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5.1 SUMMARY DESCRIPTION

The RCS includes those systems and components that contain or transport fluids to or from the reactor core. These systems form a major portion of the nuclear system process barrier. This chapter provides information regarding the RCS and pressure-containing appendages out to and including isolation valves. This group of components is defined as the RCPB in 10CFR50.2(v) as follows:

"Reactor coolant pressure boundary means all those pressure-containing components of boiling and pressurized water-cooled nuclear power reactors, such as pressure vessels, piping, pumps, and valves, which are:

a. Part of the reactor coolant system, or

b. Connected to the reactor coolant system, up to and including any and all of the following:

    1. The outermost containment isolation valve in system piping which penetrates primary reactor containment

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

    3. The reactor coolant system safety and relief valves."

Section 5.4 of this chapter also deals with various subsystems to the RCPB that are closely allied to it. These are briefly reviewed below.

The nuclear pressure relief system protects the RCPB from damage due to overpressure. To protect against overpressure, pressure-operated relief valves are provided to discharge steam from the NSSS to the suppression pool. The nuclear pressure relief system also acts to automatically depressurize the NSSS if there is a LOCA in which the HPCI system fails to maintain RPV water level. Depressurization of the NSSS allows the low pressure core cooling systems to supply enough cooling water to adequately cool the fuel.

The RCPB leak detection system, described in Section 5.2.5, detects system leakage inside the primary containment so that appropriate action can be taken before the integrity of the nuclear system process barrier is impaired.

The RPV and appurtenances are described in Section 5.3. The major safety functions of the RPV are to maintain water over the core and to act as a radioactive material barrier. The RPV meets the requirements of applicable codes and criteria. The possibility of brittle fracture is considered, and suitable design and operational limits are established that avoid conditions in which 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 following plant load demand without adjusting the control rods. The reactor recirculation system is designed to provide a slow coast-down of flow so that fuel thermal limits cannot be exceeded as a result of recirculation system malfunctions. The arrangement of the reactor recirculation system piping is such that a piping failure cannot compromise the integrity of the floodable inner volume of the reactor vessel, thereby helping to ensure adequate core cooling following a LOCA.

The main steam line flow restrictors are venturi-type flow devices. One restrictor is installed in each main steam line inside the primary containment. The restrictors are designed to limit the loss of coolant resulting from a main steam line break outside the primary containment. The coolant loss is limited so that RPV water level remains above the top of the core during the time required for the MSIVs to close. This action maintains the integrity of the fuel cladding (fuel barrier).

The MSIVs automatically close to isolate the nuclear system process barrier if a pipe break occurs downstream of the isolation valves, thereby limiting the loss of coolant and the release of radioactive materials from the NSSS. Two MSIVs are installed on each main steam line, one inside and the other outside the primary containment. Closure of either of the two MSIVs acts to seal the primary containment if a main steam line break occurs inside the primary containment.

The RCIC system provides makeup water to the core during a reactor shutdown when feedwater flow is not available. The system is started either automatically on receipt of a low reactor water level signal or manually by the operator. Water is pumped to the core by a turbine-driven pump using reactor steam.

The RHR system includes a number of pumps and heat exchangers that can be used to cool the NSSS in a variety of situations. During normal shutdown and reactor servicing, the RHR system removes residual and decay heat. The RHR system allows decay heat to be removed whenever the main heat sink (main condenser) is not available (e.g., hot standby). Another operational mode of the RHR system is LPCI. LPCI operation is an ESF system for use during a postulated LOCA. This operation is described in Section 6.3. Another mode of RHR system operation allows heat to be removed from the primary containment following a LOCA.

The RWCU system functions to maintain the required quality of reactor coolant by circulating coolant through a system of filter/ demineralizers.

Design and performance characteristics of the RCS and its various components are shown in Table 5.4-1.

5.1.1 SCHEMATIC FLOW DIAGRAM

A flow diagram and process information for the reactor vessel and recirculation loops under normal steady-state full power operating conditions is presented in Figure 5.1-1. Coolant volumes in the reactor vessel and recirculation loops under full power conditions are presented in Figure 5.1-2.

5.1.2 PIPING AND INSTRUMENTATION DIAGRAMS

Piping and instrumentation diagrams covering the systems included within the RCS and connected systems are presented in the following:

a. Nuclear boiler shown in drawing M-41

b. Nuclear boiler vessel instrumentation shown in drawing M-42

c. Main steam shown in drawing M-01

d. Feedwater system shown in drawing M-06

e. Recirculation system shown in drawing M-43

f. RCIC system shown in drawing M-42 and M-50

g. RHR system shown in drawing M-51

h. RWCU system shown in drawing M-44.

5.1.3 ARRANGEMENT DRAWINGS

A section drawing of the reactor enclosure and primary containment is shown in drawings M-123 and M-138. The routing of reactor recirculation, feedwater, main steam, and RHR piping inside the primary containment is shown in Figure 5.1-5. Piping layout drawings for the reactor recirculation piping and main steam piping inside primary containment are presented in Figures 5.1-6 and 5.1-7, respectively.

5.2 INTEGRITY OF REACTOR COOLANT PRESSURE BOUNDARY

This section discusses measures employed to provide and maintain the integrity of the RCPB for the plant design lifetime.

5.2.1 COMPLIANCE WITH CODES AND CODE CASES

5.2.1.1 Compliance with 10CFR50.55a

A table that shows compliance with the rules of 10CFR50, "Codes and Standards", is included in Section 3.2. As stated in Table 3.2-1, note 7, alternative codes to those required by 10CFR50.55a were used for primary pressure boundary components.

5.2.1.2 Applicable Code Cases

The RPV 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. 10CFR50.55a requires code case approval only for Class 1 components. These code cases contain requirements or special rules that 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. The listed code cases are either in accordance with the recommendations of Regulatory Guide 1.84 and Regulatory Guide 1.85 or special written acceptance has been obtained for their use. Conformance to Regulatory Guide 1.84 and Regulatory Guide 1.85 for other than Class 1 components is discussed in Section 1.8.

5.2.2 OVERPRESSURE PROTECTION

Overpressure protection for the RCPB is provided by the nuclear pressure relief system. The nuclear pressure relief system includes 14 MSRVs, which are dual-function safety/relief valves.

5.2.2.1 Design Basis

Overpressure protection is provided in conformance with GDC 15. Preoperational and startup instructions are given in Chapter 14.

5.2.2.1.1 Safety Design Bases

The nuclear pressure relief system is designed to perform the following functions:

a. Prevent overpressurization of the nuclear system that could lead to the failure of the RCPB

b. Provide automatic depressurization for small breaks in the nuclear system occurring with misoperation of the HPCI system so that the LPCI and the core spray systems can operate to protect the fuel barrier

c. Permit verification of its operability

d. Withstand adverse combinations of loadings and forces resulting from normal, upset, emergency, or faulted conditions

5.2.2.1.2 Power Generation Design Bases

The nuclear pressure relief system MSRVs are designed to meet the following power generation bases:

a. Discharge to the containment suppression pool

b. Correctly reclose following operation so that maximum operational continuity can be obtained

5.2.2.1.3 Discussion

The ASME B&PV Code requires that each vessel designed to meet Section III be protected from overpressure under upset conditions as discussed in subsection S.2.3 of GESTAR II (Reference 4.1-1). The MSRV 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 pressure relief system is evaluated in Sections 6.3 and 7.3.

The following detailed criteria are used in the selection of MSRVs:

a. Meet the requirements of ASME Section III

b. Qualify for 100% of nameplate capacity credit for the overpressure protection function

c. Meet other performance requirements such as response time, etc. as necessary to provide relief functions

The MSRV discharge piping is designed, installed, and tested in accordance with the ASME Section III, Class 3. Fatigue evaluation of the unsubmerged portion of the MSRV discharge piping and downcomers in the wetwell has been performed in accordance with ASME Section III, Class 1 fatigue rules.

5.2.2.1.4 Main Steam Safety/Relief Valve Capacity

The MSRV capacity is adequate to limit the primary system pressure, including transients, to the requirements of the ASME Section III, "Nuclear Vessels", up to and including the Summer 1969 Addenda for LGS. The essential ASME requirements that are all met by this analysis are as follows.

It is recognized that the protection of vessels in a nuclear power plant is dependent on many protective systems to relieve or terminate pressure transients. Installation of pressure-relieving devices may not independently provide complete protection. The MSRV sizing evaluation assumes credit for operation of the RPS, which may be tripped by either of two sources: a direct position switch signal or an indirect high neutron flux trip signal. The direct scram trip signal is derived from position switches mounted on the MSIVs 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 prior to 10% travel of full stroke. The pressure switches are actuated when a fast closure of the turbine control valves is initiated. Credit is taken for the SRVs in their ASME Code qualified self-actuating mode.

The rated capacity of the pressure-relieving devices is 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 Section 15.2.

Full account is taken of the pressure drop on both the inlet and discharge sides of the valves. Each MSRV discharges into the suppression pool through a separate discharge pipe that is designed to achieve sonic flow conditions through the valve, thus providing flow independence to discharge piping losses.

Table 5.2-5 lists the systems that could initiate during the design basis overpressure event.

5.2.2.2 Design Evaluation

5.2.2.2.1 Method of Analysis

The nuclear boiler system pressure protection was designed using an analytical model representing all essential dynamic characteristics of the system. This model 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, pressure, and load demand. 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.

Detailed descriptions of the one-dimensional kinetics model (ODYN) is and the three-dimensional kinetics model (TRACG) are documented in References 5.2-7 and 5.2-34 respectively. MSRVs are simulated in a nonlinear representation, and the models thereby allow full investigation of the various valve response times, valve capacities, and actuation setpoints that are available in applicable hardware systems.

Further descriptions of these models are given in GESTAR II (Reference 4.1-1).

5.2.2.2.2 System Design

An analysis was conducted to demonstrate that the steam flow capacity of the MSRVs is sufficient for power rerate operation.

5.2.2.2.2.1 Operating Conditions

a. Operating power = 3527 MWt

b. Vessel dome pressure P =1068 psia

c. Steam flow = 15.35×10^6 lb/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 MSIVs and a turbine-generator trip with a coincident closure of the turbine steam bypass system valves, that represent the most severe abnormal operational transients 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 indirectly derived scrams; therefore, it is used as the overpressure protection basis event.

The setpoint values used for the overpressure analysis have been reviewed to ensure that both instrument uncertainties and drift allowance are included in the transient simulation of the highest attainable transient pressures. The Technical Specification values for reactor steam dome pressure will be established consistent with the initial dome pressure of 1068 psia used for the analysis. The effect of an ATWS recirculation pump trip on reactor pressure was simulated using an analytical trip setpoint value of 1164 psia. This trip occurs approximately 4.5 seconds after the transient is initiated.

5.2.2.2.2.3 Scram

a. Scram reactivity curve - Figure 5.2-2

b. CRD scram motion Figure 5.2-3

5.2.2.2.2.4 MSRV Transient Analysis Specifications

a. Valve groups: 3

b. Pressure setpoint (maximum safety limit):

group 1 - 1205 psig

group 2 - 1215 psig

group 3 - 1226 psig

The setpoints are assumed at a conservatively high level above the nominal setpoints as shown by Table 5.2-2. This is to account for initial setpoint errors and any instrument setpoint drift that might occur during operation. Typically the assumed setpoints in the analysis are 3% above the actual nominal setpoints. Highly conservative MSRV response characteristics are also assumed.

