shipment and Installation

The pressure vessel is shop fabricated in shell ring sections which are shipped to the site in a suitably protected condition. The shell rings are then field erected and finally assembled in place at the site. The completed reactor vessel is given a thorough cleaning and examination following the hydro test. The vessel is tightly sealed for storage to prevent entry of dirt or moisture. Preparations for shipment and storage were in accordance with detailed written procedures. Suitable measures are taken to assure that vessel integrity is maintained; for example, access controls are applied to personnel entering the vessel, weather protection is provided, and periodic inspections are performed.

5.3.3.6 Operating Conditions

Procedural controls on plant operation are implemented to hold thermal stresses within acceptable ranges. These restrictions on coolant temperature are:

(1) The average rate of change of reactor coolant temperature during normal heatup and cooldown shall not exceed 100°F during any one-hour period.

(2) If the coolant temperature difference between the dome (inferred from Psat) and the bottom head drain exceeds 145°F, neither reactor power level nor recirculation pump flow shall be increased.

(3) The pump in an idle reactor recirculation loop shall not be started unless the coolant temperature in that loop is within 50°F of average reactor coolant temperature.

The limit regarding the normal rate of heatup and cooldown (Item 1) assures that the vessel closure, closure studs, vessel support skirt, and control rod drive housing and stub tube stresses and usage remain within acceptable limits. The limit regarding a vessel temperature limit on recirculation pump operation and power level increase restriction (Item 2) augments the Item 1 limit in further detail by assuring that the vessel bottom head region will not be warmed at an excessive rate caused by rapid sweep out of cold coolant in the vessel lower head region by recirculation pump operation or natural circulation (cold coolant can accumulate as a result of control drive in leakage and/or low recirculation flow rate during startup or hot standby). The Item 3 limit further restricts operation of the recirculation pumps to avoid high thermal stress effects in the pumps and piping, while also minimizing thermal stresses on the vessel nozzles.

The above operational limits, when maintained, ensure that the stress limits within the reactor vessel and its components are within the thermal limits to which the vessel was designed for normal operating conditions. To maintain the integrity of the vessel in the event that these

operational limits are exceeded, the reactor vessel has also been designed to withstand a limited number of transients caused by operator error. Also, for abnormal operating conditions where safety systems or controls provide an automatic temperature and pressure response in the reactor vessel, the reactor vessel integrity is maintained since the severest anticipated transients have been included in the design conditions. Therefore, it is concluded that the vessel integrity will be maintained during the most severe postulated transients, since all such transients are evaluated in the design of the reactor vessel. The postulated transient for which the vessel has been designed is discussed in Subsection 5.2.2.

5.3.3.7 Inservice Surveillance

Inservice inspection of the reactor pressure vessel will be in accordance with the requirements discussed in Subsection 5.2.4. The materials surveillance program will monitor changes in the fracture toughness properties of ferritic materials in the reactor vessel beltline region resulting from their exposure to neutron irradiation and thermal environment. Specimens of actual reactor beltline material will be exposed in the reactor vessel and periodically withdrawn for impact testing. Operating procedures will be modified in accordance with test results to assure adequate brittle fracture control.

Material surveillance programs and inservice inspection programs are in accordance with applicable ASME Code requirements, and provide assurance that brittle fracture control and pressure vessel integrity will be maintained throughout the service lifetime of the reactor pressure vessel.

5.3.4 References

5.3-1 Faynshtein, K., and D. R. Pankratz, "Power Uprate Engineering Report for Susquehanna Steam Electric Station, Units 1 and 2," General Electric Report NEDC-32161P, as revised by PP&L Calculation EC-PUPC-1001, Revision 0, March, 1994.

5.3-2 Carey, R. G., "Susquehanna Steam Electric Station Unit 1 Vessel Surveillance Materials Testing and Fracture Toughness Analysis," General Electric Report GE-NE-523-169-1292, Revision 1, October, 1993. Attached to PP&L letter PLA-3953, R. G. Byram to C. L. Miller, NRC, "Susquehanna Steam Electric Station, Submittal of Reactor Vessel Material Surveillance Test Report per 10CFR50 Appendix H for Unit 1," April 8, 1993.

5.3-3 Contreras, G. W., "Susquehanna Steam Electric Station Unit 2 Vessel Surveillance Materials Testing and Fracture Toughness Analysis," General Electric Report GE-NE-523-107-0893, Revision 1, October, 1993. Attached to PP&L Letter PLA-4126, R. G. Byram to C. L. Miller, NRC, "Susquehanna Steam Electric Station, Submittal of Revision to Reactor Vessel Material Surveillance Test Report per 10CFR50 Appendix H for Unit 2," May 19, 1994.

5.3-4 DuBord, R.M., “Susquehanna Steam Electric Station Unit 1 Fabrication of New Surveillance Capsule with Reconstituted Charpy Specimens” General Electric Nuclear Energy report GE-NE-523-A054-0595, May 1995.

5.3-5 DuBord, R.M., “Susquehanna Steam Electric Station Unit 2 Fabrication of New Surveillance Capsule with Reconstituted Charpy Specimens” General Electric Nuclear Energy report GE-NE-523-A055-0595, May 1995.

5.3-6 Structural Integrity Associates Report No. SIR-00-167, Revision 0, “Revised Pressure-Temperature Curves for Susquehanna Units 1 and 2”, January 2001.

5.3-7 BWRVIP-86: BWR Vessel Internals Project, BWR Integrated Surveillance Program Implementation Plan, February 2002 including future revisions.

5.3-8 BWRVIP-78: BWR Vessel Internals Project, BWR Integrated Surveillance Program Plan, February 2002.

5.3-9 BWRVIP Responses to NRC Staff RAIs dated December 22, 2001 and May 30, 2001.

5.3-10 NRC Safety Evaluation regarding the Integrated Surveillance Program, February 1, 2002.

TABLE 5.3-1a

APPENDIX G MATRIX FOR SUSQUEHANNA SES UNIT 1

Appendix G Para. No.	Topic	Comply Yes/No or N. A.	Alternate Actions or Comments
I, II	Introduction; Definitions	--	
III.A	Compliance With ASME Code, Section NB-2300	Yes	See Section 5.3.1.5.1.2 for discussion.
III.B.1	Location & Orientation of Impact Test Spec	Yes	See III.A, above.
III.B.2	Materials Used to Prepare Test Specimens	No	Compliance except for CVN orientation and CVN upper shelf.
III.B.3	Calibration of Temp. Inst. and Charpy Test Machines	No	Paragraph NB-2360 of the ASME B&PV code, Section IV, was not in existence at the time of purchase of the Susquehanna SES Unit 1 reactor pressure vessel. However, the requirements of the 1971 edition of the ASME B&PV Section III Code, Summer 1971 addenda, were met. For the discussions of the GE interpretations of compliance and NRC acceptance see References 1 and 2. The temperature instruments and Charpy Test Machines calibration data are retained until the next recalibration. This is in accordance with Reg. Guide 1.88 Rev. 2, GE Alternative Position 1.88, and ANSI N45.2.9, 1974. Therefore, the instrument calibration data for Susquehanna SES Unit 1 would not be currently available.
III.B.4	Qualification of Testing Personnel	No	No written procedures were in existence as required by the Regulation; however, the individuals were qualified by on-the-job training and past experience. For the discussion of the GE interpretation of compliance and NRC acceptance see References 1 and 2.
III.B.5	Test Results Recording and Verification	Yes	See References 1 and 2.
III.C.1	Test Conditions	No	See III.A, III.B.2, above.
III.C.2	Materials Used to Prepare Test Specimens for Reactor Vessel Beltline	Yes	Compliance on base metal and weld metal tests. Test weld not made on same heat of base plate, necessarily.
IV.A.1	Acceptance Standard of Materials	--	
IV.A.2.a	Calculated Stress Intensity Factor	Yes	
IV.A.2.b	Requirements for Nozzles, Flanges and Shell Region Near Geometric Discontinuities	No	Plus 60°F was added to the RT_{NDT} for the reactor vessel flanges. For feedwater nozzles, the results of the BWR/6 analysis were adjusted to Susquehanna SES Unit 1 RT_{NDT} conditions.
IV.A.2.c	RPV Metal Temperature Requirement When Core is Critical	Yes	

TABLE 5.3-1a

APPENDIX G MATRIX FOR SUSQUEHANNA SES UNIT 1

Appendix G Para. No.	Topic	Comply Yes/No or N. A.	Alternate Actions or Comments
IV.A.2.d	Minimum Permissible Temp. During Hydro Test	Yes	
IV.A.3	Materials for Piping, Pumps and Valves	No	Main steamline piping is in compliance. See Subsection 5.2.3.3.1 for discussions on pumps and valves.
IV.A.4	Materials for Bolting and Other Fasteners	Yes	Closure studs tested at preload or lowest service temperature minus 60°F to give lowest values of 40 ft-lb and 25 mils at +10°F. This is equivalent to meeting current requirements of 45 ft-lb and 25 mils at +70°F.
IV.B	Minimum Upper Shelf Energy for RPV Beltline	No	No upper shelf tests run. However, recommend acceptance based upon lowest longitudinal CVN's for plates at +10°F of 37 ft-lb (40% shear) for heat C0776-1 (0.12% Cu), 31 ft-lb (30% shear) for heat C2433-1 (0.10% Cu), and 44 dt-lb (50% shear) for heat B5083-1 (0.14% Cu). Lowest CVN's for welds are 51 ft-lb (no % shear records) at +10°F for 0.04% Cu. End-of-life upper shelf values (100% shear) are predicted to be in excess of 50 ft-lb, based upon preceding data and Regulatory Guide 1.99.
IV.C	Requirement for Annealing when RT_{NDT} 200°F	N/A	
V.A	Requirements for Material Surveillance Program	See App. H	
V.B	Conditions for Continued Operation	Yes	See section 5.3.1.5.1.1, 5.3.1.5.1.2., 5.3.1.5.1.3., 5.3.1.5.1.4, 5.3.1.5.1.6, 5.3.1.6 and Table 5.3-1b
V.C	Alternative if V.B Cannot be Satisfied	---	---
V.D	Requirement for RPV Thermal Annealing if V.C Cannot be Met	N/A	
V.E	Reporting Requirement for V.C and V.D	N/A	

References:

1. Letter MFN-414-77, G. G. Sherwood (GE) to Edson G. Case (NRC) dated October 17, 1977.

2. Letter, Robert B. Minogue (NRC) to G. G. Sherwood (GE) dated February 14, 1978.

TABLE 5.3-1b

APPENDIX H MATRIX FOR SUSQUEHANNA SES UNIT 1

APPENDIX H PARA. NO.	TOPIC	COMPLY YES/NO OR N.A.	ALTERNATE ACTIONS OR COMMENTS
I	Introduction	N/A	
II.A	Fluence <10^{17} n/cm^2 – Surveillance Program Not Required	N/A	
II.B	Standards Requirements (ASTM) for Surveillance	No	Noncompliance with ASTM E185-73 in that the surveillance specimens are not necessarily from the limiting beltline material. Specimens are from representative beltline material, however, and can be used to predict behavior of the limiting material. Heat and heat/lot numbers for surveillance specimens are to be supplied.
II.C.1	Surveillance Specimen Shall be Taken from Locations Alongside the Fracture Test Specimens (Section III.B of Appendix G)	No	Noncompliance in that specimens may not have necessarily been taken from along-side specimens required by Section III of Appendix G and transverse CVNs may not be employed. However, representative materials have been used, and RT_{NDT} shift appears to be independent of specimen orientation.
II.C.2	Locations of Surveillance Capsules in RPV	Yes	Code basis is used for attachment of brackets to vessel cladding. See Section 5.3.1.6.4.
II.C.3.a	Withdrawal Schedule of Capsules, RT_{NDT}<100°F	Yes	Starting RT_{NDT} of limiting material is based on alternative action (see Paragraph III.A of Appendix G). One capsule complete. Other capsules are scheduled and tested in accordance with Reference 5.3-7 through 5.3-10..
II.C.3.b	Withdrawal Schedule of Capsules, RT_{NDT}<200°F	N/A	
II.C.3.c.	Withdrawal Schedule of Capsules, RT_{NDT}<200°F	N/A	
III.A	Fracture Toughness Testing Requirements of Specimens	Yes	See Section 5.3.1.6
III.B	Method of Determining Adjusted Reference Temp. for Base Metal, HAZ and Weld Metal	Yes	See Section 5.3.1.5
IV.A	Reporting Requirements of Test Results	Yes	See Section 5.3.1.6
IV.B	Requirement for Dosimetry Measurement	Yes	See Section 5.3.1.6.2, 5.3.1.6.4
IV.C	Reporting Requirements of Press/Temp. Limits	Yes	See Section 5.3.2

TABLE 5.3-2a

APPENDIX G MATRIX FOR SUSQUEHANNA UNIT 2

Appendix G Para. No.	Topic	Comply Yes/No or N. A.	Alternate Actions or Comments
I, II	Introduction; Definitions	--	
III.A	Compliance With ASME Code, Section NB-2300	Yes	See Subsection 5.3.1.5.1.2 for discussion.
III.B.1	Location & Orientation of Impact Test Spec	Yes	See III.A, above.
III.B.2	Materials Used to Prepare Test Specimens	No	Compliance except for CVN orientation and CVN upper shelf.
III.B.3	Calibration of Temp. Inst. and Charpy Test Machines	No	Paragraph NB-2360 of the ASME B&PV code, Section IV, was not in existence at the time of purchase of the Susquehanna SES Unit 2 reactor pressure vessel. However, the requirements of the 1971 edition of the ASME B&PV Section III Code, Summer 1971 addenda, were met. For the discussions of the GE interpretations of compliance and NRC acceptance see References 1 and 2. The temperature instruments and Charpy Test Machines calibration data are retained until the next recalibration. This is in accordance with Reg. Guide 1.88 Rev. 2, GE Alternative Position 1.88, and ANSI N45.2.9, 1974. Therefore, the instrument calibration data for Susquehanna SES Unit 2 would not be currently available.
III.B.4	Qualification of Testing Personnel	No	No written procedures were in existence as required by the Regulation; however, the individuals were qualified by on-the-job training and past experience. For the discussion of the GE interpretation of compliance and NRC acceptance see References 1 and 2.
III.B.5	Test Results Recording and Certification	Yes	See References 1 and 2.
III.C.1	Test Conditions	No	See III.A, III.B.2, above.
III.C.2	Materials Used to Prepare Test Specimens for Reactor Vessel Beltline	Yes	Compliance on base metal and weld metal tests. Test weld not made on same heat of base plate, necessarily.
IV.A.1	Acceptance Standard of Materials	--	
IV.A.2.a	Calculated Stress Intensity Factor	Yes	
IV.A.2.b	Requirements for Nozzles, Flanges and Shell Region Near Geometric Discontinuities	Yes	Plus 60°F was added to the RT_{NDT} for the reactor vessel flanges. For other nozzles and discontinuities, the results of the BWR/6 analysis were adjusted to Susquehanna SES Unit 2 RT_{NDT} conditions.
IV.A.2.c	RPV Metal Temperature Requirement When Core is Critical	Yes	

TABLE 5.3-2a

APPENDIX G MATRIX FOR SUSQUEHANNA UNIT 2

Appendix G Para. No.	Topic	Comply Yes/No or N. A.	Alternate Actions or Comments
IV.A.2.d	Minimum Permissible Temp. During Hydro Test	Yes	
IV.A.3	Materials for Piping, Pumps and Valves	No	Main steamline piping is in compliance. See Subsection 5.2.3.3.1 for discussions on pumps and valves.
IV.A.4	Materials for Bolting and Other Fasteners	Yes	Current toughness requirements for closure head studs are met at 10°F.
IV.B	Minimum Upper Shelf Energy for RPV Beltline	No	No upper shelf tests run. However, recommend acceptance based upon lowest longitudinal CVN's for plates at +10°F of 45 ft-lb (50% shear) for heat C2421-3 (0.10% Cu), 50 ft-lb (50% shear) for heat C2929-1 (0.13% Cu), and 39 ft-lb (40% shear) for heat C2433-2 (0.10% Cu). Lowest CVN's for welds are 22, 30, 31, 43, 55 ft-lb (no % shear records) at -20°F with 0.06% Cu. The scatter in energy data at -20°F indicates transition behavior and the probability that upper shelf is in excess of 50 ft-lb (for 100% shear). End-of-life upper shelf values (100% shear) are predicted to be in excess of 50 ft-lb, based upon preceding data and Regulatory Guide 1.99.
IV.C	Requirement for Annealing when RT_{NDT} 200°F	N/A	
V.A	Requirements for Material Surveillance Program	See App. H	
V.B	Conditions for Continued Operation	Yes	See section 5.3.1.5.1.1, 5.3.1.5.1.2., 5.3.1.5.1.3., 5.3.1.5.1.4, 5.3.1.5.1.6, 5.3.1.6 and Table 5.3-2b
V.C	Alternative if V.B Cannot be Satisfied	---	---
V.D	Requirement for RPV Thermal Annealing if V.C Cannot be Met	N/A	
V.E	Reporting Requirement for V.C and V.D	N/A	

References:

1. Letter MFN-414-77, G. G. Sherwood (GE) to Edson G. Case (NRC) dated October 17, 1977.
2. Letter, Robert B. Minogue (NRC) to G. G. Sherwood (GE) dated February 14, 1978.

*This table references the 1980 Appendices G and H requirements. The engineering rationale used to meet the 10CFR50 Appendices G and H requirements of 1980 are equally applicable to meeting the 10CFR50 Appendices G and H Requirements of 1983.