5.2.2.2.2.5 MSRV Capacity

Sizing of MSRV capacity is based on establishing an adequate margin from the peak vessel pressure to the vessel code limit (1375 psig) in response to the reference transients (Section 5.2.2.2.2.2).

5.2.2.2.3 Evaluation of Results

5.2.2.2.3.1 MSRV Capacity

The design pressure of the reactor vessel and reactor coolant boundary is 1250 psig, and the ASME allowable overpressure limit is 1375 psig. The Main Steam Isolation Valve Closure with Flux Scram (MSIVF) event is the design basis event to demonstrate compliance to the ASME vessel overpressure protection criteria and is analyzed during every cycle-specific reload licensing process. The rerate analysis conservatively assumes that the position scram fails and the event terminates on a high neutron flux scram signal. The closure of all MSIVs causes an abrupt pressure increase in the RPV, which is mitigated by the actuation of the SRVs. The MSIVF event was analyzed at dome pressure of 1068 psia, 2% overpower for initial power, 110% of rated core flow and normal feedwater temperature. The MSIVF event was analyzed with all SRVs in service, 110% of rated core flow and normal feedwater temperature. The MSIVF event was analyzed with three (3) SRVs out of service (OOS) (3% setpoint tolerance), and two (2)SRVs out of service (3% setpoint tolerance).

The calculated reactor dome pressures (psig) for these cases are as follows:

three (3) SRVs OOS	1329
two (2) SRVs OOS	1318

The calculated reactor vessel bottom head pressures (psig) for these cases are as follows:

three (3) SRVs OOS	1348
two (2) SRVs OOS	1338

Since the analysis for three (3) SRVs OOS with 3% SRV setpoint tolerance exceeds the Technical Specification reactor dome pressure safety limit of 1325 psig, the number of SRVs allowed OOS by the Technical Specifications has been changed from three (3) to two (2). The reactor vessel bottom head pressures remain below the ASME overpressure limit of 1375 psig.

Under the "General Requirements for Protection Against Overpressure" as given in ASME Section III, credit can be allowed for a scram from the RPS. In addition, credit is also taken for the protective circuits that are indirectly derived when determining the required MSRV capacity. The backup reactor high neutron flux scram is conservatively applied as a design basis in determining the required capacity of the MSRVs. 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 SRV 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 MSRV 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 the back pressure on each MSRV from exceeding 40% of the valve inlet pressure, thus ensuring choked flow in the valve orifice and no reduction of valve capacity due to the discharge piping. Each MSRV has its own separate discharge line.

5.2.2.3 Piping & Instrument Diagrams

Drawing M-41 and Figure 5.2-6 show the schematic location of pressure-relieving devices for the following:

a. The RCS

b. The primary side of the auxiliary or emergency systems interconnected with the primary system

c. Any blowdown or heat dissipation system connected to the discharge side of the pressure-relieving devices

The schematic arrangement of the MSRVs is shown in Figures 5.2-6 and 5.2-7. Drawings M-41 and M-59 are the P&IDs of the nuclear boiler and PCIG systems.

5.2.2.4 Equipment and Component Description

5.2.2.4.1 Description

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

The MSRVs provide two main protection functions:

a. Overpressure safety/relief operation. The valves open to limit a pressure rise.

b. Depressurization operation. The ADS valves open automatically as part of the ECCS if HPCI fails for events involving small breaks in the nuclear system process barrier (Sections 6.3 & 7.3).

Chapter 15 discusses the events that are expected to activate the MSRVs. Chapter 15 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 MSRV 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 the RHR system can dissipate this heat. 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 MSRV is shown in Figure 5.2-8. It is opened by either of two modes of operation:

a. The safety mode of operation involves extension of the pressure sensing bellows which engages the first stage pilot disc when reactor pressure approaches the valve setpoint. Once the pilot valve starts to open, the upstream seating force is eliminated, resulting in a net increase in the force tending to open the pilot valve. Opening of the first stage pilot valve admits fluid to the operating piston of the second stage valve, causing it also to open. Opening of the second stage pilot valve vents the chamber over the main valve piston to the downstream side of the valve. This venting action creates a differential pressure across the main valve piston almost equal to the system pressure and in a direction tending to open the valve.

b. The relief, or power-actuated mode of operation uses an auxiliary actuating device that directly opens the second stage valve, which vents the chamber over the main valve piston, causing the main valve to open.

The pneumatic operator is so arranged that, if it malfunctions, it will not prevent the pilot disc from lifting if steam inlet pressure reaches the set pressure.

The MSRVs can be operated in the power-actuated mode by remote manual controls from the control room.

The MSRVs are designed to operate to the extent required for overpressure protection in the following accident environments:

a. 340°F for 3 hours at drywell pressure $\leq$ 45 psig

b. 320°F for an additional 3 hour period, at drywell pressure $\leq$ 45 psig

c. 250°F for an additional 18 hour period, at 25 psig

d. 200°F at 20 psig for an additional 24 hour period, following which the valves remain fully open or closed for the remainder of a 100 day period, provided that an air and a power supply are available. No power/air supply is required to keep the valve closed.

The ADS uses selected MSRVs for depressurization of the reactor, as described in Sections 6.3 and 7.3. Each of the MSRVs used for automatic depressurization is equipped with an air accumulator and check valve arrangement. Each ADS accumulator is sized to provide two ADS valve actuations at 70% of drywell design pressure because that is the maximum pressure for which rapid reactor depressurization through the ADS valves is required. For large breaks that result in higher drywell pressure, sufficient reactor depressurization occurs due to the break to preclude the need for ADS. One ADS actuation at 70% of drywell design pressure is sufficient to depressurize the reactor and allow inventory makeup by the low pressure ECCS. However, for conservatism, the ADS accumulators are sized to allow two ADS actuations at 70% of drywell design pressure. This design provides sufficient gas to the ADS valves to permit depressurization until the RHR shutdown cooling mode can be initiated (short-term).

An allowable leakage criteria is applied only to the short-term ADS SRV operations because backup nitrogen bottles provide a longer term supply and provision is made for an infinite long-term supply through the use of external connections. An allowable leakage criteria of 164 scc/min was established to ensure that there would be sufficient pneumatic pressure to depressurize the RPV from the HPCI/RCIC operating pressure to the RHR shutdown cooling operating pressure range using two ADS SRV actuations over a period of 6 hours. Calculations indicate that this leakage criteria will ensure ADS operability for periods in excess of 6 hours for a range of containment conditions that might accompany the need for ADS operation. This short-term duration is sufficient to ensure that the ADS valves can perform their functions as described in Chapter 15, Accident Analyses.

To account for any possible increase in leakage due to a harsh environment and/or seismic event, a surveillance test leakage limit of 78 scc/min was established to provide additional margin as compared to the allowable leakage criteria of 164 scc/min.

The pneumatic components that make up the ADS accumulator system were designed for low leakage in a harsh environment and seismic event. A safety-grade pneumatic supply of nitrogen bottles is available to provide a backup supply if the ADS accumulator system leakage rate should exceed its allowable limit. These bottles are part of the long-term ADS gas supply subsystem of the Containment Instrument Gas System described in Section 9.3.1.3.2.

It should be understood that “short-term” and “long-term” as used here to describe the gas supplies for the ADS valves are not synonymous with the same terms when used to describe ECCS functions. In the context of ADS valve gas supplies, short-term is the first 6 hours of postulated events requiring power operation of ADS valves. This is the timeframe during which the plant would typically be brought to cold shutdown conditions following postulated events. In the context of ECCS functions, short-term refers to a shorter time period of ECCS operation after postulated reactor coolant pressure boundary breaks. A coolable core geometry is assured for subsequent long term cooling. Thus, the long-term gas supply subsystem is not required to support short-term ADS valve ECCS functions nor is it, typically, required to support the first few hours of ADS valve ECCS long-term cooling functions.

Testing and/or analysis has been performed, as described below, to verify that a harsh environment and/or seismic event would not increase the leakage rate of the ADS pneumatic supply system.

The pneumatic system solenoid valves have been qualified under the qualification program for electrical equipment to remain functional under conditions simulating the environment following a postulated design basis LOCA. The pneumatic system spring-loaded, soft-seated valves have been qualified to remain leak-tight after a seismic event. A materials and design review was performed to ensure that the check valves would not experience a significant increase in leakage due to post-LOCA environmental conditions. The functional capability of the check valves was further reviewed as part of the mechanical equipment qualification program. In any event, the long-term safety-grade pneumatic supply will provide the motive force for ADS valve operation.

Each MSRV discharges steam through a discharge line to a point below the minimum water level in the suppression pool. MSRV discharge line piping from the valve to the suppression pool consists of two parts. The first part is attached at one end to the valve and at its other end to a pipe anchor. The main steam piping, including this portion of the MSRV discharge piping, is analyzed as a complete system.

The second part of the MSRV discharge piping extends from the 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.

The MSRV discharge piping is designed to limit valve outlet pressure to 40% of maximum valve inlet pressure with the valve wide open at steady-state flow. Water in the line more than a few feet above the 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 MSRV discharge line to prevent the drawing of an excessive amount of water up into the line as a result of steam condensation following termination of relief operation. The MSRVs 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 and CS systems to operate as a backup for the HPCI system. Further descriptions of the operation of the automatic depressurization feature are found in Sections 6.3 and 7.3 .

MSRV's A, C and N, controlled from the RSP, are provided with accumulators and check valves, which enhance the power-actuated mode of operation of these MSRV's. The accumulators and check valves are shown in drawing M-41, Sheet 2.

5.2.2.4.2 Design Parameters

The specified operating transients for components within the RCPB are given in Table 5.2-9. See Section 3.7 for a 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 RCS against environmental effects are discussed in Section 3.11.

The complete ADS accumulator system and associated equipment and control circuitry are included in the LGS Environmental Qualification Program for electrical and mechanical equipment and are qualified to accommodate the effects of, and be compatible with, the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents as stated in GDC 2 and 4 of 10CFR50, Appendix A.

5.2.2.4.2.1 Main Steam Safety/Relief Valve

The discharge area of each MSRV is 20.63 in^2, and the coefficient of discharge K_D is equal to 0.8 ($K = 0.9\ K_D$).

The design pressure and temperature of the valve inlet and outlet are 1250 psig @ 575°F and 500 psig @ 470°F, respectively.

The valves are designed to achieve the maximum practicable number of actuations consistent with state-of-the-art technology.

See Figure 5.2-8 for a schematic cross-section of the valve.

5.2.2.5 Mounting of Main Steam Relief Valves

The MSRVs are located on the main steam piping header. The mounting consists of a special contour nozzle and an oversized flange connection. This provides a high integrity connection that withstands the thrust, bending, and torsional loadings to which the main steam pipe and MSRV discharge pipe are subjected. This includes the following:

a. The thermal expansion effects of the connecting piping

b. The dynamic effects of the piping due to an SSE

c. The reactions due to transient unbalanced wave forces exerted on the MSRVs during the first few seconds after the valve is opened and before the time when steady-state flow is established (with steady-state flow, the dynamic flow reaction forces are self-equilibrated by the valve discharge piping)

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

In no case are allowable valve flange loads exceeded, nor does 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 MSRV discharge are in Section 3.9.3.3.

5.2.2.6 Applicable Codes and Classification

The vessel overpressure protection system is designed to satisfy the requirements of ASME Section III. The general requirements for protection against overpressure of ASME Section III recognize that reactor vessel overpressure protection is one function of the reactor protection systems and allows the integration of pressure relief devices with the reactor protection systems of the nuclear reactor.

Hence, credit is taken for the reactor protection system as a complementary pressure protection device. The NRC has also adopted the ASME Codes as part of its requirements in 10CFR50.55a.

5.2.2.7 Material Specification

Material specifications of pressure-retaining components of MSRVs are listed in Table 5.2-3.

5.2.2.8 Process Instrumentation

Overpressure protection process instrumentation is listed in drawing M-42.

ADS valve gas supply instrumentation is described in Sections 7.6 and 9.3.1.3.

5.2.2.9 System Reliability

The system is designed to satisfy the requirements of ASME Section III; therefore, it has high reliability. The consequences of failure are discussed in Section 15.1.4.

5.2.2.10 Inspection and Testing

The MSRVs are tested at the vendor's shop in accordance with quality control procedures to detect defects and prove operability before installation. The following tests are conducted:

a. Hydrostatic test at specified test conditions

b. Pneumatic seat leakage test at 90% of set pressure, with maximum permitted leakage of 30 bubbles per minute being emitted from a 0.250 inch diameter hole submerged ½ inch below a water surface or an equivalent test using an approved test medium

c. Set pressure test: valve pressurized with saturated steam, with the pressure rising to the valve set pressure. Valve must open at nameplate set pressure ±1%.

d. Response time test: each valve is tested to demonstrate acceptable response time

The valves are installed as received from the factory. The GE equipment specification requires certification from the valve manufacturer that design and performance requirements have been met. This includes capacity and blowdown requirements. The setpoints are adjusted, verified, and indicated on the valves by the vendor. The specified manual and automatic actuation relief mode of each MSRV is verified during the preoperational test program.

It is not feasible to test the MSRV setpoints while the valves are in place. The valves are mounted on 1500 pound primary service rating flanges. They can be removed for maintenance or bench checks and reinstalled during normal plant shutdowns. The valves are tested to check set pressure in accordance with the requirements of the Technical Specifications. The external surface and seating of all MSRVs are 100% visually inspected when the valves are removed for maintenance or bench checks. Valve operability is verified during the preoperational test program as discussed in Chapter 14.

A surveillance test of the ADS accumulator system will be conducted every refueling cycle under the LLRT program. Although the accumulator system is not considered to be a part of the primary containment boundary, this test under the LLRT program ensures completion of the surveillance test. Two tests will be performed at 90 psig. For the first test, the isolation boundaries will be the normal supply check valve, the seismic supply check valve, and the de-energized actuation solenoid. Vents are provided on the upstream side of the isolation check valves to meet the single isolation valve criteria. For the second test, the solenoid will be energized so that the SRV actuator and the solenoid valve vent port become part of the boundaries. In both tests, the leakage criteria will be 78 scc/min. as discussed in Section 5.2.2.4.1.

ADS valves gas supply pressure is monitored periodically by Operators in accordance with ECCS Technical Specifications.

5.2.3 Reactor Coolant Pressure Boundary Materials

5.2.3.1 Material Specifications

Table 5.2-3 lists the principal pressure-retaining materials and the appropriate material specifications for the RCPB 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

Materials in the primary system are primarily austenitic stainless steel and Zircaloy cladding. Reactor water chemistry limits are established to provide an environment favorable to these materials. Limits are placed on conductivity, pH, 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 unusual materials in the coolant. Chloride limits are specified to mitigate stress-corrosion cracking of stainless steel. Other controls and limits are implemented which would further reduce stress-cracking or decrease crack growth rates, for example: Hydrogen Water Chemistry: Noble Metals as applicable based on industry experience and optimal water chemistry guidelines for BWRs. For further information, see Reference 5.2-2.

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 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. These measurements were determined in a wetting and drying situation using alkaline-phosphate treated boiler water and, therefore, are of limited significance to BWR conditions. They are, however, a qualitative indication of trends.

When conductivity is in its normal range, pH, chloride, and other impurities affecting conductivity may be assumed to be within their normal range. When conductivity becomes abnormal, chloride measurements are made to determine whether or not they are also out of their normal operating values. This would not necessarily be the case. Conductivity could be high due to the presence of a neutral salt that would not have an effect on pH or chloride. In such a case, high conductivity alone is not a cause for shutdown. In some types of water-cooled reactors, conductivities are high because of the purposeful use of additives. In BWRs, however, where high purity water chemistry at near-neutral pH is maintained, conductivity provides a good and prompt measure of the quality of the reactor water. Significant changes in conductivity provide a warning mechanism so the operator can investigate and remedy the condition before reactor water limits are reached. Methods available to the operator for correcting the off-standard condition include operating the RWCU system, reducing the input of impurities, and placing 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.

a. Normal plant operation

The BWR system water chemistry is conveniently described by following the system cycle as shown in Figure 5.2-9. Reference to Table 5.2-4 is made as numbered on the diagram and correspondingly in the table.

For normal operation starting with the condenser/ hotwell, condensate water is processed through condensate cleanup system, resulting in effluent water quality represented in Table 5.2-4. As the effluent is pumped through the feedwater heater train, zinc and/or iron may be injected into the feedwater stream by the Zinc Injection Passivation (GEZIP) Skid to adjust feedwater chemistry. The additions to feedwater are used to adjust the final water chemistry in the reactor vessel and primary loop.

The GEZIP skid is used to suspend components in demineralized water and to dilute the suspension with additional demineralized water during injection. The addition of soluble zinc oxide is used to reduce the levels of corrosion products and cobalt deposits in the primary piping and components. This reduces the radioactive contamination buildup and dose levels. Iron oxide is added to maintain feedwater iron levels within the industry recommended range of 0.5-1.5 ppb. This reduces the redistribution of radioactive cobalt crud and dose levels. Impurity limits are placed on the compounds added to the GEZIP tank to limit the impact on reactor chemistry to a maximum increase of 0.5 ppb sulfates and 0.5 ppb halogens. The chloride and oxygen levels in the demineralized water and GEZIP tank have a negligible impact on the chloride and oxygen levels in feedwater due to the small quantity injected when compared to the total feedwater flow.

The effluent from the condensate cleanup system is pumped through the feedwater heater train and enters the reactor vessel at an elevated temperature and typically with a chemical composition as shown in Table 5.2-4.

During normal plant operation, boiling occurs in the reactor, decomposition of water takes place due to radiolysis, and oxygen and hydrogen gas are formed. Due to steam generation, stripping of these gases from the water phase takes place, and the gases are carried with the steam through the turbine to the condenser. The oxygen level in the steam, resulting from this stripping process, is typically observed to be about 20 ppm (Table 5.2-4). Deaeration takes place at the condenser, and the gases are removed from the process by SJAEs. The deaeration is completed to a level of approximately 20 ppb (0.02 ppm) oxygen in the condensate. The capability exists for adding oxygen to condensate water to maintain oxygen at levels which protect system piping from corrosion when needed.

The dynamic equilibrium in the reactor vessel water phase, established by the steam-gas stripping and the radiolytic formation (principally) rates, corresponds to a nominal value of approximately 200 ppb (0.2 ppm) of oxygen at rated operating conditions. Slight variations around this value have been observed as a result of differences in neutron flux density, core flow, and recirculation flow rate.

The RWCU system is provided for removal of impurities in the primary system. The cleanup process consists of filtration/ demineralization and serves to maintain a high level of water purity in the reactor coolant.

Typical chemical parametric values for the reactor water are listed in Table 5.2-4 for various plant conditions.

Additional water input to the reactor vessel originates from the CRD cooling water. The CRD water is of approximately feedwater quality. Separate filtration for purification and removal of insoluble corrosion products takes place within the CRD system before the water enters the drive mechanisms and reactor vessel.

During plant conditions other than normal operation, additional inputs and mechanisms are present, some of which are outlined in the following discussion.

b. Plant conditions outside normal operation

During periods of plant conditions other than normal power production, transients take place, particularly with regard to the oxygen levels in the primary coolant. Oxygen levels in the primary coolant vary from the normal during plant startup, plant shutdown, hot standby, and when the reactor is vented and depressurized. The hotwell condensate absorbs oxygen from the air when vacuum is broken on the condenser. Before the startup and input of feedwater to the reactor, vacuum is established in the condenser and deaeration of the condensate takes place by mechanical vacuum pump, SJAE operation, and condensate recirculation. During these plant conditions, continuous input of CRD cooling water takes place as described previously.

1. Plant depressurized and reactor vented

During certain periods, such as refueling and maintenance outages, the reactor is vented to the condenser or to the atmosphere. Under these circumstances the reactor cools, and the oxygen concentration increases to a maximum value of 8 ppm. Equilibrium between the atmosphere above the reactor water surface, the CRD cooling water input, residual radiolytic effects, and the bulk water is established after some time .

2. Plant startup/shutdown

During these conditions, significant changes in oxygen concentration take place.

(a) Plant startup

Depending on the duration of the plant shutdown before 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 decreases rapidly as a function of water temperature increase and corresponding oxygen solubility in water. The oxygen level reaches a minimum of about 20 ppb (0.02 ppm) at a coolant temperature of about 380°F, at which point an increase takes 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.

A further increase in nuclear 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, 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.

(b) Plant shutdown

On 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 takes place due to residual energy, the oxygen concentration in the coolant decreases to a minimum value determined by the steaming rate temperature. If venting is performed, a gradual increase to essentially oxygen saturation at coolant temperature takes place, reaching a maximum value of <8 ppm oxygen.

(c) Oxygen in piping and in 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 any other directly related or associated parts is that of air saturation at ambient temperature. At no time or location in the water phase do oxygen levels exceed the nominal value of 8 ppm. As temperature is increased and oxygen solubility is thus decreased accordingly, the oxygen concentration is 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, inventories may contain less than the maximum limitation of 8 ppm dissolved oxygen.

Conductivity of the reactor coolant is continuously monitored. These measurements provide reasonable assurance of adequate surveillance of the reactor coolant.

Grab samples may be taken from the locations shown in Table 5.2-6 for special and noncontinuous measurements such as pH, oxygen, chloride, and radiochemical measurements.

The relationship of chloride concentration to specific conductance measured at 25°C is shown in Figure 5.2-10. The values essentially bracket the values of common chloride salts or mixtures at the same chloride concentration. Surveillance requirements are based on these relationships.

In addition to this program, limits, monitoring, and sampling requirements are imposed on the condensate, condensate cleanup system, and feedwater to detect and correct off-specification conditions.

The sampling frequency when reactor water has a low specific conductance is adequate for calibration and routine audit purposes. When specific conductance increases and higher chloride concentrations are possible or when continuous conductivity monitoring is unavailable, increased sampling is provided (see plant Technical Specifications).

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

The primary coolant conductivity monitoring instrumentation, ranges, accuracy sensor, and indicator locations are shown in Table 5.2-6.

3. Water purity during a condenser leakage

   To protect against a major condenser tube leak, sufficient instrumentation and monitoring is provided so that there is adequate demineralizer capacity margin available to permit orderly shutdown of the reactor in case of a serious condenser leak. Details are discussed in Section 10.4.1.