TABLE 5.3-2b

APPENDIX H MATRIX FOR SUSQUEHANNA SES UNIT 2

APPENDIX H PARA. NO.	TOPIC	COMPLY YES/NO OR N.A.	ALTERNATE ACTIONS OR COMMENTS
I	Introduction	N/A	
II.A	Fluence $<10^{17}$ n/cm^2 – Surveillance Program Not Required	N/A	
II.B	Standards Requirements (ASTM) For Surveillance	No	Noncompliance with ASTM E185-73 in that the surveillance specimens are not necessarily from the limiting beltline material. Specimens are from representative beltline material, however, and can be used to predict behavior of the limiting material. Heat and heat/lot numbers for surveillance specimens are to be supplied.
II.C.1	Surveillance Specimen Shall be Taken from Locations Alongside the Fracture Test Specimens (Section III.B of Appendix G)	No	Noncompliance in that specimens may not have necessarily been taken from alongside specimens required by Section III of Appendix G and transverse CVN's may not be employed. However, representative materials have been used, and RT_{NDT} shift appears to be independent of specimen orientation.
II.C.2	Locations of Surveillance Capsules in RPV	Yes	Code basis is used for attachment of brackets to vessel cladding. See Subsections 5.3.1.6.4.
II.C.3.a	Withdrawal Schedule of Capsules, $RT_{NDT}<100°F$	Yes	Starting RT_{NDT} of limiting material is based on alternative action (see Paragraph III.A of Appendix G). One capsule completed. Other capsules are scheduled and tested in accordance with References 5.3-7 through 5.3-10.
II.C.3.b	Withdrawal Schedule of Capsules, $RT_{NDT}<200°$	N/A	
II.C.3.c	Withdrawal Schedule of Capsules, $RT_{NDT}<200°$	N/A	
III.A	Fracture Toughness Testing Requirements of Specimens	Yes	See Section 5.3.1.6
III.B	Method of Determining Adjusted Reference Temp. for Base Metal, HAZ and Weld Metal	Yes	See Section 5.3.1.5
IV.A	Reporting Requirements of Test Results	Yes	See Section 5.3.1.6
IV.B	Requirement for Dosimetry Measurement	Yes	See Section 5.3.1.6.2, 5.3.1.6.4
IV.C	Reporting Requirements of Press/Temp. Limits	Yes	See Section 5.3.2

Table Rev. 56

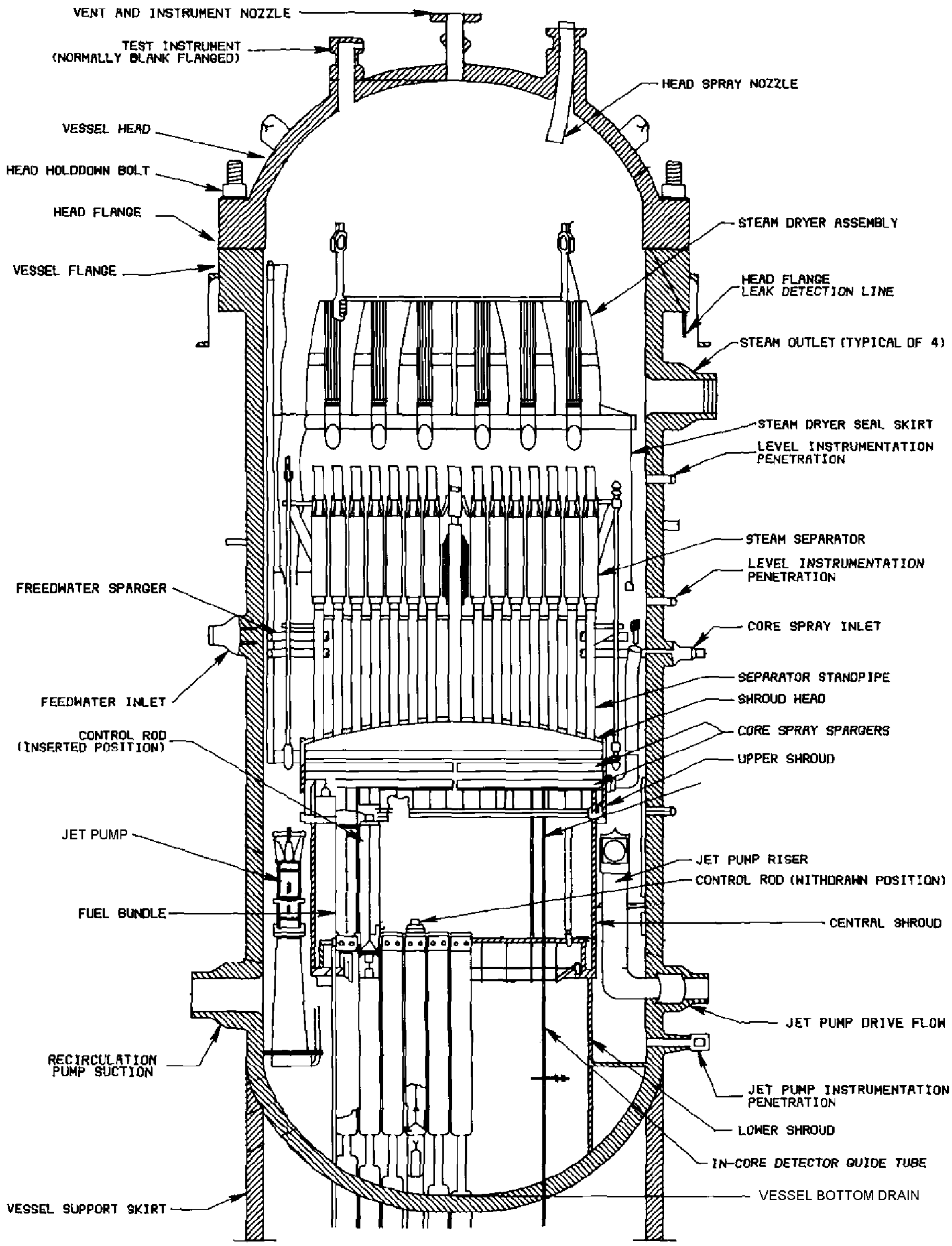
TABLE 5.3-3

REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM-WITHDRAWAL SCHEDULE

Specimen Holder	Vessel Location	Lead Factor *	Withdrawal Time (EFPY)
UNIT 1			
131C7717G1	300°	1.20	Spare
131C7717G2	120°	1.20	22
131C7717G3	30°	1.20	6 (Actual Date - Fall 1992)
G3 Reconstituted Specimens	30°	1.20	Spare
UNIT 2			
131C7717G1	300°	1.20	Spare
131C7717G2	120°	1.20	Spare
131C7717G3	30°	1.20	6 (Actual Date - Fall 1992)
G3 Reconstituted Specimens	30°	1.20	Spare

* At 1/4 T.

\# Withdrawal Time is in accordance with Reference 5.3-7 through 5.3-10..



REACTOR VESSEL
COMPOSITE DRAWING

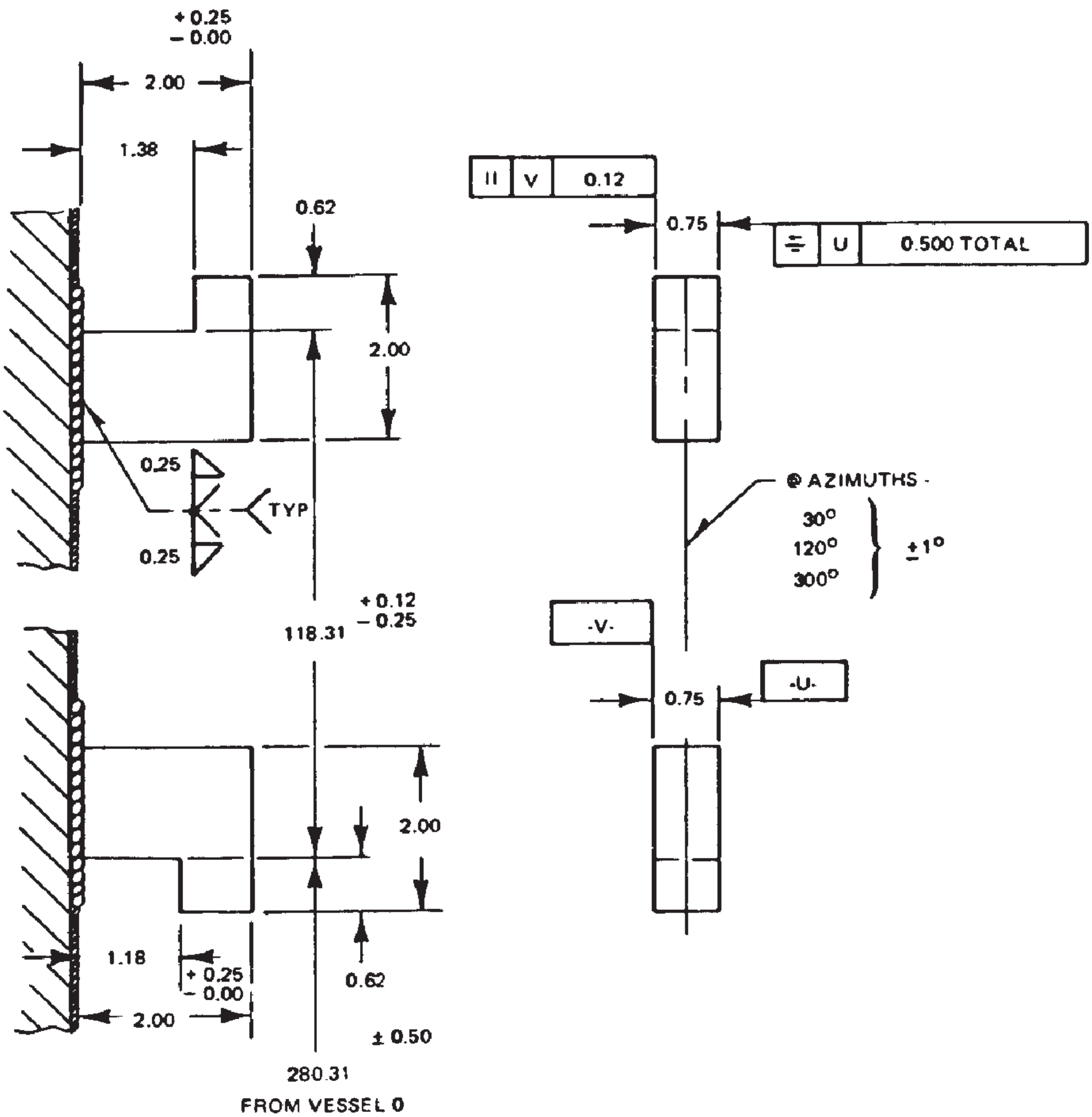
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

REACTOR VESSEL

FIGURE 5.3-1, Rev.54

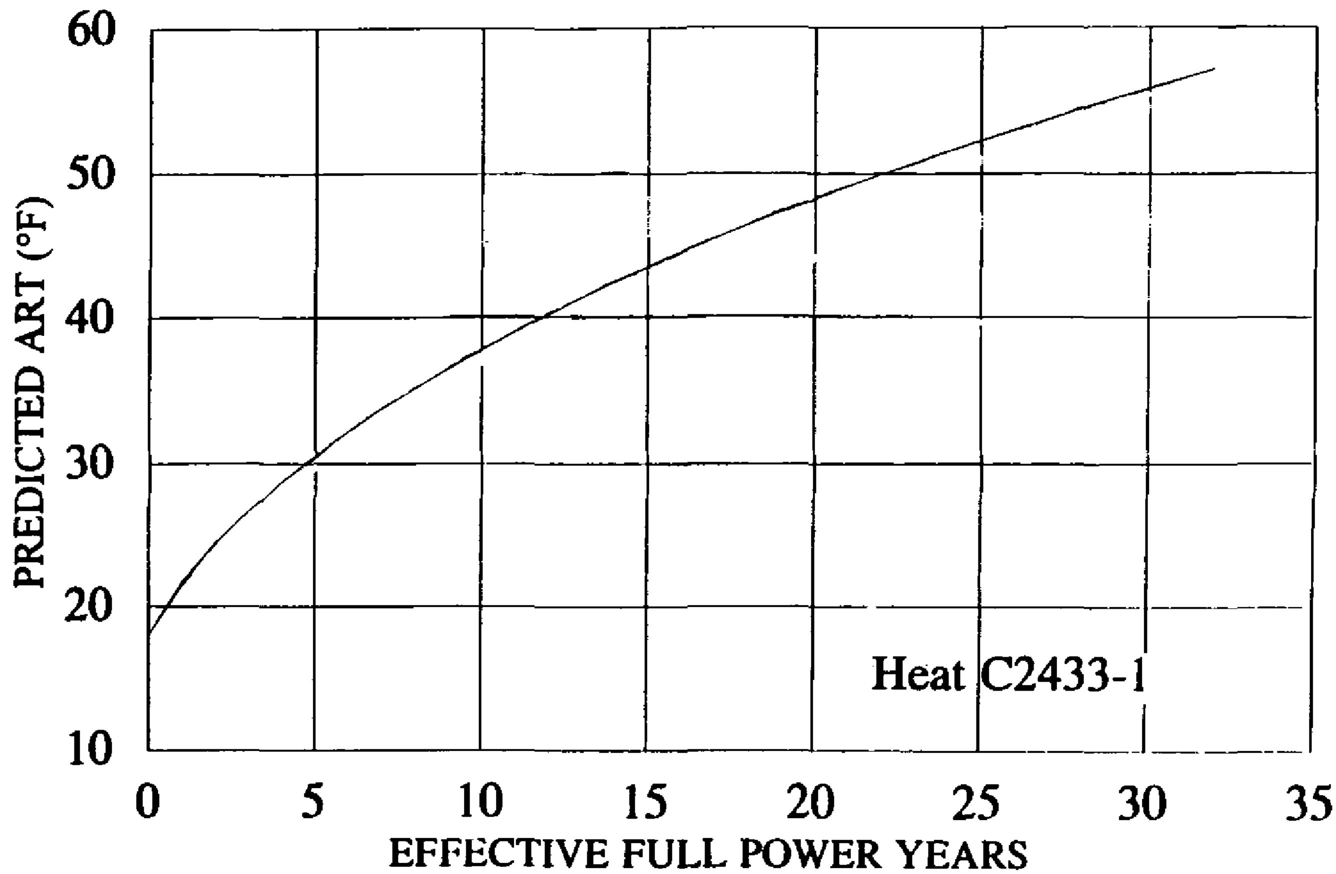
AutoCAD: Figure Fsar 5_3_1.dwg



FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
BRACKET FOR HOLDING SURVEILLANCE CAPSULE FIGURE 5.3-3, Rev.47

AutoCAD: Figure Fsar 5_3_3.dwg



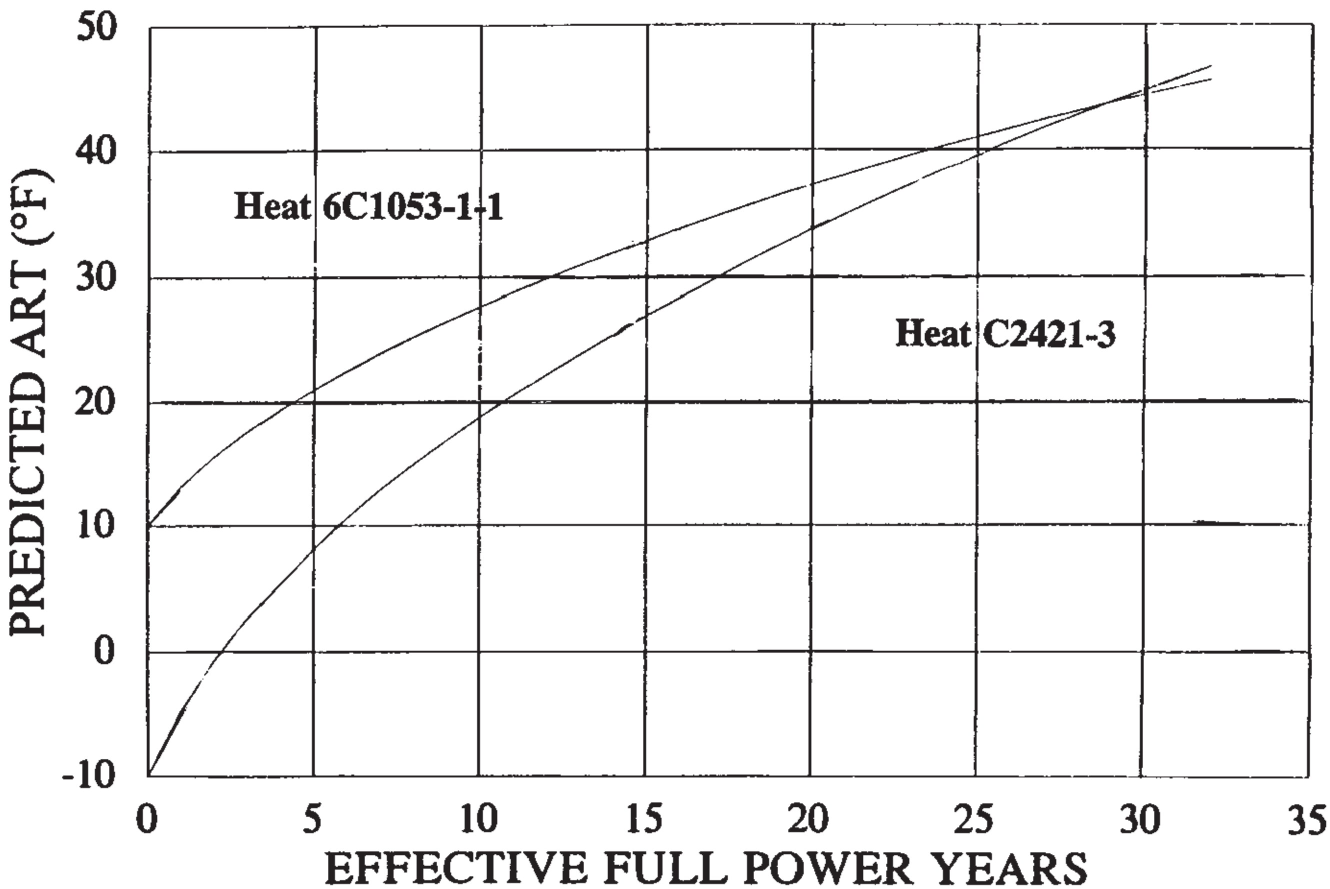
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

UNIT 1
PREDICTED ADJUSTED REFERENCE
TEMPERATURE VS. EFFECTIVE FULL
POWER YEARS OF OPERATION

FIGURE 5.3-4C, Rev.50

AutoCAD: Figure Fsar 5_3_4C.dwg



FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

UNIT 2
PREDICTED ADJUSTED REFERENCE
TEMPERATURE VS. EFFECTIVE FULL
POWER YEARS OF OPERATION

FIGURE 5.3-4D, Rev.50

AutoCAD: Figure Fsar 5_3_4D.dwg

5.4 COMPONENT AND SUBSYSTEM DESIGN

5.4.1 REACTOR RECIRCULATION PUMPS

5.4.1.1 Safety Design Bases

The reactor recirculation system has been designed to meet the following safety design bases:

(1) An adequate fuel barrier thermal margin shall be assured during postulated transients.

(2) A failure of piping integrity shall not compromise the ability of the reactor vessel internals to provide a refloodable volume.

(3) The system shall maintain pressure integrity during adverse combinations of loadings and forces occurring during abnormal, accident, and special event conditions.

5.4.1.2 Power Generation Design Bases

The reactor recirculation system meets the following power generation design bases:

(1) The system shall provide sufficient flow to remove heat from the fuel.

(2) The system shall provide load change capability over the range of 65 to 100% rated power.

(3) System design shall minimize maintenance situations that would require core disassembly and fuel removal.

5.4.1.3 Description

The reactor recirculation system consists of the two recirculation pump loops external to the reactor vessel. These loops provide the piping path for the driving flow of water to the reactor vessel jet pumps (see Figure 5.4-1, Dwgs. M-143, Sh. 1, M-143, Sh. 2 and M1-B31-13, Sh. 3) Each external loop contains one high capacity variable speed motor-driven recirculation pump, two motor-operated gate valves for pump maintenance, and a gate valve in the bypass line around the discharge gate valve. Each loop contains a flow measuring system. The recirculation loops are part of the reactor coolant pressure boundary and are located inside the drywell structure. The jet pumps are reactor vessel internals. Their location and mechanical design are discussed in Subsection 3.9.5. However, certain operational characteristics of the jet pumps are discussed in this subsection. A tabulation of the important design and performance characteristics of the reactor recirculation system is shown in Table 5.1-1. The head, NPSH, flow, and efficiency curves of the recirculation pumps are shown in Figure 5.4-3.

The recirculated coolant consists of saturated water from the steam separators and dryers that has been subcooled by incoming feedwater. This water passes down the annulus between the reactor vessel wall and the core shroud. A portion of the coolant flows from the vessel, through the two external recirculation loops, and becomes the driving flow for the jet pumps. Each of the two external recirculation loops discharges high pressure flow into an external manifold from

which individual recirculation inlet lines are routed to the jet pump risers within the reactor vessel. The remaining portion of the coolant mixture in the annulus becomes the driven flow for the jet pumps. This flow enters the jet pump at suction inlets and is accelerated by the driving flow. The flows, both driving and driven, are mixed in the jet pump throat section and result in partial pressure recovery. The balance of recovery is obtained in the jet pump diffuser section (see Figure 5.4-5). The adequacy of the total flow to the core is discussed in Section 4.4.

There is a four-inch bypass line around each pump discharge gate valve in the recirculation loop. The bypass line is used when returning a pump to service. The pump is started at slow speed with the main discharge valve closed and the bypass valve open. Pump speed is not increased above the minimum setpoint until after the main valve has been opened. There is actually a very low probability that a recirculation loop that has been allowed to cool would need to be placed in service again with the nuclear system hot. The only valid reason for closing the pump discharge valve, the discharge bypass valve and the suction valve, is to prevent leakage out of that portion of the recirculation loop between the valves, e.g., excessive leakage through the pump mechanical seal. A leak of this nature cannot be repaired without shutting the plant down to permit access to the drywell; the nuclear system would in all probability be cooled prior to repairing the leak.

The allowable heatup rate for the recirculation pump casing is the same as the reactor vessel. If one loop is shut down, the idle loop can be kept hot by leaving the discharge and suction gate valves open; this permits the reactor pressure plus the active jet pump head to cause reverse flow in the idle loop.

Because the removal of the reactor recirculation gate valve internals would require unloading the core, the objective of the valve trim design is to minimize the need for maintenance of the valve internals. The valves are provided with high quality backseats that permit renewal of stem packing while the system is full of water.

The feedwater flowing into the reactor vessel annulus during operation provides subcooling for the fluid passing to the recirculation pumps and jet pumps, thus providing additional net positive suction head (NPSH) available beyond that provided by the pump location below the reactor vessel water level. If feedwater flow is below the minimum value which provides adequate NPSH for full speed recirculation pump operation, the pump speed is automatically limited.

When preparing for hydrostatic tests, the nuclear system temperature must be raised above the vessel nil ductility transition temperature limit. The vessel is heated by operating the recirculation pumps and/or by core decay heat.

Connections to the piping on the suction and discharge sides of the pumps, as shown on Dwgs. M-143, Sh. 1, M-143, Sh. 2, and M1-B31-13, Sh. 3 provide a means to flush and decontaminate the pump and adjacent piping. The piping low point drain, designed for the connection of temporary piping, is used during flushing or decontamination.

Each recirculation pump is a single stage, variable speed, vertical, centrifugal pump equipped with mechanical shaft seal assemblies. The pump is capable of satisfactory performance while operating continuously at any speed corresponding to a power supply frequency range of 11.5 to 57.5 Hz for 60 Hz power supply.

The recirculation pump shaft seal assembly consists of two individual seals built into a cartridge or cartridges which can be readily replaced without removing the motor from the pump. The

seal assembly is designed to require minimum maintenance over a long period of time, regardless of whether the pump is stopped or is operating at various speeds with water at various pressures and temperatures. Each seal is designed for a life of one year based on a 90 percent probability factor. Each individual seal in the cartridge is capable of sealing against pump design pressure so that any one seal can adequately limit leakage in the event that the other seal should fail. A breakdown orifice is provided in the pump casing to reduce leakage in the event of a gross failure of both shaft seals. Provision is made for monitoring the pressure drop across each individual seal as well as the cavity temperature of each seal. Provision is also made for piping the seal leakage to a flow measuring device.

Each recirculation pump motor is a variable speed ac electric motor which can drive the pump over a controlled range of 20 percent to 109.5 percent of rated pump speed. The motor is designed to operate continuously at any speed within the power supply frequency range of 11.5 Hz to 57.5 Hz for 60 Hz power supply. Electrical equipment is designed, constructed, and tested in accordance with the applicable sections of the NEMA Standards.

A variable frequency ac motor-generator set located outside the drywell supplies power to each recirculation pump motor. The pump motor is electrically connected to the generator and is started by engaging the variable speed coupling between the generator and the motor.

The combined rotating inertias of the recirculation pump and motor, motor-generator set, and the variable speed coupling are chosen to provide a slow coastdown of flow following normal shutdown and/or loss of power to the drive motors, so that the core is adequately cooled. For RPT transients, the RPT breakers disconnect the M/G set interia to achieve a rapid coastdown to limit the heatflux across the cladding.

Pump casing and valve bodies are designed for a 40-year life and are welded into the piping system with no plans to remove them from the system for maintenance or overhaul. Since the system must perform for the period of extended operation, aging of equipment is managed to ensure it continues to perform its intended function. Removable parts of the pump such as wear rings, impellers, bearings, etc. are designed for as long a life as practical, and as a design objective, they should have a life between overhaul or major maintenance cycle of more than five years. Pump seals and valve packings are expected to have a useful service life in excess of an operating cycle to afford convenient replacement during the refueling outage.

The recirculation system piping is of all-welded construction and is designed and constructed to meet the requirements of the ASME Code.

The reactor recirculation system pressure boundary equipment is designed as Seismic Category I.

Vibration snubbers located at the top of the motor and at the bottom of the pump casing are designed to resist the horizontal reactions.

The recirculation piping, valves, and pumps are supported by hangers to avoid the use of piping expansion loops that would be required if the pumps were anchored. In addition, the recirculation loops are provided with a system of restraints designed so that reaction forces associated with any split or circumferential break do not jeopardize primary containment integrity. This restraint system provides adequate clearance for normal thermal expansion movement of the loop. Because possible pipe movement is limited to slightly more than the clearance required for thermal expansion movement, no impact loading on limit stops is

considered. A more detailed discussion of the recirculation piping restraints can be found in Section 3.6.

The recirculation system piping, valves, and pump casings are covered with thermal insulation having a total maximum heat transfer rate of 65 Btu/hr-ft^2 with the system at rated operating conditions.

The insulation is the all-metal reflective type. It is prefabricated into components for field installation. Removable insulation is provided at various locations to permit periodic inspection of the equipment.

5.4.1.4 Safety Evaluation

Reactor recirculation system malfunctions that pose threats of damage to the fuel barrier are described and evaluated in Chapter 15. It is shown in Chapter 15 that none of the malfunctions result in significant fuel damage. The recirculation system has sufficient flow coastdown characteristics to maintain fuel thermal margins during abnormal operational transients.

The core flooding capability of a jet pump design plant is discussed in detail in the emergency core cooling systems document filed with the NRC as a General Electric topical report (Reference 5.4-1). The ability to reflood the BWR core to the top of the jet pumps as shown schematically in Figure 5.4-6 and as discussed in Reference 5.4-1 applies to all jet pump BWRs and does not depend on the plant size or product line.

Piping and pump design pressures for the reactor recirculation system are based on peak steam pressure in the reactor dome, appropriate pump head allowances, and the elevation head above the lowest point in the recirculation loop. Piping and related equipment pressure parts are chosen in accordance with applicable codes. Use of the listed code design criteria assures that a system designed, built, and operated within design limits has an extremely low probability of failure caused by any known failure mechanism.

General Electric Purchase Specifications require that the recirculation pumps first critical speed shall not be less than 130% of operating speed. Calculation submittal was required and verified by General Electric Design Engineering.

General Electric Purchase Specifications require that integrity of the pump case be maintained through all transients and that the pump remain operable through all normal and upset transients. The design of the pump and motor bearings are required to be such that dynamic load capability at rated operating conditions is not exceeded during the safe shutdown earthquake. Calculation submittal to General Electric was required.

Pump overspeed occurs during the course of a LOCA due to blowdown through the broken loop pump. Design studies determined that rotating component failure missiles caused by the overspeed was not sufficient to cause damage to the containment or to vital equipment, consequently no provision is made.

5.4.1.5 Inspection and Testing

Quality control methods were used during fabrication and assembly of the reactor recirculation system to assure that design specifications were met. Inspection and testing was carried out as described in Chapter 3. The reactor coolant system was thoroughly cleaned and flushed before fuel was loaded initially.

During the preoperational test program, the reactor recirculation system was hydrostatically tested at 125% reactor vessel design pressure. Preoperational tests on the reactor recirculation system also included checking operation of the pumps, flow control system, and gate valves, as discussed in Chapter 14.

During the startup test program, horizontal and vertical motions of the reactor recirculation system piping and equipment were checked and supports were adjusted, as discussed in Chapter 14.

5.4.2 STEAM GENERATORS (PWR)

Not applicable to this BWR.

5.4.3 REACTOR COOLANT PIPING

The reactor coolant piping is discussed in Subsection 5.4.1. The recirculation loops are shown in Figures 5.4-1, Dwgs. M-143, Sh. 1, M-143, Sh. 2 and M1-B31-13, Sh. 3. The design characteristics are presented in Table 5.4-1.

5.4.4 MAIN STEAMLINE FLOW RESTRICTORS

5.4.4.1 Safety Design Bases

The main steamline flow restrictors are engineered safety features and designed:

(1) To limit the loss of coolant from the reactor vessel following a steamline rupture outside the containment to the extent that the reactor vessel water level remains high enough to provide cooling within the time required to close the main steamline isolation valves.

(2) To withstand the maximum pressure difference expected across the restrictor, following complete severance of a main steamline.

(3) To limit the amount of radiological release outside of the drywell prior to MSIV closure.

(4) To provide trip signal for MSIV closure.

5.4.4.2 Description

A main steamline flow restrictor (see Figure 5.4-7) is provided for each of the four main steam lines. The restrictor is a complete assembly welded into the main steamline. It is located upstream of the MSIVs.

The restrictor limits the coolant blowdown rate from the reactor vessel in the event a main steamline break occurs outside the containment to the maximum (choke) flow of 6.98×10^6 lb/hr at 1050 psia upstream pressure. The restrictor assembly consists of a venturi-type nozzle insert welded, in accordance with applicable code requirements, into the main steamline. The flow restrictor is designed and fabricated in accordance with ASME "Fluid Meters," 5th edition, 1959.

The flow restrictor has no moving parts. Its mechanical structure can withstand the velocities and forces associated with a main steamline break. The maximum differential pressure is conservatively assumed to be 1375 psi, the reactor vessel ASME Code limit pressure.

The ratio of venturi throat diameter to steamline inside diameter of approximately 0.5 results in a maximum pressure differential (unrecovered pressure) of about 10 psi at 100% of rated flow. This design limits the steam flow in a severed line less than 200% rated flow, yet it results in negligible increase in steam moisture content during normal operation. The restrictor is also used to measure steam flow to initiate closure of the main steamline isolation valves when the steam flow exceeds preselected operational limits.

5.4.4.3 Safety Evaluation

In the event a main steamline should break outside the containment, the critical flow phenomenon would restrict the steam flow rate in the venturi throat to 200% of the rated value. Prior to isolation valve closure, the total coolant losses from the vessel are not sufficient to cause core uncovering and the core is thus adequately cooled at all times.

Analysis of the steamline rupture accident (see Chapter 15) shows that the core remains covered with water during the time required for MSIV closure and that the amount of radioactive materials released to the environs through the main steamline break does not exceed the guideline values of published regulations.

Tests on a scale model determined final design and performance characteristics of the flow restrictor. The characteristics include maximum flow rate of the restrictor corresponding to the accident conditions, unrecoverable losses under normal plant operating conditions, and discharge moisture level. The tests showed that flow restriction at critical throat velocities is stable and predictable.

The steam flow restrictor is exposed to steam of 1/10 to 2/10% moisture flowing at velocities of approximately 150 ft/sec (steam piping ID) to 600 ft/sec (steam restrictor throat). ASTM A351 (Type CF8) cast stainless steel was selected for the upstream insert steam flow restrictor material because it has excellent resistance to erosion-corrosion in a high velocity steam atmosphere. The excellent performance of stainless steel in high velocity steam appears to be due to its resistance to corrosion. A protective surface film forms on the stainless steel which prevents any surface attack and this film is not removed by the steam.

Hardness has no significant effect on erosion-corrosion. For example, hardened carbon steel or alloy steel will erode rapidly in applications where soft stainless steel is unaffected.

Surface finish has a minor effect on erosion-corrosion. If very rough surfaces are exposed, the protruding ridges or points will erode more rapidly than a smooth surface. Experience shows that a machined or a ground surface is sufficiently smooth and that no detrimental erosion will occur.

5.4.4.4 Inspection and Testing

Because the flow restrictor forms a permanent part of the main steamline piping and has no moving components, no testing program is planned. Only very slow erosion will occur with time, and such a slight enlargement will have no safety significance. Stainless steel resistance to corrosion has been substantiated by turbine inspections at the Dresden Unit 1 facility, which have revealed no noticeable effects from erosion on the stainless steel nozzle partitions. The Dresden inlet velocities are about 300 ft/sec and the exit velocities are 600 to 900 ft/sec. However, calculations show that, even if the erosion rates are as high as 0.004 in. per year, after 60 years of operation the increase in restrictor choked flow rate would be no more than 5%. A 7.8% increase in the radiological dose calculated for the postulated main steamline break accident is not significant.

5.4.5 MAIN STEAMLINE ISOLATION SYSTEM

5.4.5.1 Safety Design Bases

The main steamline isolation valves are engineered safety features and, individually or collectively, shall:

(1) Close the main steamlines within the time established by design basis accident analysis to limit the release of reactor coolant.

(2) Close the main steamlines slowly enough that simultaneous closure of all steamlines will not induce transients that exceed the nuclear system design limits.

(3) Close the main steamline when required despite single failure in either valve or in the associated controls, to provide a high level of reliability for the safety function.

(4) Use separate energy sources as the motive force to close independently the redundant isolation valves in the individual steamlines.

(5) Use local stored energy (compressed air and springs) to close at least one isolation valve in each steam pipeline without relying on the continuity of any variety of electrical power to furnish the motive force to achieve closure.

(6) Be able to close the steamlines, either during or after seismic loadings, to assure isolation if the nuclear system is breached.

(7) Have capability for testing, during normal operating conditions, to demonstrate that the valves will function.

5.4.5.2 Description

Two isolation valves are welded in a horizontal run of each of the four main steam pipes; one valve is as close as possible to the inside of the drywell and the other is just outside the primary containment. The length of main steam pipe between the inner and outer MSIVs is approximately 14' 9-1/2" for main steam lines A and D, and approximately 16' 3-1/2" for main steam lines B and C. Inner and outer MSIVs are both 5'3" in length.

Each main steamline isolation valve is a 26 in. Y-pattern, globe valve. Uprated steam flow through each valve is 4.1.35 x 10^6 lb/hr. The main disc or poppet is attached to the lower end of the stem. Normal steam flow tends to close the valve, and higher inlet pressure tends to hold the valve closed. The bottom end of the valve stem closes a small pressure balancing hole in the poppet. When the hole is open, it acts as a pilot valve to relieve differential pressure forces on the poppet. Valve stem travel is sufficient to give flow areas past the wide open poppet approximately equal to the seat port area. The poppet travels approximately 90% of the valve stem travel to close the main disc; approximately the last 10% of travel to close the pilot hole. The air cylinder can open the poppet with a maximum differential pressure of 200 psi across the isolation valve in a direction that tends to hold the valve closed.

A 45-degree angle permits the inlet and outlet passages to be streamlined; this minimizes pressure drop during normal steam flow and helps prevent debris blockage. The pressure drop at 100% of rated flow is 7.8 psi maximum. The valve stem penetrates the valve bonnet through a stuffing box that has a single set of square graphite packing. To help prevent leakage through the stem packing, the poppet backseats when the valve is fully open.

Attached to the upper end of the stem is an air cylinder that opens and closes the valve and a hydraulic dashpot that controls its speed. The speed is adjusted by a valve in the hydraulic return line bypassing the dashpot piston. Valve closing time is adjustable to between 3 and 10 seconds.

The air cylinder is supported on the valve bonnet by actuator support and spring guide shafts. Helical springs around the spring guide shafts provide additional closing force. The motion of the spring seat member actuates a scram switch in the 90% open valve position and indicator light switches in the 90% open and 10% open valve positions. The valve is operated by pneumatic pressure and by the action of compressed springs. The control unit is attached to the air cylinder. This unit contains three types of control valves - pneumatic, ac, and ac from another source that open and close the main valve and exercise it at slow speed. Remote manual switches in the control room enable the operator to operate the valves.

Operating air is supplied to the valves from the plant air system. An air tank between the control valve and a check valve provides backup operating air. The MSIV actuator (cylinder plus spring) and backup air tank are sized to close the MSIV in 10 seconds while isolated from the Containment Instrument Gas system, concurrently with the containment pressurized to analyze DBA conditions. For specific accident breaks, the effects of the LOCA fluid jet could entirely remove the pneumatic assist for the inboard MSIVs. Under these jet impingement effects, the springs alone are capable of fully closing the inboard valves within 50 seconds under the same DBA conditions. See Section 3.6.1.2 for the evaluation, and acceptance criteria regarding the effects of jet impingement.

Each valve is designed to accommodate saturated steam at plant operating conditions, with a moisture content of approximately 0.25%, an oxygen content of 30 ppm, and a hydrogen

content of 4 ppm. The valves are furnished in conformance with a design pressure and temperature rating in excess of plant operating conditions to accommodate plant overpressure conditions.

In the worst case if the main steamline should rupture downsteam of the valve, steam flow would quickly increase to 169.5% of rated flow. Further increase is prevented by the venturi flow restrictor inside the containment.

During approximately the first 75% of closing, the valve has little effect on flow reduction, because the flow is choked by the venturi restrictor. After the valve is approximately 75% closed, flow is reduced as a function of the valve area versus travel characteristic.

The design objective for the valve is a minimum of 40 years service at the specified operating conditions. Operating cycles are estimated to be 100 cycles per year during the first year and 50 cycles per year thereafter. These cycles bound the cycles projected for the period of extended operation.

In addition to minimum wall thickness required by applicable codes, a corrosion allowance of 0.120-in. minimum is added to provide for 40-years service. Considering increased corrosion allowance for extended operation, there is sufficient margin to minimum wall thickness to allow extended service to 60 years.