5.2.3.2.2.1 Conformance with Regulatory Guide 1.56 (July 1978) - Maintenance of Water Purity in Boiling Water Reactors

Conformance with Regulatory Guide 1.56 of the design and associated instrumentation and operating procedures for the RWCU and condensate cleanup systems is discussed in Sections 5.4.8 and 10.4.6, respectively.

Maintenance of coolant chemistry at a level comparable to the recommendations of Regulatory Guide 1.56 is discussed in Section 10.4.1 and Technical Specifications.

5.2.3.2.3 Compatibility of Construction Materials with Reactor Coolant

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

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

b. Nickel base alloys - Inconel 600 and Inconel 750X

c. Carbon steel and low alloy steel

d. Some 400 series martensitic stainless steel (all tempered at a minimum of 1100°F)

e. Colmonoy and Stellite hard-facing material

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.

Contaminants in the reactor coolant are controlled to very low limits per the reactor water quality specifications. The reactor coolant monitoring and control programs are based on industry guidelines for BWR water chemistry and are designed to protect construction materials from adverse affects.

5.2.3.2.4 Compatibility of Construction Materials with External Insulation and Reactor Coolant

The construction materials exposed to external insulation are:

a. Solution annealed austenitic stainless steels, Types 304, 304L, 316, and 316L

b. Carbon and low alloy steel

Two types of external insulation are employed on BWRs. The reflective metal insulation used does not contribute to any surface contamination and has no effect on construction materials. The nonmetallic insulation used on stainless steel piping and components is in accordance with Regulatory Guide 1.36 (February 1973) and complies with the requirements of the following industry standards:

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

b. RDT-Ml2-1T, 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 prevent damage or contamination during shipment and storage.

These same construction materials may be exposed to external penetration seals. Those penetration seals that could contribute to surface contamination of the construction materials are in accordance with Reg. Guide 1.36 (February 1973) as discussed above, or any similar standard which is at least as stringent as Reg. Guide 1.36 (February 1973).

Because of the high purity water quality of the BWR coolant, leakage exposes materials to essentially the same effects as that of demineralized water which is negligible.

5.2.3.3 Fabrication and Processing of Ferritic Materials

5.2.3.3.1 Fracture Toughness of Ferritic Materials

The ferritic materials of the containment pressure boundary were purchased and impact-tested prior to the issuance of the Summer 1977 Addenda to ASME Section III. These materials have been reviewed and found to be acceptable within the context of GDC 51 based on the following information.

Materials for AE-supplied pipe, valves, and flued heads for systems that are part of the containment pressure boundary are approved impact quality materials as listed in ASME Section III. The impact test requirements were derived from ASME Section III, 1971 Edition, including Winter 1972 Addenda. All materials required to be tested were tested satisfactorily and documented. Records for all AE-supplied containment pressure boundary piping, valves, and flued heads are available for inspection.

The material used for the fabrication of the penetration assemblies, personnel air locks, hatches, and drywell head of the containment pressure boundary are impact quality materials conforming to ASME Section III, subsection B, article 12. All materials required to be tested in accordance with paragraph NC2311 of the Summer 1977 Addenda to ASME Section III were impact-tested satisfactorily in accordance with the requirements of ASME Section III, 1968 Edition, including Winter 1969 Addenda, or later addenda or editions. Records for tested materials of the containment pressure boundary penetration assemblies, personnel air locks, hatches, and drywell head are available for inspection.

The materials of the containment pressure boundary used for the NSSS flued head fittings, main steam piping, and MSIVs, which are also part of the RCPB, meet the GDC 51 requirements for fracture toughness as demonstrated by their conformance to 10CFR50, Appendix G, which is discussed below.

The materials used for the NSSS flued head fittings and main steam piping were satisfactorily impact-tested at 0°F and 70°F, respectively, in accordance with ASME Section III, 1971 Edition with Summer 1972 Addenda. The materials used for the NSSS MSIVs were exempted from toughness testing, but have been reviewed against the criteria of 10CFR50, Appendix G and are considered to be acceptable based on the information provided in Section 5.2.3.3.1.4 below.

5.2.3.3.1.1 Piping

Toughness testing of the main steam piping is in compliance with 10CFR50, Appendix G, since it was tested at +70°F in accordance with ASME Section III, 1971 edition with Summer 1972 Addenda.

5.2.3.3.1.2 Safety/Relief Valves

The SRVs are exempted by the ASME Code from toughness testing because of their 6 inch size. This is consistent with 10CFR50, Appendix G.

5.2.3.3.1.3 Flued Head Fittings

Testing of the flued head fittings is in compliance with 10CFR50, Appendix G. These materials were impact-tested in accordance with ASME Section III, 1971 Edition with Summer 1972 Addenda. The test temperature was 0°F.

5.2.3.3.1.4 Main Steam Isolation Valves

The MSIVs were procured to meet the requirements of the 1968 ASME Nuclear Draft for Pumps and Valves Code, which did not require toughness testing for the subject valve materials. These were exempted because they are subjected to less than 20% of design pressure at temperatures less than 250°F.

The LGS Units 1 and 2 MSIV body materials are A216 Grade WCB carbon steel castings. Table 5.2-12 shows the typical chemical composition and heat treatment of these castings. Although impact tests were not run, these materials are considered to have adequate toughness to meet the current code requirements (i.e., 25 mil lateral expansion). Evidence of this design adequacy is provided in Table 5.2-13 which presents similar MSIV body material data from other BWR projects identified as Projects A through F. These materials received heat treatments equivalent to those experienced in LGS Units 1 and 2.

The bonnet (i.e., cover) materials are A105 Grade 2 forgings. Table 5.2-11 lists available information for the materials used to fabricate the valve covers. These materials were normalized at 1650°F for 12 hours and air cooled. Evidence of toughness for SA105 forgings is given in the July 1978 issue of "Metal Progress", (pages 35-39). This article (Reference 5.3-1) shows charpy V-notch in excess of 25 mils lateral expansion at +40°F and NDTT values no greater than -10°F for SA105 material normalized at 1565°F for 4 hours and air cooled after forging. Similar results would be expected for the LGS MSIV bonnet materials. MSIVs modified with nose guided poppets have bonnets made from SA105 forgings which are equivalent to the original material.

Additional toughness data for SA105 forging materials obtained from fittings in another BWR plant is presented in Table 5.2-10. These materials were normalized at 1650°F for 4 hours and air cooled. The toughness data given is for longitudinally oriented specimens whereas the code requirements are for transverse specimens. However, prior GE impact test experience with carbon steel material indicates it is appropriate to approximate transverse properties at about 40% of the corresponding longitudinal properties. On this basis, the data given in Table 5.2-10 demonstrates that the transverse properties meet the 25 mil lateral expansion code requirements.

5.2.3.3.2 Control of Welding

5.2.3.3.2.1 Regulatory Guide 1.50 (May 1973) - Control of Preheat Temperature Employed for Welding of Low Alloy Steel

This guide delineates preheat temperature control requirements and welding procedure qualifications supplementing those in ASME Sections III and IX.

For the GE scope of supply, Regulatory Guide 1.50 is not employed as a design basis for LGS. However, the procedures in use either conform with the guide or are evaluated as satisfying the guide through the use of alternate approaches.

Preheat temperatures employed for the welding of low alloy steel meet or exceed the recommendations of ASME Section III, Subsection NA. Either components were held for an extended time at preheat temperature to ensure removal of hydrogen, or preheat was maintained until postweld 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 the Bechtel scope of supply, this guide is followed except for the following alternate approaches:

a. With respect to the position in Paragraph C.1.a of the guide that the procedure qualification should require that a minimum preheat and a maximum interpass temperature be specified, there is conformance when impact testing in accordance with Subarticle 2300 of ASME Section III, Division 1, is required. The maximum interpass temperature shall be 500°F unless otherwise specified. When impact testing is not required, specification of a maximum interpass temperature in the welding procedure is not necessary to ensure that the required mechanical properties are met. The minimum preheat temperatures of ASME Section III, Appendix D are required to be met regardless of whether impact testing is required.

b. Regarding the position in Paragraph C.1.b of the guide that the procedure qualification should require that the welding procedure be qualified at the minimum preheat temperature, the welding procedure qualification requirements of ASME Sections III and IX are considered to be more than adequate.

c. Regarding the position of Paragraph C.2 of the guide that, for production welds, the preheat temperature should be maintained until a postweld heat treatment has been performed, the position is conformed with for Class 1 pressure vessels with nominal thickness greater than 1 inch. Maintenance of preheat beyond completion of welding until postweld heat treatment is not required for thinner sections since experience has indicated that delayed cracking in the weld or heat-affected zone is not a problem.

   Current usage of low alloy steel in piping, pumps and valves is minimal and normally is limited to Class 3 construction. When low alloy steel piping, pumps and valves are used, preheat is maintained until welding is complete but not until postweld heat treatment is performed, since the conditions which cause delayed cracking in the weld or heat-affected zone are not present.

   When the regulatory guide positions or above alternate approaches are not met, the weld is subject to rejection; however, the soundness of the weld may be verified by an acceptable examination procedure.

5.2.3.3.2.2 Regulatory Guide 1.34 - Control of Electroslag Weld Properties

No electroslag welding was performed on BWR components.

5.2.3.3.2.3 Regulatory Guide 1.71 - Welder Qualification for Areas of Limited Accessibility

The qualification for areas of limited accessibility is discussed in Section 5.2.3.4.2.3.

5.2.3.3.2.4 Regulatory Guide 1.43 - Control of Stainless Steel Weld Cladding of Low Alloy Steel Components

This guide applies to welding of cladding to low alloy steels made to coarse grain practice. LGS vessel plate and nozzle forgings are made to fine grain practice and a low heat input process is used. Other components are not clad. Therefore, the guide is not applicable.

5.2.3.3.3 Nondestructive Examination of Ferritic Tubular Products - Regulatory Guide 1.66 (October 1973)

This regulatory guide was withdrawn on September 28, 1977, by the NRC because the additional requirements imposed by the guide were satisfied by the ASME Code. The following discussion applies only to activities performed prior to that time.

This guide described a method of implementing requirements acceptable to the NRC regarding nondestructive examination requirements of tubular products used in the RCPB.

For the GE scope of supply, wrought tubular products were supplied in accordance with applicable ASTM/ASME material specifications. Also, the specification for the tubular product used for CRD housings specified ultrasonic examination to paragraph NB-2550 of ASME Section III.

These RCPB components met the requirements of the ASME Codes existing at the time of the placement of the order that predated Regulatory Guide 1.66. At the time of the placement of the orders, the 10CFR50, Appendix B requirements and ASME code requirements ensure adequate control of quality for the products.

For the Bechtel scope of supply, there was partial conformance with the guide in that tubular products used for Class 1, 2, and 3 components were ultrasonically examined in accordance with the requirements of ASME Section III, Division 1, 1971 Edition with Addenda through Winter 1971. In addition, the ultrasonic examination procedures were in accordance with ASTM E213-68.

5.2.3.3.4 Moisture Control for Low Hydrogen, Covered Arc Welding Electrodes

All low hydrogen covered welding electrodes are stored in controlled storage areas, and only authorized persons are permitted to release and distribute electrodes. Electrodes are received in hermetically sealed canisters. After removal from the sealed containers, electrodes which are not immediately used are placed in storage ovens which are maintained at about 250°F (generally 200°F minimum).