Design specification ambient conditions for normal plant operation are 135°F normal temperature, 150°F maximum temperature, 100% humidity, in a radiation field of 15 rad/hr gamma and 25 rad/hr neutron plus gamma, continuous for design life. The inside valves are not continuously exposed to maximum conditions, particularly during reactor shutdown, and valves outside the primary containment and shielding are in ambient conditions that are considerably less severe.

The main steamline isolation valves are designed to close under accident environmental conditions referenced in section 3.11.

To resist sufficiently the response motion from the safe shutdown earthquake, the main steamline valve installations are designed as Seismic Category I equipment. The valve assembly is manufactured to withstand the safe shutdown earthquake forces applied at the mass center of the extended mass of the valve operator, assuming the cylinder/spring operator is cantilevered from the valve body and the valve is located in a horizontal run of pipe. The stresses caused by horizontal and vertical seismic forces are assumed to act simultaneously. The stresses in the actuator supports caused by seismic loads are combined with the stresses caused by other live and dead loads, including the operating loads. The allowable stress for this combination of loads is based on the allowable stress set forth in applicable codes. The parts of the main steam isolation valves that constitute a process fluid pressure boundary are designed, fabricated, inspected, and tested as required by the ASME Code, Section III.

5.4.5.3 Safety Evaluation

In a direct cycle nuclear power plant the reactor steam goes to the turbine and to other equipment outside the containment. Radioactive materials in the steam are released to the environs through process openings in the steam system or escape from accidental openings. A

large break in the steam system can drain the water from the reactor core faster than it is replaced by feedwater.

The analysis of a complete, sudden steamline break outside the containment is described in Chapter 15. The analysis shows that the fuel barrier is protected against loss of cooling if main steam isolation valve closure is within specified limits including instrumentation delay to initiate valve closure after the break. The calculated radiological effects of the radioactive material assumed to be released with the steam are shown to be well within the guideline values for such an accident.

The shortest closing time (approximately 3 sec.) of the main steam isolation valves is also shown in Chapter 15, to be satisfactory. The switches on the valves initiate reactor scram when specific conditions (extent of valve closure, number of pipe lines included, and reactor power level) are exceeded (see Subsection 7.2.1). The pressure rise in the system from stored and decay heat may cause the nuclear system relief valves to open briefly, but the rise in fuel cladding temperature will be insignificant. No fuel damage results.

The ability of this 45-degree, Y-design globe valve to close in a few seconds after a steamline break, under conditions of high pressure differentials and fluid flows with fluid mixtures ranging from mostly steam to mostly water, has been demonstrated in a series of dynamic tests. A full-size, 20-inch valve was tested in a range of steam-water blowdown conditions simulating postulated accident conditions (Reference 5.4-2).
The following specified hydrostatic, leakage, and stroking tests, as a minimum, are performed by the valve manufacturer in shop tests:

(1) To verify its capability to close between 3 and 10 sec, each valve is tested at rated pressure (1000 psig) and no flow. The valve is stroked several times, and the closing time is recorded. The valve is closed by spring only and by the combination of air cylinder and springs. The closing time is slightly greater when closure is by springs only.

(2) Leakage is measured with the valve seated and backseated. The specified maximum seat leakage, using cold water at design pressure, is 2 cm^3/hr/in. of nominal valve size. In addition, an air seat leakage test is conducted using 50 psi pressure upstream. Maximum permissible leakage is 0.1 scfh/in of nominal valve size. There must be no visible leakage from either set of stem packing at hydrostatic test pressure. The valve stem is operated a minimum of three times from the closed position to the open position, and the packing leakage still must be zero by visual examination.

(3) Each valve is hydrostatically tested in accordance with the requirements of the applicable edition and addenda of the ASME Code. During valve fabrication, extensive nondestructive tests and examinations are conducted. Tests include radiographic, liquid penetrant, or magnetic particle examinations of casting, forgings, welds, hardfacings, and bolts.

(4) The spring guides, the guiding of the spring seat member on support shafts, and rigid attachment of the seat member assure correct alignment of the actuating components. Binding of the valve poppet in the internal guides is prevented by making the poppet in the form of a cylinder longer than its diameter and by applying stem force near the bottom of the poppet.

After the valves are installed in the nuclear system, each valve is tested in accordance with the requirements of Chapter 14.

Two isolation valves provide redundancy in each steamline so either can perform the isolation function, and either can be tested for leakage after the other is closed. The inside valve, the outside valve, and their respective control systems are separated physically.

The design of the isolation valve has been analyzed for earthquake loading. The cantilevered support of the air cylinder, hydraulic cylinder, springs, and controls is the key area. The increase in loading caused by the specified earthquake loading does not result in stresses exceeding ASME allowable, or prevent the valve from closing as required.

Electrical equipment that is associated with the isolation valves and operates in an accident environment is limited to the wiring, solenoid valves, and position switches on the isolation valves. The expected pressure and temperature transients following an accident are discussed in Chapter 15.

5.4.5.4 Inspection and Testing

The main steam isolation valves can be functionally tested for operability during plant operation and refueling outages. The test provisions are listed below. During refueling outage the main steam isolation valves can be functionally tested, leak-tested, and visually inspected.

The main steam isolation valves can be tested and exercised individually to the 90% open position, because the valves still pass rated steam flow when 90% open.

The main steamline isolation valves can also be tested and exercised individually to the fully closed position if reactor power is reduced sufficiently to avoid scram from reactor overpressure or high flow through the steamline flow restrictors.

Leakage from the valve stem packing will become suspect during reactor operation from measurements of leakage into the drywell, or from observations or similar measurements in the steam tunnel.

The leak rate through the pipeline valve seats (pilot and poppet seats) can be measured accurately during shutdown by either of the following described procedures:

(1) With the reactor at approximately 125°F and normal water level and decay heat being removed by the RHR system in the shutdown cooling mode, all main steam isolation valves are closed utilizing both spring force and air pressure on the operating cylinder.

(2) A full peak accident pressure, 48.6 psig, test in the accident direction by pressurizing the entire Reactor Vessel to test pressure, or using qualified steam line plugs.

(3) A one-half peak accident pressure, 24.3 psig, test by pressurizing between each inboard and outboard MSIV.

During prestartup tests following an extensive shutdown, the valves will receive the same system leakage/hydrostatic pressure test (approximately 1035 psig) that is imposed on the primary system.

Such a test and leakage measurement program ensures that the valves are operating correctly and that a leakage trend is detected.

5.4.6 REACTOR CORE ISOLATION COOLING SYSTEM

5.4.6.1 Design Bases

5.4.6.1.1 Residual Heat and Isolation

5.4.6.1.1.1 Residual Heat

The RCIC system shall initiate and discharge a specified constant flow into the reactor vessel over a specified pressure range within a 30 second time interval. The RCIC water discharged into the reactor vessel varies between a temperature of 40°F up to and including a temperature of 140°F.

Redundantly the HPCI system performs the same function, hence, providing single failure protection. Both systems use different electrical power sources of high reliability, which permit operation with either on site power or offsite power. Additionally, the RHR system performs a residual heat removal function.

The RCIC system design is to include interfaces with redundant leak detection devices namely:

(1) A high pressure drop across a flow device in the steam supply line equivalent to 300 percent of the design RCIC turbine steam demand.

(2) A high area temperature, utilizing temperature switches as described in the leak detection system. High area temperature shall be alarmed in the control room.

(3) A low reactor pressure of 50 psig minimum.

(4) A high pressure between the turbine exhaust rupture diaphragms.

These devices, powered by the redundant power supplies, automatically isolate the steam supply to the RCIC turbine.

Other isolation bases are defined in Subsection 5.4.6.1.1.2.

5.4.6.1.1.2 Isolation

Isolation valve arrangements include the following:

(1) Two RCIC lines are to penetrate the coolant pressure boundary for the reactor. The first is the RCIC steamline which branches off one of the main steamlines between the reactor vessel and the main steam isolation valve. This line is to have two automatic motor operated isolation valves. One is located inside and the other outside primary containment. An automatic solenoid actuated inboard RCIC isolation bypass valve is also used. The isolation signals noted earlier close these valves.

(2) The RCIC pump discharge line is the other line, however it indirectly penetrates the reactor pressure vessel through the main feedwater line. The main feedwater line, described elsewhere, provides the required isolation valves inside and immediately outside the containment. The RCIC system provides the remote motor operated stop valve outside containment for isolation.

(3) The RCIC turbine exhaust line vacuum breaker system line is to have two automatic motor operated valves and two check valves. This line runs between the suppression pool air space and the turbine exhaust line down stream of the exhaust line check valve. Positive isolation shall be automatic via a combination of low reactor pressure and high drywell pressure.

The vacuum breaker valve complex is placed outside containment due to a more desirable environment. In addition, the valves are readily accessible for maintenance and testing.

(4) The RCIC pump suction line, minimum flow pump discharge line, and turbine exhaust line all penetrate the primary containment and are submerged in the suppression pool. The isolation valves for these lines are all outside primary containment and require remote-manual operation except the minimum flow valves which actuate automatically. Additionally, the turbine gland seal system vacuum pump discharges beneath the water level of the suppression pool after penetrating the primary containment. The isolation valve for the line is located outside primary containment and requires remote manual operation.

<u>5.4.6.1.2 Reliability, Operability, and Manual Operation</u>

<u>5.4.6.1.2.1 Reliability and Operability (Also see subsection 5.4.6.2.4)</u>

The RCIC System as noted in Table 3.2-1 is designed commensurate with the safety importance of the system and its equipment. Each component is individually tested to confirm compliance with system requirements. The system as a whole was tested during both the startup and pre-operational phases of the plant to set a base mark for system reliability. To confirm that the system maintains this mark, functional and operability testing is performed at predetermined intervals throughout the life of the reactor plant.

A design flow functional test of the RCIC system may be performed during normal plant operation by drawing suction from the condensate storage tank and discharging through a full flow test return line to the condensate storage tank. The discharge valve to the reactor vessel (via feedwater system) remains closed during the test, and reactor operation remains undisturbed. All components of the RCIC System shall be capable of individual functional testing during normal plant operation. Control system decision shall provide automatic return from test to operating mode if system initiation is required. There are three exceptions: 1) Auto/manual initiation on the flow controller. This feature is required for operator flexibility during system operation. 2) Steam inboard/outboard isolation valves. Closure of either or both of these valves requires operator action to properly sequence their opening (see Subsection 5.4.6.2.5.1). An alarm sounds when either of these valves leaves the fully open position. 3) Other bypassed or otherwise deliberately rendered inoperable parts of the system which affect the capability to perform a safety function shall be automatically indicated in the control room at the system level. Capability shall exist to manually initiate indication of system inoperability for

manual initiation of system level indication shall exist for items not readily automated.

5.4.6.1.2.2 Manual Operation (Also see Subsection 5.4.6.2.5.2)

In addition to the automatic operational features, provisions shall be included for remote-manual startup, operation, and shutdown of the RCIC System, provided initiation or shutdown signals do not exist.

5.4.6.1.3 Loss of Offsite Power

The RCIC System power is to be derived from a highly reliable source that is maintained by either onsite or offsite power. (Refer to Subsection 5.4.6.1.1)

5.4.6.1.4 Physical Damage

The system is designed to the requirements of Table 3.2-1 commensurate with the safety importance of the system and its equipment. (Also see Subsection 5.4.6.2.4)

5.4.6.1.5 Environment

The system is to operate for the time intervals and the environmental conditions specified in Section 3.11.

5.4.6.2 System Design

5.4.6.2.1 General

5.4.6.2.1.1 Description

The RCIC System consists of a turbine, pump, piping, valves, accessories, and instrumentation designed to assure that sufficient reactor water inventory is maintained in the reactor vessel to permit adequate core cooling to take place. This prevents reactor fuel overheating during the following conditions:

(1) Should the vessel be isolated and maintained in the hot standby condition.

(2) Should the vessel be isolated and accompanied by loss of coolant flow from the reactor feedwater system.

(3) Should a complete plant shutdown under conditions of loss of normal feedwater system be started before the reactor is depressurized to a level where the shutdown coolant system can be placed into operation.

Following a reactor scram, steam generation will continue at a reduced rate due to the core fission product decay heat. At this time the turbine bypass system will divert the steam to the main condenser, and the feedwater system will supply the make-up water required to maintain

reactor vessel inventory. In the event the reactor vessel is isolated, and the feedwater supply is unavailable, relief valves are provided to automatically (or remote manually) maintain vessel pressure within desirable limits. The water level in the reactor vessel will drop due to continued steam generation by decay heat.

Upon reaching a predetermined low level, the RCIC System is initiated automatically. The turbine driven pump will supply demineralized make- up water from the condensate storage tank to the reactor vessel; an alternate source of water is available from the suppression pool. When a predetermined low water level in the CST is reached, determined by conservative NPSH calculations, RCIC pump suction is automatically transferred to the suppression pool. This suction transfer can be remotely overridden to realign suction to the CST. The turbine is driven with a portion of the decay heat steam from the reactor vessel, and will exhaust to the suppression pool.

During RCIC operation, the suppression pool acts as the heat sink for steam generated by reactor decay heat. This will result in a rise in pool water temperature. Heat exchangers in the Residual Heat Removal System are used to maintain pool water temperature within acceptable limits by cooling the pool water.

5.4.6.2.1.2 Diagrams

The following diagrams are included for the RCIC Systems.

(1) A schematic P&ID (Dwgs. M-149, Sh. 1 and M-150, Sh. 1 shows all components, piping, points where interface system and subsystems tie together, and instrumentation and controls associated with subsystem and component actuation.

(2) A schematic "Process Diagram" (Dwg. M1-E51-81, Sh. 1 shows temperatures, pressures, and flows for RCIC operation and system process data hydraulic requirements.

5.4.6.2.1.3 Interlocks

The following defines the various electrical interlocks:

(1) There are 4 key locked valves, the F007, F008, F059, and F060. There are two key locked reset switches, the RCIC Auto Isolation Signal A and B Reset Switches, and two key locked Bypass switches, the Division 1 and Division 2 MOV Overload Bypass switches.

(2) F031's limit switch activates when fully open and closes F010 and F022.

(3) F059's limit switch activates when full open and clears F045 permissive so F045 can open.

(4) F045's limit switches activate when the valve reaches an intermediate open position (approximately 40%). One limit switch stops the valve opening, another initiates a time delay relay. The relay times out in approximately seven seconds and it re-energizes the

F045 valve and the RCIC startup ramp function. The ramp function, the time delay relay and the limit switch reset each time F045 is closed.

(5) F045's limit switch activates when fully closed and permits F004, F005, F025 and F026 to open and closes F013 and F019. The switch also starts one (1) RCIC Room Cooler.

(6) The turbine trip throttle valve limit switch activates when fully closed and closes F013 and F019.

(7) The combined pressure switches at reactor low pressure and high drywell pressure when activated close F062 and F084.

(8) Either 110% overspeed, high turbine exhaust pressure, low pump suction pressure, or an isolation signal actuates and closes the turbine trip throttle valve. When signal is cleared, the trip throttle valve must be reset from control room.

(9) 122.4% overspeed trips the mechanical trip at the turbine and the trip throttle valve. The former is reset at the turbine and then the latter is reset in the control room.

(10) An isolation signal closes F007, F008, F088 and other valves as noted above in items 6 and 8.

(11) An initiation signal opens F010 if closed, F013 and F045; starts barometric condenser vacuum pump; and closes F022 if open.

(12) High and low inlet RCIC steamline drain pot levels, respectively, open and close F054.

(13) The combined signal of low flow plus pump discharge high pressure opens F019. F019 closes on increased flow. Also see items 5 and 6 above.

(14) Reactor high water level will close F045 and place the RCIC System in a partial standby configuration. The system will be ready to restart without any operator action if it receives a vessel low water level signal.

(15) A low water level in the Condensate Storage Tank (CST) will automatically switch the RCIC pump suction from the CST to the suppression pool.

5.4.6.2.2 Equipment and Component Description

5.4.6.2.2.1 Design Conditions

Operating parameters for the components of the RCIC System, listed below, are shown on Dwg. M1-E51-81, Sh. 1.

(1) One 100% capacity turbine and accessories

(2) One 100% capacity pump assembly and accessories

(3) Piping, valves, and instrumentation for:

a. Steam supply to the turbine

b. Turbine exhaust to the suppression pool

c. Supply from the condensate storage tank to the pump suction.

d. Supply from the suppression pool to the pump suction.

e. Pump discharge to the feedwater system, including a test line to the condensate storage tank, a minimum flow bypass line to the suppression pool, and a cooling water supply to accessory equipment.

The basis for the design conditions was the ASME Section III, Nuclear power plant components.

5.4.6.2.2.2 Design Parameters

Design parameters for the RCIC system components are listed below. See Dwgs. M-149, Sh. 1 and M-150, Sh. 1 for cross-references of component numbers listed below:

(1)	RCIC Pump Operation (P203)		
	Flow Rate	Injection Flow - 600 gpm Cooling Water Flow - 16 gpm	
	Water Temperature Range	40°F to 140°F (continuous system operation)	
	NPSH Required	21.3 ft maximum	
	Minimum NPSH Available (Suction from suppression pool and 1210 psia reactor pressure)	39.5 feet	
	Minimum Flow Condition (Minimum by-pass flow)	75 gpm	
	Minimum Discharge Pressure	125 psig	
	Developed Head Maximum	3060 ft @ 1225 psia Reactor Pressure 525 ft @ 165 psia Reactor Pressure	
	BHP, Not to Exceed	750 HP @ 3060 feet Developed Head 100 HP @ 525 feet Developed Head	
	Design Pressure	1500 psig	
(2)	RCIC Turbine Operation (S212)		
		H.P. Condition	L. P. Condition
	Reactor Press	1225 psia	165 psia
	(Sat. Temp.)		
	Steam Inlet Pressure	1210 psia	150 psia
	Turbine Exhaust Press	15-25 psia	15-25 psia
	Design Inlet Pressure	1250 psig + saturated temperature	
	Design Exhaust Pressure	165 psig + saturated temperature	

(3)	<u>RCIC Orifice Sizing</u> <u>Coolant Loop Orifice</u> (D009)	Size with piping arrangement to ensure maximum pressure of 75 psia at the lube oil cooler inlet, and a minimum pressure of 45 psia at the spray nozzles at the barometric condenser.
	Minimum Flow Orifices (D005 & 14905)	Size with piping arrangement to ensure minimum flow of 75 gpm with MO-F019 fully open.
	Test Return Orifices (D006 & 14906)	Size with piping arrangement to simulate pump discharge pressure required when the RCIC System is injecting design flow with the reactor vessel pressure at 165 psia.
	Leak-Off Orifices (D008 & D010A & B)	Size for 1/8 inch d minimum, 3/16 d maximum.
	Steam Exhaust Drain Pot Orifice (D004)	Size for 1/8 inch diameter minimum, 3/16 inch diameter maximum
	Suction Strainer Open Area Size (F401A & B)	Size to block particles 1/8"
(4)	<u>Valve Operation Requirements</u>	
	Steam Supply Valve (F045)	Close against full pressure within 15 seconds to minimize overfilling the reactor vessel when level reaches +54. Fully open against full pressure within 20 seconds to support achieving rated RCIC flow rate of 600 gpm within 30 seconds of system initiation. The valve control circuit initiates a time delay relay when the valve reaches 40% open. The relay times out in approximately seven seconds, and re-energizes the F045 to fully open. The time from fully closed to fully open (including time delay) shall be ≤ 20 seconds.
	Pump Discharge Valves (F012/F013)	Open and/or close against full pressure within 15 seconds.
	Pump Minimum Flow Bypass Valve (F019)	Open and/or close against full pressure within 5 seconds.
	Steam Supply Isolation Valves (F007/F008)	Close against full pressure at a minimum rate of 12 inches per minute.
	Cooling Water Pressure Control Valve (F015)	Self-contained downstream sensing control valve capable of maintaining constant downstream pressure of 75 psia.
	Pump Suction Relief Valve (F017)	180 psig Relief Setting; 24.1 gpm at 10% accumulation.
	Cooling Water Relief Valve (F018)	Size to prevent over pressurizing piping, valves and equipment in the cooling loop in the event of failure of pressure control valve F015.
	Pump Test Return Valve (F022)	Shall be capable of throttling against a differential pressure of 1371 psid.
	Relief Valve Barometric Condenser (F033)	Relief valve shall be capable of retaining 10 inches of mercury vacuum at 140°F ambient, with a set pressure of 5-7 psig and a flow of 20 gpm at 10% accumulation.
	Turbine Exhaust Isolation Valve(F059)	Shall open and/or close against 16 psi differential pressure at a temperature of 206°F. Physically locate at the highest point in the exhaust line on a horizontal run, as close to the containment as practical.

	Vacuum Pump Discharge Isolation (F060)	Shall open and/or close against 16 psi differential pressure at temperature of 206°F. Physically locate at the highest point in the line on a horizontal run, as close to the containment as practical.
	Check Valve, Vacuum Pump Discharge (F028)	Shall be located at the highest point in the line on a horizontal run, with adjacent piping arranged to provide a continuous downward slope, from the upstream side of the check valve to the barometric condenser and downstream of the check valve to the suppression pool.
	Check Valve, Turbine Exhaust (F040)	Shall be located at the highest point in the line on a horizontal run, with adjacent piping arranged to provide a continuous downward slope, from the upstream side of the check valve to the turbine exhaust drain pot and downstream of the check valve to the suppression pool.
	Vacuum Breaker Isolation Valves (F062 & F084)	Shall open and/or close against a differential pressure of 0 psi at a minimum rate of 12 inches per minute.
	Warmup Line Isolation Valve (F088)	Shall open and/or close against a full differential pressure of 1210 psi at a minimum rate of 12 inches per minute. The valve and valve associated equipment shall be capable of proper functional operation during maximum ambient conditions, refer to BWR Equipment Environmental Interface Data Reference in Paragraph 2.2.
	Vacuum Breaker Check Valves (F063 & F064)	Shall open with a minimum pressure drop (less than 0.5 psi) across the valve seat.
(5)	<u>Rupture Disc Assemblies</u> (D001, D002)	Utilized for turbine casing protection, includes a mated vacuum support to prevent rupture disc reversing.
	Rupture Pressure Flow Capacity	150 psig ± 10 psig 60,000 lb/hr @ 165 psig
(6)	<u>Condensate Storage Requirements</u>	135,000 gallons (Total) reserve storage, per unit, for both HPCI and RCIC Systems.
(7)	<u>Piping RCIC Water Temperature</u>	The maximum water temperature range for continuous system operation shall not exceed 140°F. However, due to potential short-term operation at higher temperatures, piping expansion calculations are based on 170°F.
(8)	<u>Ambient Conditions</u> Normal Plant Operation Isolation Conditions	<u>Temperature</u> <u>Relative Humidity</u> 60 to 100°F 95 148°F 100

5.4.6.2.3 Applicable Codes and Classifications

The RCIC system is classified as a Safe Shutdown System for an isolation event with a loss of feedwater. This classification requires RCIC to be safety related. FSAR Section 7.1.2a.1.18 discusses this information in detail. The RCIC System components within the drywell up to and including the outer isolation valve is designed in accordance with ASME Code, Section III, Class

1, Nuclear Power Plant Components. The RCIC System is also designed as Seismic Category I equipment.

The RCIC system component classifications and those for the condensate storage system are given in Table 3.2-1.

5.4.6.2.4 System Reliability Considerations

To assure that the RCIC will operate when necessary and in time to prevent inadequate core cooling, the power supply for the system is taken from immediately available energy sources of high reliability. Added assurance is given in the capability for periodic testing during station operation. Evaluation of reliability of the instrumentation for the RCIC shows that no failure of a single initiating sensor either prevents or falsely starts the system.

In order to assure HPCI or RCIC availability for the operational events noted previously, certain design considerations are utilized in design of both systems.

(1) Physical Independence. The two systems are located in separate areas in the reactor building. Piping runs are separated and the water delivered from each system enters the reactor vessel via different nozzles.

(2) Prime Mover Diversity and Independence. Prime mover independence is achieved by using separate steamlines to drive the HPCI and RCIC steam turbines. Additionally separate divisions of power are used for HPCI and RCIC.

(3) Control Independence. Control independence is secured by using different battery systems to provide control power to each unit. Separate detection initiation logics are also used for each system.

(4) Environmental Independence. Both systems are designed to meet Safety Class I requirements. Environment in the equipment rooms is maintained by separate auxiliary systems.

(5) Periodic Testing. A design flow functional test of the RCIC can be performed during plant operation by taking suction from the condensate storage tank and discharging through the full flow test return line back to the condensate storage tank. The discharge valve to the reactor feedwater line remains closed during the test, and reactor operation is undisturbed. Control system design provides automatic return from test to operating mode if system initiation is required during testing.

(6) General. Valve position indication and instrumentation alarms are displayed in the control room.

5.4.6.2.5 System Operations

Automatic and Manual actions required for the various modes of RCIC are defined below.

5.4.6.2.5.1 Automatic Operation

Automatic startup of the RCIC system due to an initiation signal from reactor low water level requires no operator action. To permit this automatic operation, the operator must verify that the following steps have been taken to prepare the system for the standby mode and correct as required. Further steps describe action during operation and shutdown.

(1) Verify the flow controller has the correct flow set point and is in the automatic mode.

(2) Verify that the turbine trip throttle valve is in the full open position. If not fully open, the valve may need to be reset.

There are two trips for the turbine. The mechanical overspeed trip actuates a mechanical trip linkage and requires the trip to be reset at the turbine itself before the trip throttle valve can be reopened. The electrical overspeed trip actuates a solenoid mounted on the trip throttle valve. Because the mechanical trip linkage is not actuated by the electrical overspeed trip, the turbine trip throttle valve may be reopened from the control room once the overspeed signal is cleared. See Dwgs. M-149, Sh. 1 and M-150, Sh. 1 for component identification.

(3) Verify power is available to all components.

(4) Verify that the two RCIC steam isolation valves have been properly sequenced open.

(5) Verify that the RCIC turbine exhaust line isolation valve and vacuum breaker valves are open.

(6) Verify that the two isolation logic "reset" devices have been reset.

(7) Verify condensate transfer system header pressure (keep-fill system). If pressure is not being maintained, the RCIC system pump can be started up and run in the test mode until the pressure is restored.

(8) Verify manual valves are positioned correctly and administratively controlled. This verification requires one to be out of control room. Administrative control will minimize subsequent checks.

(9) Verify water is available in the condensate storage tank.

(10) Verify oil is available in RCIC turbine oil reservoir; and the turbine and pump are ready to run as defined by the technical manuals for the turbine and pump.

(11) During extended periods of operation and when the normal water level is again reached, the HPCI system may be manually tripped and the RCIC system flow controller may be adjusted and switched to manual operation. This prevents unnecessary cycling of the two systems. Subsequent starts of RCIC turbine and pump must be operator controlled until rated flow is reached by use of the trip throttle valve or manually initiated if F045 is first closed. Note: Should RCIC flow be inadequate HPCI flow will automatically come on again.

(12) Adjust flow controller set point as required to maintain desired reactor water level.

(13) When RCIC operation is no longer required, manually trip the RCIC system and turn the flow controller back to automatic.

(14) Close the steam supply valve to turbine F045.

(15) Reset the turbine trip throttle valve.

(16) Stop the barometric condenser vacuum pump.

(17) Close the cooling water supply valve F046.

(18) Verify that valves F005, F025, and F026 reopen automatically after valve F045 was closed. Note: Valve F004 is normally closed and opens as required by signal from barometric condenser.

(19) Verify system is in the standby configuration per Dwgs. M-149, Sh. 1 and M-150, Sh. 1.

5.4.6.2.5.2 Test Loop Operation

This operating mode is manually initiated by the operator. Operator action is required as defined below:

(1) Verification made in steps 1 through 10 of Subsection 5.4.6.2.5.1 shall be completed.

(2) All motor operated valves shall be positioned as shown on Dwgs. M-149, Sh. 1 and M-150, Sh. 1.

(3) Open F059 and F022 fully.

(4) Start barometric condenser vacuum pump.

(5) Open F046.

(6) Open F045.

(7) Verify that valves F004, F005, F025 and F026 automatically closed after valve F045 opened.

(8) Adjust F022 to obtain a pump discharge pressure of 300 psig.

(9) Observe turbine RPM on speed indicator.

(10) Turn RMS switch for F019 to open position and release. Observe that valve F019 cycles fully open and closed by watching position lights. Also observe turbine speed indicator to verify speed increases during this cycling. If speed increases it confirms that the minimum flow line valves and electrical logic properly function.

(11) Further adjust F022 to simulate reactor pressure plus line losses to reactor pressure at time of test or actual line pressure drop to reactor (if available) plus reactor pressure.

(12) While turbine is running, check and record the following:

a. Pump Suction Pressure

b. Pump Discharge Pressure

c. Turbine Steam Exhaust Pressure

d. Turbine Steam Inlet Pressure

e. Pump Flow

f. Turbine Speed

(13) When the test is completed, manually trip the turbine.

(14) When the turbine speed indicator reaches "0" RPM, close the test bypass valve to the HPCI test return line which then goes to the condensate storage tank F022.

(15) Close redundant shut off valve E41-F011 to the HPCI test return line which then goes to the condensate storage tank.

(16) Follow steps 14 through 19 of Subsection 5.4.6.2.5.1.

5.4.6.2.5.3 Steam Condensing (Hot Standby) Operation

This section has been deleted.

5.4.6.2.5.4 Limiting Single Failure

The most limiting single failure with the RCIC and its HPCI backup system is the failure of HPCI. If the capacity of RCIC System with a HPCI failure is adequate to maintain reactor water level, the operator follows Subsection 5.4.6.2.5.1. If however, the RCIC capacity is inadequate, Subsection 5.4.6.2.5.1 applies, but additionally the operator may also initiate the ADS system described in Subsection 6.3.2.

5.4.6.3 Performance Evaluation

The analytical methods and assumptions in evaluating the RCIC system are presented in Chapter 15 and Appendix 15A. The RCIC system provides the flows required from the analysis (see Dwg. M1-E51-81, Sh. 1) within a 30 second interval based upon considerations noted in Subsection 5.4.6.2.4.

5.4.6.4 Preoperational Testing

The preoperational and initial startup test program for the RCIC system is presented in Chapter 14.

5.4.6.5 Safety Interfaces

The Balance of Plant-GE Nuclear Steam Supply System safety interfaces for the Reactor Core Isolation Cooling System are: (1) preferred water supply from the condensate storage tank; (2) all associated safety-related wire, cable, piping, sensors, and valves which lie outside the Nuclear Steam Supply System scope of supply. The air supply for solenoid actuated valves is a non-safety related interface.

5.4.7 RESIDUAL HEAT REMOVAL SYSTEM

5.4.7.1 Design Bases

The RHR system is comprised of two independent loops. Each loop contains two motor-driven pumps, a heat exchanger, piping, valves, instrumentation, and controls. The RHR heat exchangers are cooled by RHR service water. Each loop takes suction from the suppression pool and is capable of discharging water to the reactor vessel via a connection to the reactor recirculation loop, or back to the suppression pool via a full flow test line. Each loop can also take suction from a common line from one reactor recirculation loop. Each loop can take suction from and discharge to the fuel pool cooling system. Both loops can discharge to wetwell and drywell spray spargers. Also, one loop can discharge to the reactor vessel head spray line.

5.4.7.1.1 Functional Design Basis

The RHR system has five subsystems, each of which has its own functional requirements. Each subsystem shall be discussed separately to provide clarity.

5.4.7.1.1.1 Residual Heat Removal Mode (Shutdown Cooling Mode)

1. The function design basis of the shutdown cooling mode is to have the capability to remove heat from the reactor primary system so that the reactor coolant temperature can be reduced to 125°F after reactor shutdown, once the main condenser can no longer be used as effective heat sink.

2. With one loop in service the shutdown cooling mode of the RHR System is capable of cooling the reactor cooling system to 200 within 24 hours after shutdown. If normal shutdown cooling can not be established, the alternative shutdown cooling systems described in section 15.2.9 are capable of acceptable shutdown heat removal.

A cross-tie line exists between the loop ‘A’ and ‘B’ LPCI injection lines and the RHR Shutdown Cooling suction line. The 1 inch line provides a positive pressure drop across the LPCI injection check valves HV151F050A and HV151F050B to maintain the valves in a closed position during normal operation. Should leakage occur past the seats of the LPCI injection check valves, the excess fluid is diverted through the cross-tie line and returned to the Reactor Recirculation System (RRS) via the RHR System Shutdown Cooling suction line.

During Shutdown Cooling (SDC) operation a portion of the flow is diverted from ‘A’ and/or ‘B’ loop injection lines through the cross-tie line. The quantity of flow diverted is less than the

excess flow available during SDC operations and does not impede the capabilities of the RHR System from performing its SDC function.

5.4.7.1.1.2 Low Pressure Coolant Injection (LPCI) Mode

The functional design basis of the LPCI mode is to flood the reactor core when reactor pressure is low. One or more of the four motor-driven RHR pumps are used to pump water from the suppression pool into the reactor vessel via the recirculation loop. This ECCS mode is automatically initiated by low reactor water level (Level 1) or high drywell pressure. Equipment characteristics used for the LOCA analysis in FSAR Subsection 6.3.3.7.1 are described in Subsection 6.3.2.2.4. RHRSW can be aligned to an RHR heat exchanger in this mode of operation to ensure that the long term peak suppression pool temperature following a design basis LOCA remains within design limits, as evaluated in the SSES containment analysis described in section 6.2.1 of the FSAR. With one LPCI injection pump in service in an RHR loop, an RHR heat exchanger may be aligned to support containment cooling by limiting LPCI injection flow to 10,000 gpm and directing flow through the RHR heat exchanger by closing the HV151F048A(B) valve. Only one RHR heat exchanger is credited for long term cooling in the SSES containment analysis.

A cross-tie line exists between the loop 'A' and 'B' LPCI injection lines and the RHR Shutdown Cooling suction line. The 1 inch line provides a positive pressure drop across the LPCI injection check valves HV151F050A and HV151F050B to maintain the valves in a closed position during normal operation. Should leakage occur past the seats of the LPCI injection check valves, the excess fluid is diverted through the cross-tie line and returned to the Reactor Recirculation System (RRS) via the RHR System Shutdown Cooling suction line.

During LPCI operation a portion of the flow is diverted from the 'A' and/or 'B' loop injection lines through the cross-tie line. The quantity of flow diverted is less than the excess flow available during either one or two pump LPCI operations and does not impede the capabilities of the RHR System from performing its LPCI safety function.

5.4.7.1.1.3 Suppression Pool Cooling Mode

The functional design basis of the suppression pool cooling mode is sufficient cooling capacity to ensure that the long-term peak suppression pool temperature following a design basis LOCA remains within design basis limits. This mode may be used during normal plant operation, during a transient, or after a LOCA to remove heat from the containment. This mode is initiated and terminated via remote manual control from the control room. See Subsection 6.2.1.1.3 and Table 6.2-6.

5.4.7.1.1.4 Containment Spray Cooling Mode

The functional design basis of the containment spray cooling mode is to provide a redundant means of condensing vapor and cooling the drywell and suppression pool vapor space to control the containment pressure within design limits.

5.4.7.1.1.5 Reactor Steam Condensing Mode

This section has been deleted.

5.4.7.1.1.6 Fuel Pool Cooling Mode

The functional design basis for the fuel pool cooling mode is as follows:

a) The RHRFPC mode is designed and operated to provide cooling such that the fuel pool will be maintained at or below 125°F when the Emergency Heat Load (EHL) is resident in an isolated fuel pool. The EHL can be removed with a RHRSW inlet temperature of 89°F with only one RHR pump and heat exchanger. For crosstied fuel pools, one RHR pump and heat exchanger in one unit in combination with the normal Fuel Pool Cooling System from the adjacent unit is sufficient to maintain the fuel pools at or below 125°F with the EHL resident in one fuel pool and fuel at the scheduled offload rate in the other fuel pool. This function is described in Section 9.1.3.1b and 9.1.3.2.

b) The RHRFPC mode is designed and operated to provide sufficient cooling to prevent fuel pool boiling in the event that a seismic event causes an extended loss of both units' normal fuel pool cooling systems. This capability exists for both crosstied and isolated fuel pools.

When one RHR pump is operated in the RHRFPC mode, the spent fuel pool level must be raised to a minimum level above the weirs in order to support the design flowrate of this mode. Additional details describing this mode of RHR are contained in Sections 5.4.7.2.6c, 9.1.3.1c, 9.1.3.2, and 9.1.3.3.

5.4.7.1.2 Design Basis for Isolation of RHR System from Reactor Coolant System

The low pressure portions of the RHR system, are isolated from full reactor pressure whenever the primary system pressure is above the RHR system design pressure. See Subsection 5.4.7.1.3 for further details. In addition, automatic isolation may occur for reasons of vessel water inventory retention which is unrelated to line pressure rates. (See Subsection 5.2.5 for an explanation of the Leak Detection System and the isolation signals.) Reactor Coolant pressure boundary valves are subject to inservice inspection leakage testing requirements as provided in 10CFR50.55a (see Subsection 3.9.6).

The RHR pumps are protected against damage from a closed discharge valve by means of automatic minimum flow valves, which open on low main line flow and close on high main line flow.

5.4.7.1.3 Design Basis For Pressure Relief Capacity

The relief valves in the RHR system are sized on one of three bases:

(1) Thermal relief only

(2) Valve bypass leakage only

(3) Control valve failure and the subsequent uncontrolled flow which results.