Electrodes are distributed from sealed containers or ovens as required. At the end of each work shift, unused electrodes are returned to the storage ovens. Electrodes which are damaged, wet, or contaminated are discarded. If any electrodes are inadvertently left out of the ovens for more than one shift, they are discarded or reconditioned in accordance with the manufacturer's instructions.

5.2.3.4 Fabrication and Processing of Austenitic Stainless Steels

5.2.3.4.1 Avoidance of Stress-Corrosion Cracking

5.2.3.4.1.1 Avoidance of Significant Sensitization - Regulatory Guide 1.44 (May 1973) - Control of the Use of Sensitized Stainless Steel

The purpose of Regulatory Guide 1.44 is to address 10CFR50, Appendix A, GDC 1 and 4, and the Appendix B requirements to control "the application and processing of stainless steel to avoid severe sensitization that could lead to stress-corrosion cracking."

This guide is not used by GE as a design basis for LGS. However, the procedures used either conform with the guides or are evaluated as satisfying the guide through the use of alternate approaches.

All austenitic stainless steel was purchased in the solution heat treated condition in accordance with applicable ASME and ASTM specifications. Cooling rates from solution heat treating temperatures were required to be rapid enough to prevent sensitization. Non-sensitization was verified using ASTM A262, Practice A or E methods.

Material changes have been made to minimize the possibility of IGSCC. Except certain valves, for which alternate means of mitigating IGSCC or inspection procedures were employed to comply with the intent of Regulatory Guide 1.44, wrought austenitic stainless steel in the RCPB was changed to low carbon (0.025% maximum) material. Piping in the following systems, from the reactor vessel to the first isolation valve, is manufactured from low carbon austenitic stainless steel (AISI 316L, maximum carbon content of 0.02%): the core spray system and the RHR system (shutdown cooling suction line, shutdown cooling return line, LPCI). The RWCU system uses 316L from the recirculation system connection through the containment penetration. The recirculation system piping is manufactured from low carbon high strength austenitic stainless steel (316K). Thus, there is no piping that is nonconforming and service sensitive, as defined in NUREG-0313 (Rev 1). 316L seamless small-bore piping and forged fittings (2 inch and smaller) are not subject to the 0.02% maximum carbon content requirement.

For manual welds with gas tungsten arc and shielded metal arc welding processes, heat input was limited by weaving and welding technique restrictions. Nonweaving (stringer bead) techniques were used where possible. Where required, weaving was controlled to meet the following bead width limits:

a. For gas tungsten arc, the lesser of 5 times the filler wire diameter or 7/8 inch

b. For shielded metal arc, the lesser of 4 times the electrode core wire diameter or 5/8 inch.

For machine, automatic, and manual welding with processes except gas tungsten arc and shielded metal arc, heat input for piping was restricted to 50,000 joules/in. Interpass temperature was restricted to 350°F for all stainless steel welds. 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.

For welding of short sections of 316L pipe to forged 316 LPCI and core spray valves adjacent to the reactor, heat was controlled by use of a demineralized water heat sink. The 316L pipe sections were welded to the valves with a gas tungsten arc root pass with argon backing. Subsequent welds were made with a demineralized water backing heat sink as a means of mitigating IGSCC.

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

These controls were used to avoid severe sensitization and to conform with the intent of Regulatory Guide 1.44.

For the Bechtel scope of supply, the guide is followed except for the following alternate approaches:

a. Regarding the position in Paragraph C.3 of the guide which discusses testing for non-sensitization, there is partial conformance in that all austenitic stainless steels are furnished in accordance with applicable ASME or ASTM material specifications. Testing to determine susceptibility to intergranular attack is performed only when required by the material specification.

b. Regarding the position of Paragraph C.6 of the guide which discusses intergranular corrosion testing, welding practices are controlled to avoid severe sensitization, and heat treatment in the temperature range 800°F to 1500°F is not permitted. Since severe sensitization is avoided, testing to determine susceptibility to intergranular attack is not performed.

5.2.3.4.1.2 Process Controls to Minimize Exposure to Contaminants

Exposure to contaminants capable of causing stress-corrosion cracking of austenitic stainless steel components is avoided by carefully controlling all cleaning and processing materials that contact the stainless steel during manufacture and construction.

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

The degree of surface cleanliness achieved by the GE procedures meets the requirements of Regulatory Guide 1.44 and Regulatory Guide 1.37.

Regulatory Guide 1.37 and the associated ANSI N45.2.1 are not specifically applied in the Bechtel scope of work. However, the procedures in use are generally equivalent to the regulatory guide and standard.

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.

5.2.3.4.1.4 Noble Metals Chemical Addition

Noble Metals added to the wetted surfaces of the reactor vessel and internals in combination with low rates of feedwater hydrogen injection (Hydrogen Water Chemistry), provides improved mitigation of Stress Corrosion Cracking of the Noble Metals treated components and associated piping.

5.2.3.4.2 Control of Welding

5.2.3.4.2.1 Avoidance of Hot-Cracking - Regulatory Guide 1.31 - Control of Ferrite Content in Stainless Steel Weld Metal

Regulatory Guide 1.31 describes an acceptable method of implementing requirements with regard to the control of welding when fabricating and joining austenitic stainless steel components and systems.

Per the implementation section of Regulatory Guide 1.31 (Rev 3), April 1978, of the guide, the guide is not applicable to LGS. Nevertheless, for explanatory purposes, LGS procedures with reference to the guide are discussed below.

GE employs Regulatory Guide 1.31 (Rev 1), June 1973, as a design basis for LGS, with certain alternate approaches:

Written welding procedures which are approved by GE are required for all primary pressure boundary welds. These procedures comply with the requirements of ASME Sections III and IX and applicable regulatory guides.

All austenitic stainless steel weld filler materials were required by specification to have a minimum of 5% ferrite. Prediction of ferrite content was made by using the chemical composition in conjunction with the Schaeffler diagram. The use of the 5% minimum limit for ferrite content determined by the Schaeffler diagram has been shown to be adequate to prevent hot-cracking in austenitic stainless steel welds. An extensive test program performed by GE, with the concurrence of the NRC, demonstrated that controlling weld filler metal ferrite at 5% minimum (by Schaeffler diagram) resulted in production welds which met the requirements of Regulatory Guide 1.31 "Control of Stainless Steel Welding." A total of approximately 400 production welds in five BWR plants were measured and all welds met the requirements of BTP MTEB 5-1 "Interim Regulatory Position on Regulatory Guide 1.31, Control of Stainless Steel Welding."

Bechtel employs an alternate approach to the guide:

With respect to Paragraph C.1 of the guide, the delta ferrite determination for consumable inserts, electrodes, rod or wire filler metal used with gas tungsten arc welding process and the plasma arc welding process is predicted from a chemical analysis of as-manufactured material using the constitutional diagram in ASME Section III, figure NX-2433.1-1.

As an alternate to the magnetic method of determining the ferrite content of an undiluted weld deposit as specified in Paragraph C.1, C.2 and C.3, the chemical analysis method specified in ASME Section III, NX-2430 may be used.

5.2.3.4.2.2 Regulatory Guide 1.34 - Electroslag Welds

Electroslag welding was not employed for RCPB components.

5.2.3.4.2.3 Regulatory Guide 1.71 (December 1973) - Welder Qualification for Areas of Limited Accessibility

Regulatory Guide 1.71 recommends that weld fabrication and repair for wrought low alloy and high alloy steels, or other materials such as static and centrifugal castings and bimetallic joints, should comply with the fabrication requirements of ASME Sections III and IX. It also recommends additional performance qualifications for welding in areas of limited access.

For the GE scope of supply, Regulatory Guide 1.71 is not employed as a design basis for LGS. However, the procedures in use either conform with the guide or alternate approaches are used that are evaluated as satisfying the intent of the guide.

All ASME Section III welds were fabricated in accordance with the requirements of ASME Sections III and IX. There are few restrictive welds involved in the fabrication of BWR components. The welder qualification for welds with the most restrictive access was accomplished by mockup welding. Mockups were examined by radiography or sectioning.

For the Bechtel scope of supply, alternate approaches are employed:

> Performance qualifications for personnel who weld under conditions of limited access, as defined in Regulatory Position C.1, are maintained in accordance with the applicable requirements of ASME Sections III and IX. Additionally, responsible site supervisors are required to assign only the most highly skilled welders to limit access welding. Of course, welding conducted in areas of limited access is subjected to the required nondestructive testing and no waiver or relaxation of examination methods or acceptance criteria because of the limited access is permitted.
>
> Requalification is required when any of the essential variables of ASME Section IX are changed or when any authorized inspector questions the ability of the welder to perform satisfactorily the requirements of ASME Sections III or IX.
>
> Production welding is monitored and welding qualifications are certified in accordance with the alternate approaches discussed above.

5.2.3.4.3 Nondestructive Examination of Tubular Products - Regulatory Guide 1.66

For GE scope of supply, wrought tubular products were supplied in accordance with applicable ASTM/ASME material specifications. The specification for the tubular product used for CRD housings requires ultrasonic examination in compliance with ASME Section III, paragraph NB-2550.

All tubular products meet the requirements of ASME Codes existing at time of placement of order which predated Regulatory Guide 1.66.

Bechtel conformance with Regulatory Guide 1.66 is discussed in Section 5.2.3.3.3.

5.2.4 PRESERVICE/INSERVICE INSPECTION, EXAMINATION AND TESTING OF THE REACTOR COOLANT PRESSURE BOUNDARY

The construction permits for LGS Units 1 and 2 were issued in June 1974. Based on this date, 10CFR50.55a requires that the preservice inspection program for the RCPB meet the examination requirements set forth in ASME Section XI, 1971 edition with addenda through Winter 1972 or, alternatively, the examination requirements of subsequent editions and addenda, subject to the limitations and modifications listed in 10CFR50.55a. The LGS preservice inspection programs follow the alternative requirement.

Specifically, the Unit 1 PSI program (with the exception of the RPV and the operability testing of safety-related pumps and valves) meets the requirements of ASME Section XI, 1974 edition with addenda through Summer 1975, as modified by Appendix III of the Winter 1975 Addenda and paragraph IWA-2232 of the Summer 1976 Addenda.

At the time the LGS PSI program commenced, the latest edition of the code permissible for use was the Summer 1975 Addenda. In an effort to take advantage of improved UT methods, Appendix III of the Winter 1975 Addenda and paragraph IWA-2232 of the Summer 1976 were used. Although these items were not specifically referenced by 10CFR50.55a, they are equivalent to the comparable portions of the Summer 1978 Addenda (which was subsequently approved by 10CFR50.55a) as long as Section XI Appendix III indications greater than 50% DAC are recorded. The LGS ISI procedures do record such indications.

Supplement 7 of Appendix III permits the use of Appendix III for austenitic piping welds with certain modifications. It is our position, consistent with the PSI/ISI industry, that Appendix III (at 50% DAC recording) is more appropriate for austenitic piping weld examination than article 5 of ASME Section V. Thus for austenitic piping welds:

a. All of the Supplement 7 modifications are being used.

b. Examination sensitivity is ensured through the calibration process.

c. Where one-sided access only occurs and penetrations cannot be confirmed, a one-sided access limitation is noted in the data package for that weld.