Transients are treated by items (1) and (3); item (2) above has resulted from an excessive leak past isolation valves. RHR System pressure relief valves are set to assure that the maximum expected pressure from the worst case overpressure event does not exceed the ASME code allowable pressure for the ECCS piping.

Redundant interlocks prevent opening valves to the low pressure suction piping when the reactor pressure is above the shutdown range. These same interlocks initiate valve closure on increasing reactor pressure.

In addition a high pressure check valve will close to prevent reverse flow from the reactor if the pressure should increase. Relief valves in the discharge piping are sized to account for leakage past the check valve.

5.4.7.1.4 Design Basis With Respect to General Design Criteria 5

The RHR system for each unit does not share equipment or structures with the other nuclear unit except for the Spent Fuel Pools as discussed in Subsection 9.1.3.3. They also share the common Emergency Service Water System. Sharing of this system with respect to General Design Criteria 5 is discussed in Section 3.1.2.1.5.

5.4.7.1.5 Design Basis for Reliability and Operability

The design basis for the Shutdown Cooling mode of the RHR system is that this mode is controlled by the operator from the control room. The only operations performed outside of the control room for a normal shutdown are manual operations of local flushing water admission valves, which are the means of providing clean water to the shutdown portions of the RHR system. These portions are flushed with reactor water, suppression pool water or condensate to minimize the input of potentially harmful impurities to the reactor coolant system.

Two separate shutdown cooling loops are provided. The design basis is that, with both loops in operation, reactor coolant temperature will be reduced below 125°F after all control rods are inserted. This includes the time to depressurize the reactor, flush and preheat the RHR System. The reactor coolant can be brought below 212°F within 24 hours with only one loop in operation. With the exception of the shutdown suction, vessel head spray, and steam supply and condensate discharge lines, the entire RHR system is part of the ECCS and containment cooling systems, and is therefore required to be designed with redundancy, flooding protection, piping protection, power separation, etc. required of such systems. (see Section 6.3 for an explanation of the design bases for ECCS systems). Shutdown suction and discharge valves are required to be powered from both offsite and standby emergency power for purposes of

isolation and shutdown following a loss of offsite power. In the event either of the two shutdown supply valves fail to operate, and the shutdown supply valves cannot be opened by hand, alternate shutdown cooling is established in accordance with plant procedures. If repairs are required to the shutdown suction valves, the line can be isolated using manual valve F067 provided containment access is possible. Residual heat is absorbed by the main condenser or by the suppression pool with pool cooling by the RHR system while repairs are in process.

5.4.7.1.6 Design Basis for Protection from Physical Damage

See Section 3.12 for discussion.

5.4.7.2 Systems Design

5.4.7.2.1 System Diagrams

All of the components of the RHR system are shown in the P&ID, Dwgs. M-151, Sh. 1, M-151, Sh. 2, M-151, Sh. 3, and M-151, Sh. 4. A description of the controls and instrumentation is presented in Subsection 7.3.1.1.

A process diagram and process data for the RHR System are shown in Dwgs. M1-E11-3, Sh. 1 and M1-E11-3, Sh. 2. All of the sizing modes of the system are shown in the process data. The FCD for the RHR system is provided in Chapter 7.

When the system is operated from the control room, interlocks are provided: (1) to prevent drawing vessel water to the suppression pool; (2) to prevent opening vessel suction valves above the suction line design pressure or the discharge line design pressure, with the pumps at shutoff head; (3) to prevent inadvertent opening of drywell spray valves while in shutdown; and (4) to prevent pump start (i.e. maintaining a pump trip signal) when suction valve(s) are not open.

5.4.7.2.2 Equipment and Component Description

a. System Main Pumps

The RHR main system pumps are motor-driven deepwell pumps with mechanical seals. The motors are water cooled by the Emergency Service Water System. The pumps are sized on the basis of the LPCI mode (Mode A) and the minimum flow mode (Mode J) on Dwgs. M1-E11-3, Sh. 1 and M1-E11-3, Sh. 2. Design pressure for the pump suction structure is 220 psig with a temperature range from 40 to 360°F. Design pressure for the pump discharge structure is 500 psig. The bases for the design temperature and pressure is maximum shutdown cut-in pressures and temperature, minimum ambient temperature, and maximum shutoff head. The pump pressure vessel is carbon steel, the shaft and impellers are stainless steel. The required pump NPSH can be obtained from the pump characteristic curves provided in Figure 6.3-119. Available NPSH is provided in Section 6.3.2.2.4.1.

b. Heat Exchangers

The RHR system heat exchangers are sized on the basis of the duty for the shutdown cooling mode (Mode F on Dwg. M1-E11-3, Sh. 1). All other uses of these exchangers

require less cooling surface. The RHR heat exchanger data provided by GE is as follows:

Flow rates are 10,000 gpm (rated) on the shell side and 9,000 gpm (rated) on the tube side (service water side). Rated inlet temperature is 125°F shell side and 85°F tube side. The overall heat transfer coefficient is 218 BTU per hour per square foot per °F. The exchangers contain 7593 ft^2 of effective surface. Design temperature range of shell side is 40°F to 470°F and tube side is 32°F to 470°F. Design pressure is 450 psig on both sides, fouling factors are 0.0005 shell side and 0.002 tube side. The construction materials are carbon steel for the pressure vessel with 70-30 copper-nickel tubes and carbon steel tube sheet clad with copper nickel.

c. Valves

All of the directional valves in the system are conventional gate, globe, and check valves designed for nuclear service. The injection valves, reactor coolant isolation valves, and pump minimum flow valves are high speed valves, as operation for LPCI injection or vessel isolation requires. Valve pressure ratings are as necessary, to provide the control or isolation function; i.e., all vessel isolation valves are rated as Class 1 nuclear valves rated at the same pressure as the primary system.

d. RHR Suction Strainers

Each 24" RHR pump suction line penetrates the vertical wall of the suppression pool, leading directly to a vertical "T" arrangement whose centerline is 23" from the pool wall. Two RHR high capacity stacked disk suction strainers are mounted on the 24" "T" for each RHR pump. These strainers replaced the original conical design. Each of the two strainers provides a flow area of 204 ft^2. The strainers have sufficient capacity to filter their design debris source term under worst case conditions while maintaining strainer pressure drop below the maximum value required to provide adequate NPSH and system flow. The design debris source term consists of conservative amounts of insulation, paint chips and other drywell debris that is assumed to be destroyed by LOCA jet forces and transported to the suppression pool through the downcomers. This debris is assumed to be filtered by the strainers along with corrosion products that would exist in the suppression pool prior to a LOCA. The stacked disk strainers are designed for a maximum pressure drop of 2.5 psi at a flow of 13,800 gpm while filtering their design debris source term. Correlations between the amount of debris filtered by the strainers and strainer pressure drop are based on testing performed on one of the Susquehanna strainers and NRC approved methodology outlined in NEDO-32686, “Utility Resolution Guide for ECCS Suction Strainer Blockage”. The suppression pools are cleaned and inspected periodically to maintain corrosion product amounts at acceptable levels and to confirm the absence of miscellaneous debris that would be a strainer blockage threat.

The strainer mesh size is 0.125” +/- 0.005” which screens out all particles greater than 1/8” nominal diameter. This criteria is used in conjunction with the design of the drywell and suppression pool containment spray nozzles, which have a free passage diameter of 0.125." Particles equal to or smaller than 1/8" in size would have no effect on RHR pump operation.

The minimum height of the suppression pool water level above the "T" centerline is 11'-11". The available and required NPSH are provided in Section 6.3.2.2.4.1.

The RHR suction strainers are shown in Figures 5.4-4A and 5.4-4B.

ESF Portions of the RHR System

The ECCS (LPCI) portions of the RHR system include those sections required to operate Modes A, B, and G on Dwg. M1-E11-3, Sh. 1.

The route includes suppression pool suction strainers, suction piping, RHR pumps, RHR heat exchangers, discharge piping injection valves, and drywell piping to the reactor recirc discharge lines.

Pool cooling components include pool suction strainers, suction piping, pumps, heat exchangers and pool return lines required to operate Mode D on Dwg. M1-E11-3, Sh. 1.

Containment spray components required for Modes C-1 and C-2 on Dwg. M1-E11-3, Sh. 1 are the same as pool cooling except that the spray headers replace the pool return lines.

5.4.7.2.3 Controls and Instrumentation

Controls and instrumentation for the RHR system are described in Chapter 7. The RHR system incorporates relief valves to protect the components and piping from inadvertent overpressure conditions. The relief valve set point, capacity and method of collection are shown in Table 5.4-3.

5.4.7.2.4 Applicable Codes and Classifications

Refer to Sections 3.9 and 3.10 for discussion of applicable codes and standards.

5.4.7.2.5 Reliability Considerations

The Residual Heat Removal System has included the redundancy requirements of Subsection 5.4.7.1.5. Two completely redundant loops have been provided to remove residual heat, each powered from a separate, emergency bus. With the exception of the common shutdown line, all mechanical and electrical components are separate. Either loop is capable of shutting down the reactor within a reasonable length of time.

A cross-tie line exists between the loop ‘A’ and ‘B’ LPCI injection lines and the RHR Shutdown Cooling suction line. The 1 inch line provides a positive pressure drop across the LPCI injection check valves HV151F050A and HV151F050B to maintain the valves in a closed position during normal operation. Should leakage occur past the seats of the LPCI injection check valves, the excess fluid is diverted through the cross-tie lien and returned to the Reactor Recirculation System (RRS) via the RHR System Shutdown Cooling suction line.

During LPCI or Shutdown Cooling (SDC) operations of a portion of the flow is diverted from the ‘A’ and/or ‘B’ loop injection lines through the cross-tie line. The quantity of flow diverted is less than the excess flow available during LPCI or SDC operations and does not impede the capabilities of the RHR System from performing its SDC or LPCI safety function.

5.4.7.2.6 Manual Action

a. Residual Heat Removal (Shutdown Cooling Mode)

In shutdown cooling operation, when vessel pressure is below the shutdown cooling cut-in permissive, the system is prewarmed and stagnant water is flushed to the condenser hotwell or radwaste via valves F040 and F049 which are operated from the control room. Following verification that acceptable differential temperatures exist between the RPV steam dome and the bottom head drain; between the reactor coolant contained in an idle recirculation loop and the RPV; and that the RHR lines are full, the RHRSW system is placed in service. With the RHR Heat exchanger inlet (F047) and bypass valves (F048) open and the outlet valve (F003) closed, an RHR pump is subsequently started with flow back to the RPV regulated by return valve F017. RHR flow is established through the Heat Exchanger by throttling open outlet valve F003. The cooldown rate of reactor coolant is controlled via valves F017 (total RHR system flow) and the F048 (heat exchanger bypass flow), and/or the F003 (heat exchanger flow) and the F047 (heat exchanger inlet). All operations are performed from the control room except for opening and closing of local flush water valves.

The manual actions required for the most limiting failure are discussed in subsection 5.4.7.1.5.

b. Steam Condensing

This section has been intentionally deleted.

c. Fuel Pool Cooling Mode

Operation of RHR in the fuel pool cooling mode requires manual actions to be performed both in the control room and locally. The system will also be required to be filled and

vented, which will require the manipulation of various small manual valves. The filling operation may also include operation of the ESW system in the event the normal fill systems are unavailable. These actions are described in and controlled by plant procedures.

5.4.7.3 Performance Evaluation

Thermal performance of the RHR heat exchangers is based on the residual heat generated after rod insertion, a 125°F vessel outlet (exchanger inlet) temperature, and the flow of two loops in operation. Because shutdown is usually a controlled operation, maximum service water temperature less 10°F is used as the service water inlet temperature. These are nominal design conditions; if the service water temperature is higher, the exchanger capabilities are reduced and the shutdown time may be longer and vice versa.

5.4.7.3.1 Shutdown With All Components Available

No typical curve is included here to show vessel cooldown temperatures versus time due to the infinite variety of such curves that may be due to: (1) clean steam systems that may allow the main condenser to be used as the heat sink when nuclear steam pressure is insufficient to maintain steam air ejector performance; (2) the condition of fouling of the exchangers; (3) operator use of one or two cooling loops; (4) coolant water temperature; and (5) system flushing time. Since the exchangers are designed for the fouled condition with relatively high service water temperature, the units have excess capability to cool when first cut in at high vessel temperatures. Total flow and mix temperature must be controlled to avoid exceeding 100°F per hour cooldown rate.

5.4.7.3.2 Shutdown With Most Limiting Failure

Shutdown under conditions of the most limiting failure is a loss of the suction path for normal shutdown cooling. Reactor shutdown can be achieved using alternate shutdown cooling as described in Section 15.2.9. The capability of the heat exchanger for any time period is balanced against residual heat, pump heat, and sensible heat. The excess over residual heat and pump heat is used to reduce the sensible heat.

5.4.7.4 Preoperational Testing

The preoperational test program and startup test program as discussed in Chapter 14. will verify the capabilities of the system to provide the flows, pressures, condensing rates, cooldown rates, and reaction times required to perform all system functions as specified for the system or component in the System Data Sheets and Process Data.

5.4.8 REACTOR WATER CLEANUP SYSTEM

The reactor water cleanup system is an auxiliary system, a small part of which is part of the reactor coolant pressure boundary up to and including the outermost containment isolation

valve. The other portions of the system are not part of the reactor coolant pressure boundary and are isolatable from the reactor.

5.4.8.1 Design Bases

5.4.8.1.1 Safety Design Bases

The RCPB portion of the RWCU system:

(1) Prevents excessive loss of reactor coolant, and

(2) Prevents the release of radioactive material from the reactor.

5.4.8.1.2 Power Generation Design Bases

The reactor water cleanup system:

(1) Removes solid and dissolved impurities from reactor coolant;

(2) Discharges excess reactor water during startup, shutdown, and hot standby conditions;

(3) Minimizes temperature gradients in the recirculation piping and vessel during periods when the main recirculation pumps are unavailable.

(4) Minimizes RWCU System heat loss; and

(5) Enables the major portion of the RWCU system to be serviced during reactor operation.

5.4.8.2 System Description

The reactor water cleanup system (see Dwgs. M-144, Sh. 1, M-144, Sh. 2, M-144, Sh. 3, Dwg. M1-G33-16, Sh. 1 and Dwg. M1-G33-18, Sh. 1 continuously purifies the reactor water. The system takes its suction from the inlet of the reactor recirculation pump and from the reactor pressure vessel bottom head. The processed water is returned to the reactor pressure vessel via the feedwater system, to the main condenser or radwaste.

The cleanup system can be operated at any time during planned operations, or it may be shut down. The cleanup system is classified as a primary Power Generation System. The cleanup system is not an Engineered Safety Feature.

The major equipment of the reactor water cleanup system is located in the reactor building. This equipment includes regenerative and nonregenerative heat exchangers, filter-demineralizers with regeneration equipment and cleanup pumps. The entire system is connected by associated valves and piping; controls and instrumentation provide proper system operation. Design data for the major pieces of equipment are presented in Table 5.4-2.

Reactor water is cooled in the regenerative and nonregenerative heat exchangers, then filtered, demineralized, and returned to the reactor pressure vessel through the shell side of the regenerative heat exchanger.

The temperature of the filter-demineralizer units is limited by the resin operating temperature. Therefore the reactor coolant must be cooled before being processed in the filter-demineralizer units. The regenerative heat exchanger transfers heat from the tubeside (hot process) to the shellside (cold process). The shellside flow returns to the reactor. The nonregenerative heat exchanger cools the process further by transferring heat to the Reactor Building Closed Cooling Water System. The nonregenerative heat exchanger is sized to maintain the required filter-demineralizer temperature, even when the effectiveness of the regenerative heat exchanger is partially reduced by diversion of a portion of the process to either the main condenser or the radwaste system.

The filter-demineralizer units (see Dwg. M-145, Sh. 1 are pressure precoat type filters using filter aid and finely ground, mixed ion-exchange medium. Spent resins are not regenerable and are sluiced from the filter-demineralizer unit to a backwash receiving tank from which they are transferred to the radwaste system for processing and disposal. To prevent resins from entering the reactor recirculation system in the event of failure of a filter-demineralizer resin support, a strainer is installed on the outlet of each filter-demineralizer unit. Each strainer has a control room alarm that is energized by high differential pressure. A bypass line is provided around the filter-demineralizer units for bypassing the units.

The cleanup recirc pumps are vertical sealless units, top-hung, with the motor below the pump. Two (2) pumps are provided, each with a capacity of 100% of system design flow. To prevent contaminants in the reactor water from reaching the cleanup recirc pump motors, a purge water subsystem (see Dwg. M-144, Sh. 3 is provided. Clean water of reactor quality is taken from the control rod drive hydraulic system and injected into the cleanup recirc pump motors. Monitoring instrumentation for the cleanup recirc pumps, as well as instrumentation and controls for the purge water subsystem are located on a local control panel.

In the event of low flow or loss of flow in the system, or outboard isolation valve G33-F004 is not fully open, when filter demineralizer is available, the RWCU holding pumps are automatically started and flow is maintained through each filter-demineralizer by its own holding pump. Sample points are provided in the common influent header and in each effluent line of the filter-demineralizer units for continuous indication and recording of system conductivity. High conductivity is annunciated in the control room. The influent sample point is also used as the normal source of reactor coolant grab samples. Sample analysis also indicates the effectiveness of the filter-demineralizer units.

The suction line of the RCPB portion of the RWCU system contains two motor-operated isolation valves which automatically close in response to high flow signals. (Sections 7.6 and 5.2 describe the Leak Detection System and it is summarized in Table 5.2-8.) This action prevents the loss of reactor coolant and release of radioactive material from the reactor.

The outboard isolation valve will automatically close to prevent damage of the filter-demineralizer resins if the outlet temperature of the non-regenerative heat exchanger is high. The outboard isolation valve will also close upon manual initiation of the Standby Liquid Control System. This prevents removal of liquid poison by the cleanup system should the SLCS be in operation. The RCPB isolation valves may be remote manually operated to isolate the system equipment for maintenance or servicing.

The outboard isolation valve of Unit 1 and the inboard isolation valve of Unit 2 will close upon placing the transfer switches in 'Emergency' position. The transfer switches are located on the respective Units Remote Shutdown Panels. This action also isolates the control and indication circuits located in the Control Room Panels for these valves.

Two remote manual-operated gate valves on the return lines to the reactor provide long term leakage control. Instantaneous reverse flow isolation is provided by check valves in the RWCU piping.

Operation of the reactor water cleanup system is controlled from the main control room. Resin-changing operations, which include backwashing and precoating, are controlled from a local control panel in the reactor building. The time required to remove a unit from the line, backwash and precoat is typically less than one hour.

A functional control drawing is provided in Section 7.7.

5.4.8.3 System Evaluation

The RCPB isolation valves and piping are designed to the requirements defined in Section 3.2 and the requirements of Subsection 7.3.1.1a.2.