When using Section XI, Appendix III for either ferritic or austenitic piping welds, the following applies:

a. All indications showing signal amplitudes equal to or in excess of 20% of the reference response are evaluated to the extent that the level II or III examiner can determine their true nature.

b. The owner evaluates and takes corrective action for the disposition of any indication investigated and found to be other than geometrical or metallurgical in nature.

The Unit 1 RPV PSI program meets the requirements of ASME Section XI, 1980 edition with addenda through Winter 1980. The Unit 2 preservice inspection program and Units 1 and 2 programs for the preservice testing of safety-related pumps and valves meet the requirements of ASME Section XI, 1980 edition with addenda through Winter 1981.

For certain ASME Section XI requirements that have been determined to be impractical in the course of inspecting the components, the licensee has submitted and will submit requests for relief from the requirements to the NRC in accordance with the provisions of 10CFR50.55a.

In accordance with 10CFR50.55a, throughout the service life, RCPB components (including supports) will meet the ISI requirements, except design and access provisions and preservice examination requirements, set forth in the ASME Section XI edition and addenda that become effective, to the extent practical within the limitations of design, geometry, and materials of construction of the components. In accordance with 10CFR50.55a, inservice examination of components, inservice tests to verify operational readiness of safety-related pumps and valves, and system pressure tests conducted during the initial 10 year inspection interval will comply with the ASME Section XI edition and addenda in effect 12 months prior to the date of issuance of the operating license. In accordance with 10CFR50.55a(g)(4)(ii), the initial 10 year inspection interval commences with commercial operation. The successive 10 year inspection intervals will comply with the ASME Section XI edition and addenda in effect 12 months prior to the start of the 10 year inspection interval.

5.2.4.1 System Boundary

LGS piping was originally designed to ANSI B31.7. For the purpose of inservice inspection, ANSI B31.7, Class I is considered equivalent to ASME Section III, Class 1. The PSI/ISI program includes figures showing the systems or portions of systems within the scope of ASME Section XI. The applicable PSI/ISI programs are described in References 5.2-8 through 5.2-18. Any necessary requests for relief are addressed in these documents.

5.2.4.2 Accessibility

The access provided for the various components and parts is in accordance with ASME Section XI, Subarticle IWA-1500.

a. Sufficient space has been provided for personnel and equipment to perform examinations of the RCPB, as follows:

   1. Piping welds - The access provided depends on whether ultrasonic, surface, or visual examinations are performed.

   2. Pumps and valves - Space has been provided to disassemble and reassemble pump and valves. For visual examination, space for lighting and access sufficient to permit examination of the inner surface has been provided. For ultrasonic examination and depending on the specific design of the weld, access to equipment has been provided.

   3. Supports - Access provisions for supports requiring examination has been made and depends on the specific type and design detail of the supports.

   4. Reactor vessel - The reactor vessel insulation has been designed to allow access for examination, by a remotely operated scanning device, of the vessel longitudinal and circumferential welds, nozzle-to-vessel inside radiused sections, and nozzle-to-safe-end welds.

      The vessel-to-flange weld and flange ligaments are accessible during refueling. The closure head-to-flange weld and closure head circumferential and meridional welds are accessible from the outside.

      The vessel-to-skirt weld is accessible during refueling.

Closure studs, nuts, and washers can be removed when the vessel head is removed. This provides adequate access.

Portions of the vessel cladding and interior are accessible by removal of components. Portions of the interior attachments and core support structures are accessible by removal of components.

Only the peripheral CRD housings are accessible for volumetric examination.

b. Capability for removal and storage of structural members, shielding components, and insulation is provided.

c. Hoists and other handling machinery necessary to support inservice inspection are provided.

d. Equipment, personnel, and procedures for alternative examinations that may be required will be provided.

e. Repair and replacement procedures are provided for system components and parts, when necessary.

5.2.4.3 Examination Techniques and Procedures

a. The techniques and procedures for surface, visual, and volumetric examinations are in compliance with ASME Section XI, Subarticle IWA-2200.

b. Alternate examination methods are acceptable provided the results are equal or superior to the methods of Subarticle IWA-2220. The acceptance criteria for alternate examination methods shall be in accordance with Subarticle IWB-3100.

5.2.4.4 Inspection Intervals

Inservice inspection is performed according to the schedule outlined in Subarticle IWB-2400; all required examinations being completed within a nominal 10 year period, hereafter referred to as the inspection interval. Inspections will be scheduled to coincide with normal plant or system outages.

5.2.4.5 Examination Categories and Requirements

The PSI/ISI program provides a listing of the Class 1 components or parts, the corresponding ASME Section XI item number, examination category, the required method of examination, and the extent and frequency of examination. The applicable PSI/ISI programs are identified in References 5.2-8 through 5.2-18.

5.2.4.6 Evaluation of Examination Results

a. The standards for evaluation of examination results are in accordance with Article IWB-3000.

b. The repair procedures are in agreement with Article IWA-4000.

c. Flaw indications are characterized in accordance with the requirements of IWB-3420 and compared with the acceptance standards specified in IWB-3410.

d. The replacement procedures are in agreement with Article IWA-4000.

e. Summary reports for the RPV and piping PSI will be submitted in accordance with subarticles IWA-6000.

5.2.4.7 <u>System Leakage and Hydrostatic Pressure Tests</u>

The Class 1 leakage and hydrostatic pressure test program meets the requirements of ASME Section XI, Article IWB-5000, "System Leakage and Hydrostatic Pressure Tests." The Technical Specification requirements for operating limitations for heatup, cooldown, and system hydrostatic pressure testing will be met during such testing.

5.2.4.8 <u>Augmented Inservice Inspection</u>

Class 1 components will receive augmented inservice inspections in accordance with the documents listed below to the extent specified in the applicable PSI/ISI programs identified in References 5.2-8 through 5.2-18.

a. Regulatory Guide 1.150 (Rev 1), February 1983, with Appendix A, Alternate Method. "Ultrasonic Testing of Reactor Vessel Welds During Preservice and Inservice Examinations" and NRC Generic Letter 83-15 dated March 23, 1983.

b. NUREG-0619, November 1980, "BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking. Resolution of Generic Technical Activity A-10"

   The LGS feedwater nozzle has been modified. The new configuration is the triple-sleeve with two sister ring seals and an unclad nozzle. This ensures the longest ISI intervals in accordance with NUREG-0619.

c. IE Bulletin 80-07, April 4, 1980, "BWR Jet Pump Assembly Failure" (NUREG/CR-3052)

d. IE Bulletin 80-13, May 12, 1980, "Cracking in Core Spray Spargers"

e. NUREG-0313 (Rev 1), July 1980, "Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping. Resolution of Generic Technical Activity A-42," applicable prior to issuance of Generic Letter 88-01.

   Following issuance of Generic Letter 88-01, the criteria of NUREG-0313 (Rev 2), January 1988 become applicable for both LGS Units 1 and 2, with full implementation effective for both units by the second refueling outage at Unit 1. The ISI program for weldments in piping shall conform with the NRC staff positions identified in Generic Letter 88-01 and BWRVIP-75-A, "BWR Vessel and Internals Project Technical Basis for Revisions to Generic Letter 88-01 Inspection Schedule," as documented in the licensee's responses to Generic Letter 88-01 (References 5.2-19,20,21,23 and 25), approved by the NRC (References 5.2-24 and 5.2-26), and required by amended Technical Specifications for Units 1 and 2 (Reference 5.2-28 and 5.2-33). Details for schedule, methods, personnel, and sample expansion shall be included as augmented inspection requirements. Correspondence relating to Generic Letter 88-01 is specified in References 5.2-19 through 5.2-28.

f. BTP MEB 3-1 (NUREG-0800) for high energy piping between containment isolation valves and first outboard restraint for which no breaks are postulated.

In addition, high energy fluid system piping between containment isolation valves will receive an augmented examination as follows:

a. Protective measures and structures are located, to the greatest extent possible, so as not to prevent access for inservice inspections.

b. High energy fluid system piping between containment isolation valves is required to be either 100% volumetrically examined (circumferential welds) during each examination interval or examined in accordance with the Risk Informed Inservice Inspection Program as applied to these welds.

c. High energy piping requiring ISI receives a baseline (preservice) examination to establish the integrity of the original condition of the welds.

d. Augmented examination for high energy piping is maintained out to outboard restraints.

e. Welds between outboard containment isolation valves and piping restraints will be included in the PSI and the ISI plans.

5.2.5 Reactor Coolant Pressure Boundary Leak Detection System

5.2.5.1 Design Bases

The leak detection system is designed to:

a. Detect the occurrence of and alert operating personnel to abnormal leakage from the RCPB.

b. Detect leakage from selected portions of systems located outside the primary containment, and not a part of the RCPB.

c. Remain functional following a SSE, except as discussed in Sections 5.2.5.2.1.4 and 5.2.5.2.1.5.

5.2.5.2 Description

The RCPB leak detection system consists of temperature, pressure, flow, and radiation sensors, and associated instrumentation and alarms. The system detects, annunciates, and, in certain cases, isolates abnormal leakage in the following systems:

a. Main steam lines

b. RWCU system

c. RHR system

d. RCIC system

e. Feedwater system

f. HPCI system

g. Reactor recirculation system

h. Core spray system

A summary of isolation and/or alarms of affected systems and the methods used appears in Table 5.2-7. The table shows those leak detection methods which detect gross leakage and initiate immediate automatic isolation. Those methods which are capable of detecting small leaks initiate an alarm in the control room, at which time the operator can either manually isolate the leaking system or take other appropriate action.

5.2.5.2.1 Detection of Abnormal Leakage Within the Primary Containment

Leaks within the primary containment are detected by continuously monitoring for:

a. Abnormally high pressure and temperature within the primary containment

b. Rapid level increase in drywell equipment and floor drain sumps

c. A decrease in the reactor vessel water level

d. High gaseous radiation level in the primary containment atmosphere

e. High containment air cooler condensate flow.

In addition to these leak detection methods, selected RCPB components within the primary containment are provided with their own leak detection devices. While some of the methods provided for detecting leakage within the primary containment are not redundant in themselves, it is not postulated that any one event could render all means of leak detection inoperable. Each of the leak detection methods are discussed below. The systems and equipment provided at LGS meet or exceed the requirements and recommendations of ANSI/ISA S67.03 (1982).

5.2.5.2.1.1 Drywell Temperature Monitoring

Drywell temperature monitoring provides an indirect method of detecting RCPB leakage.

The drywell area unit coolers circulate and cool the drywell atmosphere to maintain the drywell at its design operating temperature. Cooling water is supplied to the unit coolers by the DCWS (Section 9.2.10). A temperature rise in the drywell will indicate the presence of reactor coolant or steam leakage and is detected by the drywell temperature monitors located at various elevations and at the inlet and outlet of the air coolers. A discussion of indications and alarms for drywell temperature in accordance with Regulatory Guide 1.45 is given in Section 7.5.

5.2.5.2.1.2 Drywell Pressure Monitoring

Drywell pressure monitoring provides an indirect method of detecting RCPB leakage.