5.4.9 MAIN STEAM LINES AND FEEDWATER PIPING

5.4.9.1 Safety Design Bases

In order to satisfy the safety design bases, the main steam and feedwater lines have been designed:

(1) To accommodate operational stresses, such as internal pressures and safe shutdown earthquake loads, without a failure that could lead to the release of radioactivity in excess of the guideline values in published regulations.

(2) With suitable accesses to permit in-service testing and inspections.

5.4.9.2 Power Generation Design Bases

In order to satisfy the design bases:

(1) The main steamlines have been designed to conduct steam from the reactor vessel over the full range of reactor power operation.

(2) The feedwater lines have been designed to conduct water to the reactor vessel over the full range of reactor power operation.

5.4.9.3 Description

The main steam piping is described in Section 10.3. The main steam and feedwater piping are shown in Dwgs. M-141, Sh. 1 and M-141, Sh. 2.

The feedwater piping consists of two 24-in. outside diameter lines each of which penetrate the containment and drywell and branch into three 12 inch lines which connect to the reactor vessel. Feedwater containment isolation valves are described in Table 6.2-12. The design pressure and temperature of the feedwater piping between the reactor and maintenance valve F011 are 1250 psig and 575°F. The Seismic Category I design requirements are placed on the feedwater piping from the reactor through the inboard isolation valve up to and including the outboard isolation valve in the connected piping.

The materials used in the piping are in accordance with the applicable design code and supplementary requirements described in Section 3.2.

The general requirements of the feedwater system are described in Subsections 7.1.2b.1, 7.7.1, 7.7.2 and 10.4.7.

5.4.9.4 Safety Evaluation

Differential pressure on reactor internals under the assumed accident condition of a ruptured steamline is limited by the use of flow restrictors and by the use of four main steamlines. All main steam and feedwater piping is designed in accordance with the requirements defined in Section 3.2. Design of the piping in accordance with these requirements ensures meeting the safety design bases.

5.4.9.5 Inspection and Testing

Inspection and testing are carried out in accordance with Subsection 3.9.1 and Chapter 14. In-service inspection is considered in the design of the main steam and feedwater piping. This consideration assures adequate working space and access for the inspection of selected components.

5.4.10 PRESSURIZER

Not applicable to BWR.

5.4.11 PRESSURIZER RELIEF DISCHARGE SYSTEM

Not applicable to BWR.

5.4.12 VALVES

5.4.12.1 Safety Design Bases

Valves are designed to operate under the internal pressure/temperature loading as well as the external loading experienced during the various system transient operating conditions.

The design loading combinations, design criteria and stress limits associated with ASME Section III valves are presented in Tables 3.9-6, 3.9-12 and 3.9-25.

The functional testing of active valves is covered in Subsection 3.9.3.2.b.2.

The materials for valves are covered in Subsections 5.2.3 and 6.1.1.

Inservice inspection requirements for ASME Section III valves are covered in Subsection 3.9.6.

5.4.12.2 Description

Line valves furnished are manufactured standard types, designed and constructed in accordance with the requirements of ASME Section III. All materials, exclusive of seals and packing, will endure the 40-year plant life under the environmental conditions applicable to the particular system (see Section 3.11). For service beyond 40 years, aging of equipment is managed to ensure it continues to perform its intended function. Power operators are sized to operate successfully under design-basis conditions. Furthermore, active safety related MOVs are sized to Generic Letter 89-10 requirements committed to the NRC by SSES in PLA-3311.

Motor operated valves are essentially the same type of equipment as used on currently operating plants.

5.4.12.3 Safety Evaluation

Line valves are shop tested by the manufacturer for operability. Pressure retaining parts are subject to the testing and examination requirements of Section III of the ASME Code. To minimize internal and external leakage past seating surfaces, maximum allowable leakage rates are stated in the design specifications for both back seat as well as the main seat for gate and globe valves. Valve construction materials are compatible with the maximum anticipated radiation dosage, which is listed in Section 3.11, for the service life of the valves.

Implementation of operational requirements specified in equipment specifications is ensured by document verification procedures.

The Quality Assurance programs instituted to ensure that valves meet specifications are the same as those described in Subsection 5.2.2.2.5.

Programs to establish qualification of the motor operated valve for the intended service are performed in the following areas:

Stroking time

Opening/closing at maximum differential pressure

Seismic performance

Emergency environmental condition.

In addition, active safety related MOVs meet the GL 89-10 program requirements committed to in PLA-3311.

5.4.12.4 Inspection and Testing

Valves serving as containment isolation valves are operationally tested as discussed in Subsection 6.2.4. Preoperational testing of valves is discussed in Chapter 14. In service testing of valves is discussed in Subsection 3.9.6.

Control valves 4 inches and larger and block valves 2-1/2 inches and larger were originally provided with double-packed stuffing boxes with an intermediate lantern leakoff connection to enable detection and measurement of leakage rates for valves located outside of the primary containment. Valves in the turbine building were originally provided with valve stem packing leakoff connections. Research and testing has shown that improved packing provides an effective seal to prevent leakage into the Turbine Building. As a result, these leakoff connections are in the process of being removed and packing configurations changed, as appropriate, to conform with the new requirements. As part of this effort, leakoff isolation valves and piping will be removed (or abandoned in place) and the leakoff collection header piping will be removed or abandoned in place.

Prior to installation, each motor actuator was assembled, factory tested and adjusted on the valve for proper operation, position, and torque switch setting, position transmitter function (where applicable) and speed requirements. Installed active safety related MOVs meet the GL 89-10 recommendations through current in-house programs as committed to by SSES Letter PLA-3311. These programs for the testing, inspection, and maintenance of MOVs assure that they will function when subjected to the design-basis conditions that are considered during both normal operation and abnormal events within the design basis of the plant.

5.4.13 SAFETY AND RELIEF VALVES

5.4.13.1 Safety Design Bases

Overpressure protection has been provided at isolatable portions of systems in accordance with the rules set forth in the ASME Code, Section III for Class 1, 2 and 3 components.

5.4.13.2 Description

Pressure relief valves have been designed and constructed in accordance with the same code class as that of the line valves in the system.

Table 3.9-1 lists the applicable code classes for valves and system design pressures and temperatures. The design criteria, design loading and design procedure are described in Subsection 3.9.3.

5.4.13.3 Safety Evaluation

The use of pressure relieving devices will assure that overpressure will not exceed 10% above the design pressure of the system. The number of relieving devices on a system or portion of a system have been determined on an individual component basis.

5.4.13.4 Inspection and Testing

Valves are stamped with factory ring (NR, GR) settings and applicable service back pressure range. Other design information is included on the nameplates attached on the valves. Further examinations would necessitate removal of the component. Refer to Subsection 5.2.4 for discussion of Inservice Inspection.

5.4.14 COMPONENT SUPPORTS

Support elements are provided for those components included in the RCPB and the connected systems.

5.4.14.1 Safety Design Bases

Design loading combinations, design procedures, and acceptability criteria are described in Subsection 3.9.3. Flexibility calculations and seismic analysis for piping components conform with the appropriate requirements of ASME Section III.

Spacing and size of pipe support elements were based on the piping analysis performed in accordance with ASME Section III and further described in Section 3.7.

Materials, fabrication, and inspection of pipe supporting elements for nuclear piping are in accordance with the ASME Boiler and Pressure Vessel Code, Section III. See Subsection 3.2 for applicable Code Edition.

5.4.14.2 Description

The use and location of rigid-type supports, variable or constant spring-type supports, and anchors or guides were determined by flexibility, stress, and seismic analysis.

5.4.14.3 Safety Evaluation

Design loadings used for flexibility and seismic analysis toward the determination of adequate component support systems included all transient loading conditions expected by each component. Provisions were made to provide travel stops for spring-type supports for the initial deadweight loading due to hydrostatic testing of steam systems to prevent damage to this type of support.

5.4.14.4 Inspection and Testing

After completion of the installation of a support system, all hanger elements are examined in accordance with the requirements of Chapter 14. Final adjustment capability is provided on all hanger or support types.

5.4.15 HIGH PRESSURE COOLANT INJECTION (HPCI) SYSTEM

Refer to Section 6.3 for discussion.

5.4.16 CORE SPRAY (CS) SYSTEM

Refer to Section 6.3 for discussion.

5.4.17 STANDBY LIQUID CONTROL (SLC) SYSTEM

Refer to Subsection 9.3.5 for discussion.

5.4.18 REFERENCES

5.4-1 Ianni, P.W., "Effectiveness of Core Standby Cooling Systems for General Electric Boiling Water Reactors," APED-5458, March, 1968.

5.4-2 "Design and Performance of General Electric Boiling Water Reactor Main Steam Line Isolation Valves," APED-5750, General Electric Co., Atomic Power Equipment Department, March, 1969.

5.4-3 Reference Deleted

5.4-4 NEDO-32686, “Utility Resolution Guide for ECCS Suction Strainer Blockage”, GE Nuclear Energy, October, 1998.

TABLE 5.4-2

REACTOR WATER CLEANUP SYSTEM EQUIPMENT DESIGN DATA

System Flow Rate (lbs/hr)	146,300	
MAIN CLEANUP RECIRCULATION PUMPS		
Number Required -	2	
Capacity - % (each)	100%	
Design Temperature - (°F)	575	
Design Pressure - (psig)	1,450	
Discharge Head at Shutoff - (ft)	705	
Minimum Available NPSH - (ft)	13	
HEAT EXCHANGERS		
	Regenerative	**Non-Regenerative**
Rated Capacity - (%)	100	100
Shell Side Pressure - (psig)	1,425	150
Shell Side Temperature - (°F)	575	370
Tube Side Pressure - (psig)	1,425	1,425
Tube Side Temperature - (°F)	575	575
FILTER-DEMINERALIZERS		
Number Required -	2	
Capacity - % (each)	50	
Flow/Unit - (lb/hr)	73,150	
Design Temperature - (°F)	150	
Design Pressure - (psig)	1,400	

TABLE 5.4-3

RHR RELIEF VALVE DATA

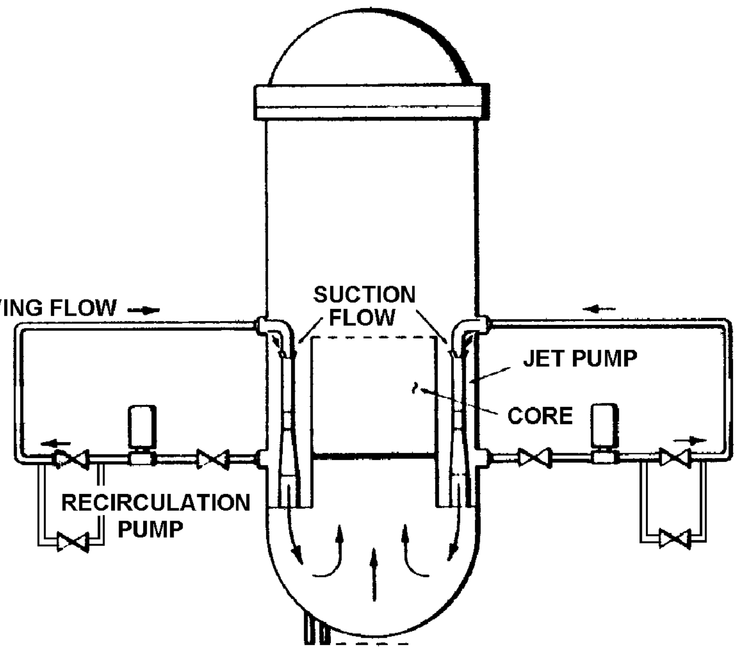
Valve Location	Valve No.	Set Point PSIG	Capacity [1] (gpm)	Method of Collection
Pump Suction Line	PSV-1F030 A, B, C & D	165	72	Liquid Radwaste
Pump Discharge Line	PSV-1F025 A&B	450	20.8	Liquid Radwaste
Head Spray Line	PSV-15113	450	38.1	Liquid Radwaste
Shutdown Supply Line (inside containment)	PSV-1F126	1250	34.6	Drywell Sump
Shutdown Supply Line (outside containment)	PSV-1F029	150	12	Liquid Radwaste
Heat Exchanger (shell side)	PSV-15106 A&B	450	20	Suppression Pool
Heat Exchanger (tube side)	PSV-11213 A&B	180	20	Liquid Radwaste
Heat Exchanger S.W. Line	PSV-11212 A&B	200	1620 gpm	Liquid Radwaste
Loop A B Cross Connect Line	PSV-15193	450	20.8	Liquid Radwaste

[1] Capacity is based on setpoint plus 10% accumulation

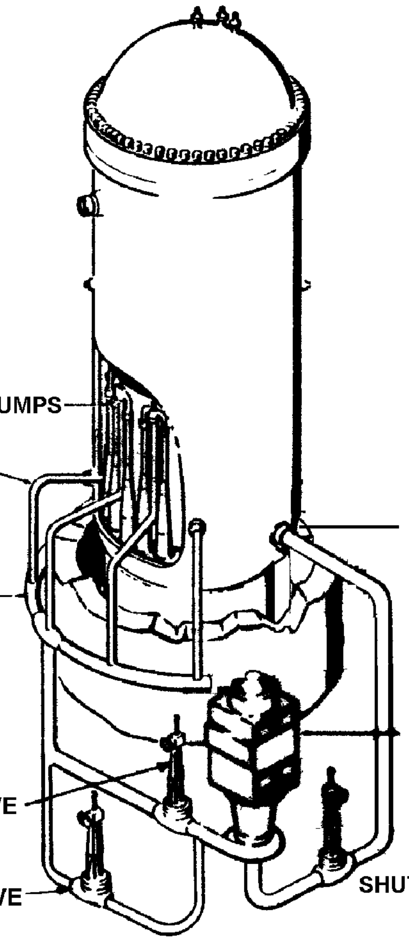
Table 5.4-4

RCPB COMPONENT DESCRIPTION

Security-Related Information
Table Withheld Under 10 CFR 2.390



ELEVATION



ISOMETRIC

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SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
RECIRCULATION SYSTEM ELEVATION AND ISOMETRIC FIGURE 5.4-1, Rev.55

AutoCAD: Figure Fsar 5_4_1.dwg

THIS FIGURE HAS BEEN
REPLACED BY DWG.
M-143, Sh. 1

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SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
Figure 5.4-2B-1 replaced by dwg. M-143, Sh. 1 FIGURE 5.4-2B-1, Rev. 57

AutoCAD Figure 5_4_2B_1.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
M-143, Sh. 2

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
Figure 5.4-2B-2 replaced by dwg. M-143, Sh. 2 FIGURE 5.4-2B-2, Rev. 55

AutoCAD Figure 5_4_2B_2.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
M1-B31-13, Sh. 3

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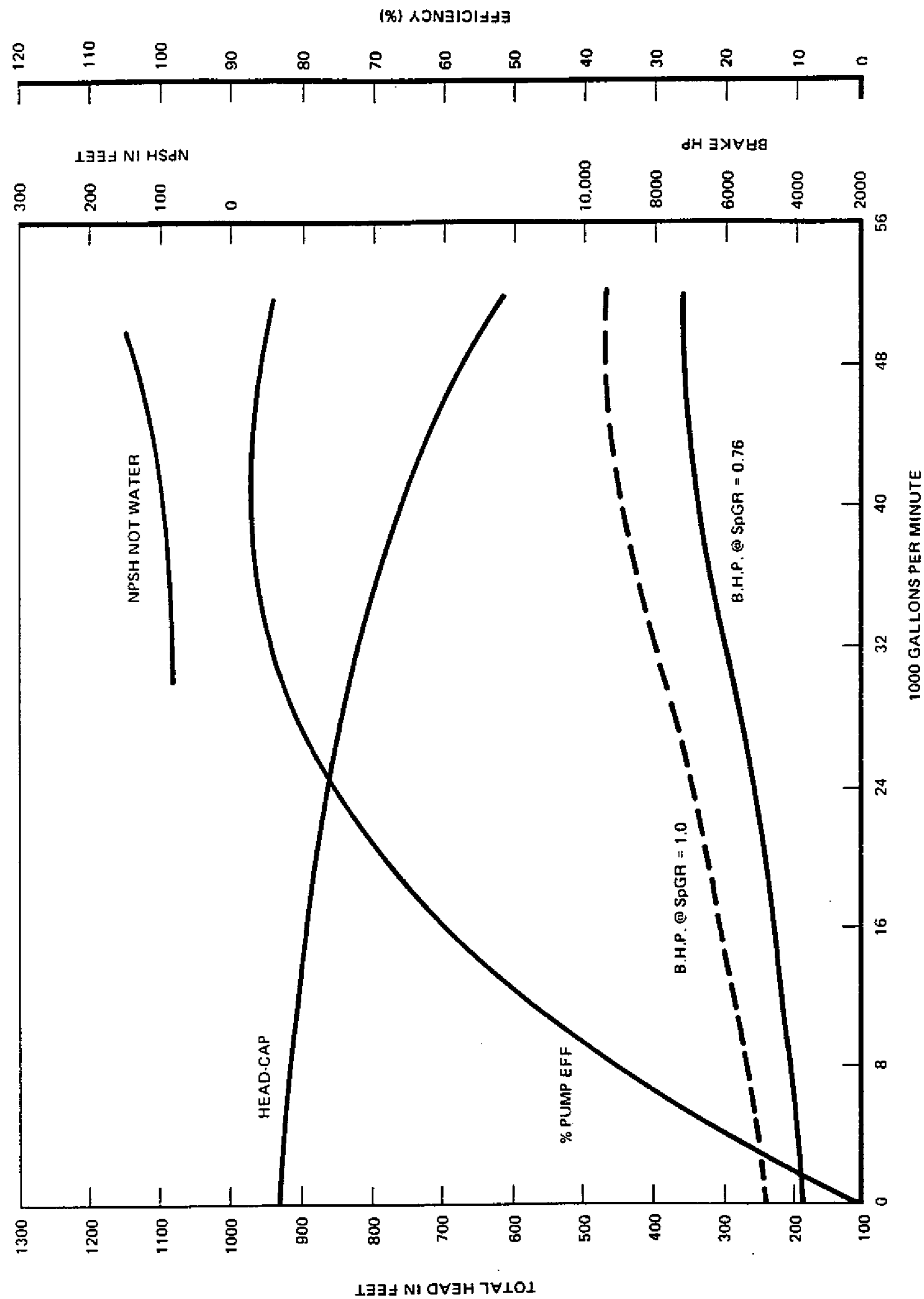
Figure 5.4-2C replaced by dwg.
M1-B31-13, Sh. 3