The drywell normally ranges from slightly below to slightly above atmospheric pressure during reactor operation. The pressure typically fluctuates slightly as a result of barometric pressure changes and/or outleakage. A pressure rise above the normally indicated value indicates leakage within the drywell. Drywell pressure monitoring is shown in drawings M-42 and M-57. A discussion of indications, alarms, and protective functions for drywell pressure in accordance with Regulatory Guide 1.45 is given in Sections 7.2, 7.3, 7.4, and 7.5.

5.2.5.2.1.3 Drywell Sump Level Monitoring

All leakage from RCPB components inside the drywell, with the exception of leakage from the MSRVs (Section 5.2.5.2.1.8), flows directly to either the drywell equipment drain sump or the drywell floor drain sump. There are no other reservoirs in the drywell of sufficient capacity to prevent leakage from draining directly to either of these sumps. Both drain sumps are identically sized, horizontal cylindrical tanks located inside the reactor vessel pedestal below the diaphragm slab and vented to the drywell atmosphere. The liquid radwaste collection system piping and instrumentation diagram is given in drawing M-61. These drain sumps are discussed below:

a. Drywell equipment drain sump - Certain RCPB components within the drywell are, by the nature of their design, normally subject to a limited amount of leakage. These components include pump seals, valve stem packings, and other equipment that cannot practicably be made to be completely leak-tight. These leakages are piped directly to the drywell equipment drain sump. All of the various drains are open only to the equipment they serve, thereby receiving leakage only from identified sources. Background leakage to this sump is determined during initial plant operation. Rates of leakage collection in excess of background indicates abnormal RCPB leakage.

b. Drywell floor drain sump - Leakage from RCPB components inside the primary containment which are not normally subject to leakage is collected by the drywell floor drain sump. This leakage, which may originate from any number of sources within the drywell, is transported to the sump via the floor drain network within the drywell. Thus, separation of unidentified leakage from the identifiable leakage routed to the equipment drain sump ensures that a small unidentified leakage that is of concern will not be masked by a larger, acceptable, identified leakage.

c. Each sump tank has its own level transmitter which is monitored by a dedicated processing unit. Normally closed drain valves are provided, enabling the level in each tank to increase as leakage drains into them. The processing unit calculates an average leak rate for a given measurement period by establishing the amount of increase in level that occurred during the period, and converting that value into volumetric terms (gpm). The processing units provide an alarm in the main control room each time the average leak rate changes by a predetermined valve since the last time that alarm was reset. The setpoint is a 1 gpm change in unidentified leakage collected in the drywell floor drain sump tank, and a 2 gpm change in identified leakage collected in the drywell equipment drain tank.

Alarms are also generated in the main control room for high total average leak rate. The high total average leak rate alarm setpoints can be adjusted at the processing unit, which is located in the main control room, as the amount of acceptable identified leakage changes during operation. Indication of the leakage rates is provided in the main control room on panel-mounted indicators. Sump tank levels (in gallons) are provided on CRT displays from the ERFDS system.

An alternative method to quantify RCPB leakage in the drywell is used when the Drywell floor drain sump monitoring system in unavailable. The alternate method allows the Drywell floor drain sump to overflow, through installed piping, to the Drywell equipment drain monitoring system to verify that RCPB leakage is within allowable limits. In this condition, all input into the Drywell equipment drain sump will be considered unidentified drywell leakage.

Level switches, which are independent of the level transmitters, open the sump tank drain valves when the level increases to an upper setpoint value and keep them open until the level decreases to lower setpoint value. The level switches then close the drain valves and reset the processing units to start a new measurement period. The measurement period must be long enough to ensure that the level transmitter loop can adequately detect the increase in level that would correspond to the 1 gpm and 2 gpm changes in leak rates described above, and yet short enough to ensure that such a leak rate will be detected within an hour. The measurement period will be less than 1 hour.

The transmitters which are located in the reactor enclosure and the processing units which are located in the main control room, are accessible during normal plant operation for calibration. The transmitters can be isolated from the sump tanks by existing bypass manifolds. Zero and span adjustments can be made using portable test equipment. The processing unit functions can be calibrated by applying known input levels at the unit and observing the response.

The DSLMS is comprised of the processing units, level transmitters, control room leakage flow indicators and interconnecting raceway and cables. The DSLMS has been demonstrated to remain operational after an SSE. The DSLMS is energized by Class 1E power. The Class 1E power to the panel is provided with a Class 1E fuse and circuit breaker in series to meet separation requirements. The DSLMS is automatically shed from the Class 1E power in the event of a LOCA (the load shedding relay, however, is not qualified for Class 1E service). Drawing M-61 shows the piping and instrument diagram for this system.

In addition to the sump level monitoring system described above, the discharge from each sump is monitored by a flow element. The measured flow rate is integrated and recorded in the control room. A control room alarm is also provided to indicate excessive discharge rates. These indications and alarms are provided in accordance with Regulatory Guide 1.45.

5.2.5.2.1.4 Drywell Air Cooler Condensate Drain Flow Monitoring

The drywell air cooler condensate drain flow monitoring system consists of six flow sensors and their associated flow transmitters that provide inputs to a separate flow monitoring device. The flow sensors are mounted in the drywell air cooler drain lines located in the drywell. There are a total of eight coolers. Two flow sensors measure the flow from two air cooler drain line headers. Four additional flow sensors measure the drain flow from each of the remaining coolers which are piped separately to the drywell floor drain sump tank. The outputs from the six flow transmitters which receive their input from the flow sensors are added to provide a total continuous drain flow rate by the use of two summing units. The continuous drain flow rate is monitored by a flow switch, located in the main control room, which will alarm if the rate exceeds 1 gpm over the preset identified leak rate. The plant operator establishes the acceptable identified drain flow rate and adjusts the setpoint of this flow switch accordingly (i.e., if the identified flow rate is established at 5 gpm, the system will be set to alarm at 6 gpm). The requirement to detect a 1 gpm increase in unidentified leakage within an hour is met by monitoring the continuous flow rate. Indication of the total continuous drain flow rate is provided in the main control room on panel-mounted indicators and CRT displays from the ERFDS system.

The calibration and test interval for the flow sensors, transmitters, and processing units is in accordance with the manufacturer's recommendation.

The drywell air cooler condensate drain flow monitoring system is qualified to withstand an OBE. The system is energized by Class 1E power. Drawing M-87 shows the piping and instrument diagram for this system.

5.2.5.2.1.5 Containment Airborne Radioactivity Monitoring

The primary containment is continuously monitored for airborne gaseous radioactivity. A drywell air sample is extracted via sample line through containment penetration X-117B at el 292', area 16. Air flow through the monitoring system is assured by the suction created by a vacuum pump. The air sample is surveyed by the GM tubes in the sampling chamber for its radioactivity content. The air sample is returned to the drywell through the same containment penetration. The level of radioactivity is recorded in the main control room in counts per minute. The range is from 10 cpm to 10^6 cpm. The corresponding concentration is 10^{-6} μCi/cc to 10^{-1} μCi/cc. Particulate and iodine monitors are not provided due to the substantial limitations of their usefulness as described below.

The noble gas monitoring equipment is shown in drawing M-26. It is not designed to be operable following an SSE.

Radioactivity level indication and alarms for loss of sample flow, high radiation, and down-scale are provided locally and in the main control room. Activity level indication in the control room is provided on a strip-chart recorder to provide trend information.

The operability of the sensor and the electronic circuitry can be verified during operations from the auxiliary equipment room. A check source is supplied with the monitor. Sample connections are also provided to facilitate additional sampling for laboratory analysis.

The radiation monitor is capable of being calibrated during power operation and will be calibrated in accordance with Technical Specifications requirements (Chapter 16).

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

5.2.5.2.1.5.1 Source of Leakage

a. Location of Leakage - The amount of activity that would become airborne following a 1 gpm leak from the RCPB will vary depending on the leak location and the coolant temperature and pressure. For example, a feedwater pipe leak may have concentration factors of 100 to 1000 lower than a recirculation line leak. A steam line leak may be a factor of 50 to 100 lower in iodine and particulate concentrations than the recirculation line leak, but the noble gas concentrations may be comparable. An RWCU leak upstream of the demineralizers and heat exchangers may 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 directly correlated to a quantity of leakage without knowing the source of the leakage.

b. Coolant Concentrations - Variations in coolant concentrations during operation can be as much as two 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 are given in Reference 5.2-6. An increase in the coolant concentrations could give increased containment concentrations when no increase in unidentified leakage occurs.

c. Other Sources of Leakage - Because 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.

5.2.5.2.1.5.2 Drywell Conditions Affecting Monitor Performance

a. 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. Due to these high equilibrium activity levels, the activity increase due to a small increase in leakage may be difficult to detect within a short period of time.

b. Purge and Pressure Release Effects - Changes in the detected activity levels have occurred during containment venting operations. These changes are of the same order of magnitude as approximately a 1 gpm leak and are sufficient to invalidate the results from iodine and particulate monitors.

c. Plateout, Mixing, Condensation, Fan Coolant Depletion - Plateout effects on measured iodine and particulate levels 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 and sample lines will remove iodines and particulates from the air. Variations in flow, temperature, and number of coolers will affect the plateout fractions. Plateout within the detector sample chamber will also add to the reduction of the iodine and particulate activity levels. The uncertainties in any estimate of plateout effects could be as much as one or two orders of magnitude.

5.2.5.2.1.5.3 Physical Properties and Capabilities of the Detector

a. Detector Range - The detector was chosen to ensure that the operating range covered the concentrations expected in the drywell. The operating range of the noble gas monitor is: 10^{-6} μCi/cc to 10^{-1} μCi/cc.

b. Sensitivity - In the absence of background radiation and equilibrium drywell activity levels, the detector has the following minimum sensitivity: 10^{-6} μCi/cc for Xe-133.

c. Counting Statistics and Monitor Uncertainties - In theory, this radioactivity monitor is statistically able to detect increases in concentration as small as 2 or 3 times the square root of the count rate, i.e., at $1x10^6$ cpm an increase of $2x10^3$ or 0.2% is detectable; at 100 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 (±5%), sample flow (±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.

d. Monitor Setpoints - Due to the uncertainty and extreme variability of the radioactivity concentrations to be measured in the containment, the use of tight alarm setpoints on the radioactivity monitor would not be practical or useful. The setpoint, 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. The alarm setpoints for the radiation monitors are set significantly above normal readings to prevent nuisance alarms.

e. 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., SRV leakage) that further complicate the correlation.

Thus, the 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 that 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, containment temperature and pressure, air cooler condensate flow, 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. 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, and abnormal increases will be investigated.

Because 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 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.

Radiation monitor alarms are not set to levels that are intended to correspond to the RCPB leakage levels because such correlations are not valid. Because the containment airborne activity levels vary by orders of magnitude during operation due to power transients, spiking, steam leaks, and outgassing from sumps, an appropriate alarm setpoint is determined by the operator based on experience with the specific plant. A setpoint level of 10 times the 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 one hour and therefore could not be considered as any more than a qualitative indication of the presence of abnormal leakage.

Due to the sum total of the uncertainties identified in the previous paragraphs, iodine and particulate monitors are not relied upon for immediate leak detection purposes. The noble gas monitor is used to give supporting information to that supplied by the sump discharge monitoring, and it 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 alarm setpoint. Grab sampling and laboratory analyses of airborne particulate, noble gas, and iodine may be used to characterize leakage detected by other means.