FIGURE 5.4-2C, Rev. 50

AutoCAD Figure 5_4_2C.doc



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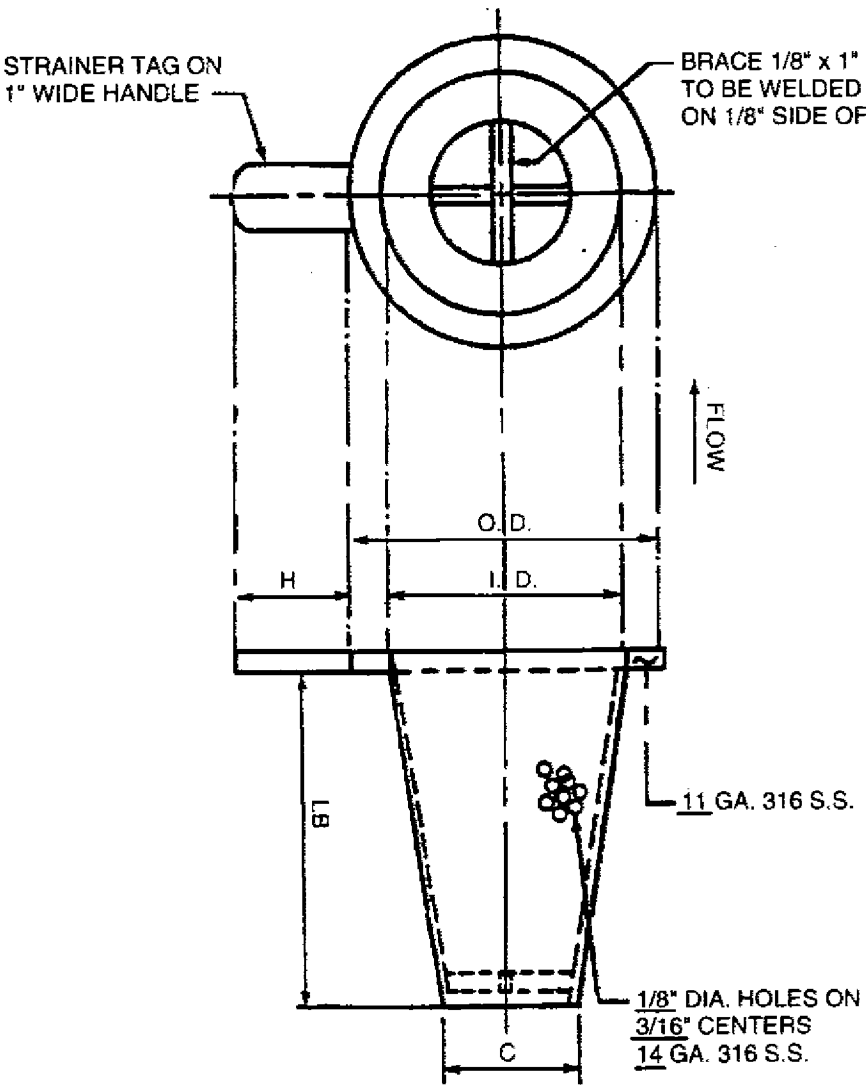
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

RECIRCULATION PUMP HEAD,
NPSH, FLOW AND
EFFICIENCY CURVES

FIGURE 5.4-3, Rev.49

AutoCAD: Figure Fsar 5_4_3.dwg



NOTES:

1) MATERIAL OF CONSTRUCTION ALL STAINLESS STEEL. SA-240-316
2) HANDLE MATERIAL TO BE 11 GA. SA-240-316
3) BOTTOM TO BE PERFORATED

TYPE 5IP BASKET STRAINER							
QUAN	SIZE	RATING	ID	OD	LB	H	C
16	24"	150#RF	22%	28 1/8	48 1/2	4	16

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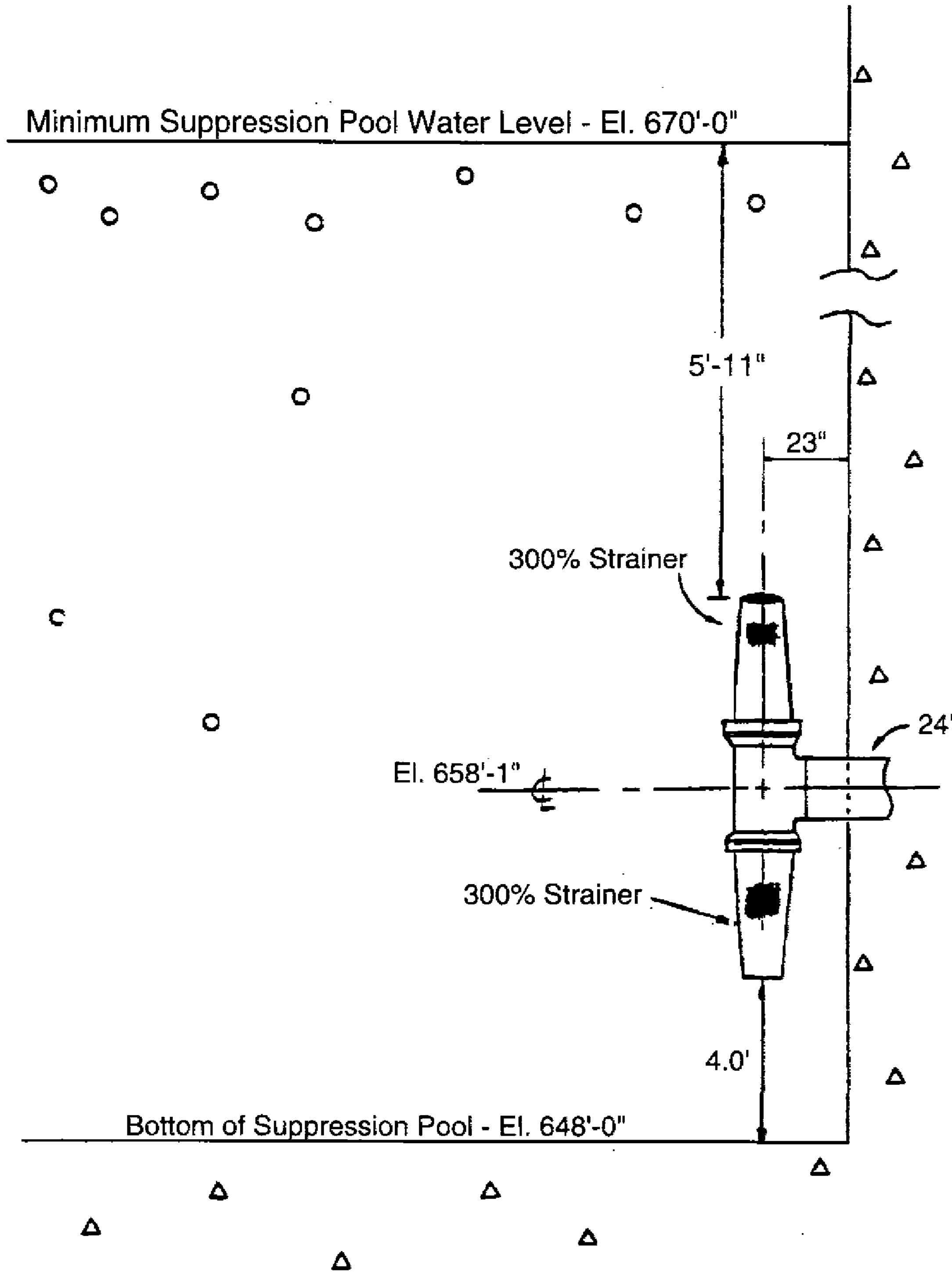
RHR SUCTION STRAINER DETAILS

FIGURE 5.4-4A, Rev.50

AutoCAD: Figure Fsar 5_4_4A.dwg



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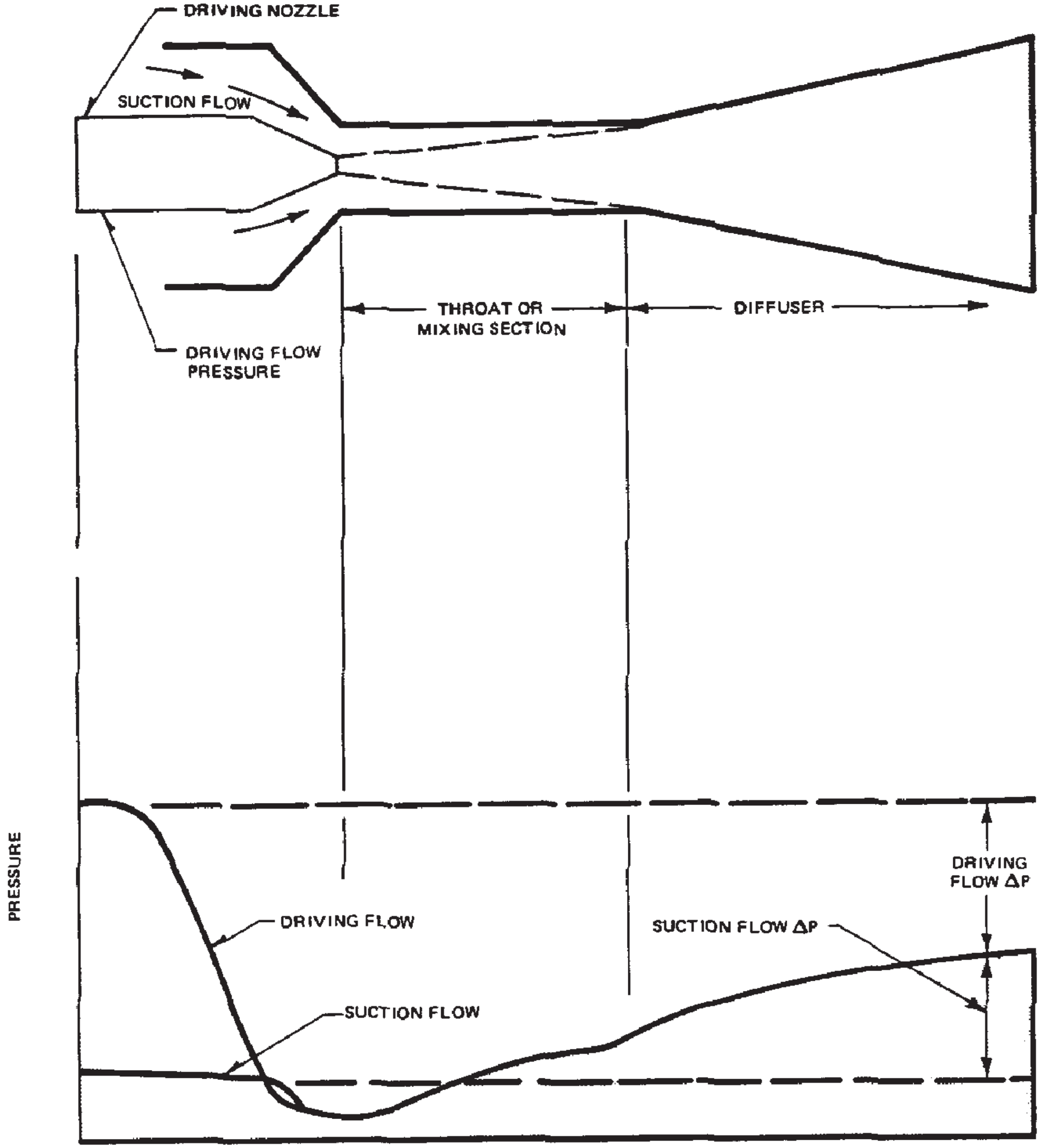
FINAL SAFETY ANALYSIS REPORT

RHR SUCTION STRAINER DETAILS

FIGURE 5.4-4B, Rev.50

AutoCAD: Figure Fsar 5_4_4B.dwg



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SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

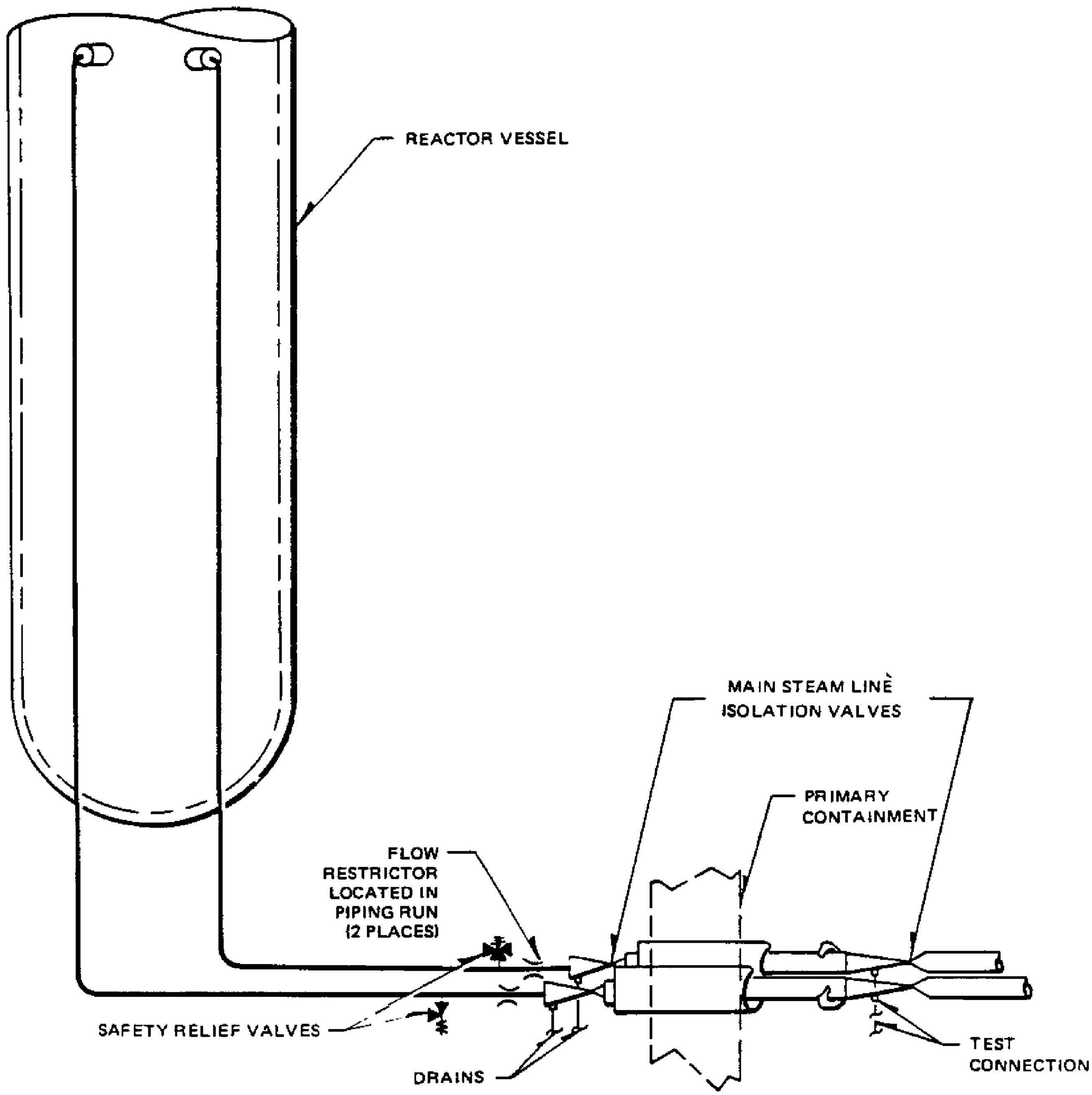
OPERATING PRINCIPLE
OF JET PUMP

FIGURE 5.4-5, Rev.49

AutoCAD: Figure Fsar 5_4_5.dwg

Security-Related Information
Figure Withheld Under 10 CFR 2.390

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
CORE FLOODING CAPABILITY OF RECIRCULATION SYSTEM FIGURE 5.4-6



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SUSQUEHANNA STEAM ELECTRIC STATION
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FINAL SAFETY ANALYSIS REPORT

MAIN STEAM LINE FLOW
RESTRICTOR LOCATION

FIGURE 5.4-7, Rev.49

AutoCAD: Figure Fsar 5_4_7.dwg

THIS FIGURE HAS BEEN
REPLACED BY DWG.
M-149, Sh. 1

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SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
Figure 5.4-9A replaced by dwg. M-149, Sh. 1 FIGURE 5.4-9A, Rev. 55

AutoCAD Figure 5_4_9A.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
M-150, Sh. 1

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SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
Figure 5.4-9B replaced by dwg. M-150, Sh. 1 FIGURE 5.4-9B, Rev. 55

AutoCAD Figure 5_4_9B.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
M1-E51-81, Sh. 1

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SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
Figure 5.4-10 replaced by dwg. M1-E51-81, Sh. 1 FIGURE 5.4-10, Rev. 52

AutoCAD Figure 5_4_10.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
M-151, Sh. 1

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SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
Figure 5.4-13-1 replaced by dwg. M-151, Sh. 1 FIGURE 5.4-13-1, Rev. 53

AutoCAD Figure 5_4_13_1.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
M-151, Sh. 2

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SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
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Figure 5.4-13-2 replaced by dwg.
M-151, Sh. 2

FIGURE 5.4-13-2, Rev. 54

AutoCAD Figure 5_4_13_2.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
M-151, Sh. 3

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SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
Figure 5.4-13-3 replaced by dwg. M-151, Sh. 3 FIGURE 5.4-13-3, Rev. 53

AutoCAD Figure 5_4_13_3.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
M-151, Sh. 4

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SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
Figure 5.4-13-4 replaced by dwg. M-151, Sh. 4 FIGURE 5.4-13-4, Rev. 54

AutoCAD Figure 5_4_13_4.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
M1-E11-3, Sh. 1

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Figure 5.4-14-1 replaced by dwg.
M1-E11-3, Sh. 1

FIGURE 5.4-14-1, Rev. 52

THIS FIGURE HAS BEEN
REPLACED BY DWG.
M1-E11-3, Sh. 2

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Figure 5.4-14-2 replaced by dwg.
M1-E11-3, Sh. 2

FIGURE 5.4-14-2, Rev. 52

AutoCAD Figure 5_4_14_2.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
M1-G33-16, Sh. 1

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Figure 5.4-15A replaced by dwg.
M1-G33-16, Sh. 1

FIGURE 5.4-15A, Rev. 52

AutoCAD Figure 5_4_15A.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
M1-G33-18, Sh. 1

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UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

Figure 5.4-15B replaced by dwg.
M1-G33-18, Sh. 1

FIGURE 5.4-15B, Rev. 52

AutoCAD Figure 5_4_15B.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
M-144, Sh. 1

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SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

Figure 5.4-16-1 replaced by dwg.
M-144, Sh. 1

FIGURE 5.4-16-1, Rev. 57

AutoCAD Figure 5_4_16_1.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
M-144, Sh. 2

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SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
Figure 5.4-16-2 replaced by dwg. M-144, Sh. 2 FIGURE 5.4-16-2, Rev. 55

AutoCAD Figure 5_4_16_2.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
M-144, Sh. 3

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SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
Figure 5.4-16-3 replaced by dwg. M-144, Sh. 3 FIGURE 5.4-16-3, Rev. 56

AutoCAD Figure 5_4_16_3.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
M-145, Sh. 1

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SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
Figure 5.4-18 replaced by dwg. M-145, Sh. 1 FIGURE 5.4-18, Rev. 55

AutoCAD Figure 5_4_18.doc