5.2.5.2.1.6 Reactor Vessel Head Seal Leak Detection

The reactor vessel head is provided with two concentric metallic seals, with a leak-off connection between the seals to permit detection of leakage through the inner seal. The connection is piped to a local pressure indicator and pressure switch. High pressure in this leak-off connection is annunciated in the control room, alerting the operator to failure of the inner seal. This system is shown in drawing M-41.

5.2.5.2.1.7 Reactor Recirculation Pump Seal Leak Detection

Each of the reactor recirculation pumps is provided with a system for monitoring leakage through each of its two mechanical seals. Two types of monitoring are provided for each pump.

Instrumentation is provided to monitor the pressure within each of the pump seal cavities. Changes in seal cavity pressure from that normally expected alerts the operator to possible reactor recirculation pump seal damage or excessive wear.

Instrumentation is also provided to monitor the flow rate in the drain line from each seal. A limited amount of leakage is expected past each of the mechanical seals. Flow instrumentation in each seal drain line actuates an annunciator in the control room wherever the leakage flow in the line becomes excessive. The instrumentation is shown in drawing M-43.

5.2.5.2.1.8 MSRV Leak Detection

Temperature sensors connected to a multipoint recorder are provided to detect MSRV leakage during reactor operation. Using a thermo-well, temperature elements are mounted, in the MSRV discharge piping downstream of the valve body. Normally, all MSRVs are in the shut-tight condition and remain at approximately the same temperature. Steam leakage past an MSRV elevates the sensed temperature in its discharge line. High discharge line temperature is annunciated in the control room, alerting the operator to MSRV leakage. The temperature sensors are shown in drawing M-41.

5.2.5.2.1.9 RPV Water Level

Under conditions of normal reactor operation at constant power, reactor water level should remain fairly constant at its programmed level since the rate of steam mass flow leaving the boiler is matched by the feedwater mass flow rate into the reactor vessel. However, given a condition of continued steam leakage from the closed system, the CST level and the reactor water level decreases.

Reactor water level is monitored by the level indicating switches associated with the containment isolation system in addition to the normal complement of process monitoring instruments. Reactor water level falling below the predetermined minimum allowable level results in switch actuation and subsequent containment isolation. Section 7.3 gives a complete description of this isolation function. The level instrumentation is shown in drawing M-42.

5.2.5.2.2 Detection of Abnormal Leakage Outside the Primary Containment

The various methods utilized for detecting leakage from components outside the primary containment are discussed below.

5.2.5.2.2.1 Main Steam Line Leak Detection Outside Primary Containment

The main steam lines are continuously monitored for leaks by the main steam line leak detection system. Steam line leaks will cause changes in at least one of the following monitored operating parameters: air temperature adjacent to the main steam lines, main steam line flow rate, or water level in the reactor vessel. If a leak is detected, the detection system actuates an annunciator in the control room and, depending upon the activating parameter, initiates steam line isolation action. The following methods are used for monitoring the above operating parameters:

a. Air temperature of the space adjacent to the main steam lines (both inside the steam tunnel and from the steam tunnel exit to the inlet of the main turbine) is monitored by a set of temperature sensors located along the length of each main steam line (drawing M-25). The temperature sensors are located or shielded so as to be sensitive to air temperature only and not to the radiated heat from hot equipment. These temperature sensors are connected to nonindicating temperature switches which initiate an alarm in the control room on abnormally high temperature. In order to prevent excessive loss of reactor coolant and the release of significant amounts of radioactive material from the RCPB, these temperature switches initiate isolation of the following lines on abnormally high temperature in the area adjacent to the main steam lines:

1. All 4 main steam lines

2. Main steam line drain

In addition to the above temperature sensors and switches used for detecting gross leakage, temperature sensors are also provided to detect possible small leaks in the main steam lines by monitoring (1) the ambient temperature within the main steam line tunnel, and (2) the temperature difference between the supply and return ventilation air for the main steam line tunnel. These sensors are also located or shielded so as to be sensitive to air temperature only and not to radiated heat from hot equipment. A rise in either the ambient temperature or the ventilation air differential temperature above the values normally indicated is alarmed in the control room, alerting the operator to possible main steam line leakage.

The heat balance for the area under consideration for the temperature sensor location is established for normal load, and that area is evaluated for various leak rates. The alarm and isolation setpoints are computed and checked by this method. The alarms and isolation setpoints associated with the above temperature devices are selected to be high enough to avoid spurious alarms and isolations, yet low enough to provide timely detection and isolation of a main steam line break.

b. Main steam flow rate within each of the main steam lines is monitored by the main steam flow restrictors and their associated differential pressure transducers and indicating switches. The outputs of the differential pressure indicating switches initiate an alarm in the control room and isolation of the following lines whenever the flow in a single steam line exceeds 140% of the line rated flow:

   1. All 4 main steam lines

   2. Main steam line drain

   The main steam line high flow alarm and isolation setpoint is set to be high enough to permit the isolation of one main steam line for test at rated power without causing an automatic isolation of the rest of the steam lines, yet low enough to provide timely detection and isolation of a gross main steam line break.

c. Reactor water level monitoring is described in Section 5.2.5.2.1.8.

5.2.5.2.2.2 RWCU System Leak Detection

Leakage in the high temperature process flow of the RWCU system external to the primary containment is detected by temperature sensors located within the RWCU equipment compartments and in the supply and return ventilation air ducts for the compartments (drawing M-25). The temperature sensors are located or shielded so as to be sensitive to air temperature only and not to the radiated heat from hot equipment. These temperature sensors are connected to nonindicating temperature switches which initiate an alarm in the control room whenever the ambient temperature or the ventilation air differential temperature of an RWCU system equipment compartment exceeds the value normally indicated. In order to prevent excessive loss of reactor coolant and the release of significant amounts of radioactive material from the RCPB, these temperature switches also initiate isolation of the RWCU system. The alarm and isolation setpoints associated with the above temperature devices are selected to be high enough to avoid spurious isolation, yet low enough to provide timely detection and isolation of a break in the RWCU system.

In addition to the temperature detection method above, leakage from the RWCU system is also detected by means of a RWCU system inlet and outlet flow comparison. If the inlet flow exceeds the outlet flow by at least 54.9 gpm for a minimum of 45 seconds, an alarm is actuated in the control room and the RWCU system is automatically isolated.

5.2.5.2.2.3 RHR System Leak Detection

RHR system leakage external to the primary containment is detected by temperature sensors in the RHR compartments (drawing M-25).

The temperature sensors are located or shielded so as to be sensitive to air temperature only and not to the radiated heat from hot equipment. These temperature sensors are connected to nonindicating temperature switches which initiate an alarm in the control room whenever the ambient temperature or the ventilation air differential temperature of a RHR system compartment exceeds the value normally expected.

5.2.5.2.2.4 RCIC and HPCI Systems Leak Detection

Leak detection components are provided for monitoring, annunciation, and, in certain cases, isolation of steam supply line leakage in the RCIC and HPCI equipment compartments and pipe-ways. Steam line leaks cause changes in air temperature adjacent to the steam lines or steam line flow rate (drawing M-25). If a leak is detected, the detection system actuates an annunciator in the control room and, depending upon the activating parameter, initiates steam line isolation action. The following methods are used for monitoring the above operating parameters:

a. As are provided for the RWCU and RHR equipment compartments, temperature sensors are located in each of the RCIC and HPCI equipment compartments and in the supply and return ventilation air ducts for the compartments. The temperature sensors are located or shielded so as to be sensitive to air temperature only and not to the radiated heat from hot equipment. These temperature sensors are connected to nonindicating temperature switches which initiate alarms in the control room whenever the ambient temperatures or the ventilation air differential temperatures of the compartments exceed the values normally indicated. In order to prevent excessive loss of reactor coolant and the release of significant amounts of radioactive material from the RCPB, the temperature switches associated with ventilation air differential temperatures of the RCIC and HPCI compartments initiate isolation of the corresponding RCIC or HPCI steam lines.

Temperature sensors are also located along the length of the RCIC and HPCI steam lines between the primary containment and the respective equipment compartments and in the equipment compartments for monitoring air temperature adjacent to the steam lines. The temperature sensors are located or shielded so as to be sensitive to air temperature only and not to the radiated heat from hot equipment. These temperature sensors are connected to nonindicating temperature switches which initiate an alarm in the control room on abnormally high temperature. In order to prevent excessive loss of reactor coolant and the release of significant amounts of radioactive material from the RCPB, the switches also initiate isolation of the corresponding RCIC or HPCI steam lines.

The alarms and isolation setpoints associated with the above temperature devices in the RCIC and HPCI equipment compartments and pipe-ways are selected to be high enough to avoid spurious isolation, yet low enough to provide timely detection and isolation of a RCIC or HPCI steam line break.

b. Steam flow rates in the RCIC and HPCI steam supply lines are monitored by venturi-type flow elements and their associated differential pressure transducers and indicating switches. The outputs of the differential pressure indicating switches initiate an alarm in the control room and isolation of the corresponding steam supply line whenever the flow in a single steam line exceeds 300% of the line rated flow.

   The RCIC and HPCI steam supply line high flow alarm and isolation setpoints are selected to provide timely detection and isolation of an RCIC or HPCI steam supply line break. Spurious isolations due to short-term flow peaks are precluded by a predetermined time delay feature.

5.2.5.2.2.5 <u>Other Leakage Detection Methods</u>

Instrumentation associated with the plant drainage system (Section 9.3.3) can provide indication of leakage of systems outside containment.

The RHR, CS, HPCI, and RCIC compartments are equipped with level switches to sense and alarm in the main control room if excessive leakage or flooding occurs in the compartment.

Excessive leakage outside the containment can also be detected by observation of abnormal reactor enclosure sump pump operating times and by sump high level alarms. This instrumentation is provided in the radwaste control room.

5.2.5.2.2.6 <u>Intersystem Leak Detection</u>

Radiation monitors are used to detect reactor coolant leakage into the RHRSW system, ESW system, and the RECW system from the RHR heat exchangers, ESW cooled compartment and the RWCU nonregenerative heat exchangers, respectively. These radiation monitoring channels are part of the PRMS (drawing M-26), and monitor for leakage into the cooling water flows downstream of the RHR heat exchangers and the RWCU system nonregenerative heat exchangers and the ESW return flow.

High radiation levels in the monitored cooling water systems is annunciated in the control room. Associated trips and interlocks for the RHRSW system pumps and the design basis and the associated instrumentation for PRMS are discussed in Section 11.5.

5.2.5.3 Indication in Control Room

Details of the leakage detection system indications are included in Section 7.6.

5.2.5.4 Limits for Reactor Coolant Leakage Inside the Containment

The total leakage inside containment consists of all leakage, both identified and unidentified, which flows to the drywell floor and equipment drain sumps. The total leakage rate limit is well within the makeup capability of the RCIC system. This limit is 30 gpm of which a limit of 5 gpm is set for unidentified leakage, and 25 gpm is allowed for identified sources.

The unidentified leakage rate limit must be low because of the possibility that most of the unidentified leakage might emanate from a single crack in the nuclear system process barrier that could be large enough to propagate rapidly.

The unidentified leakage rate limit is established at 5 gpm to allow time for corrective action before the process barrier could be significantly compromised. This 5 gpm unidentified leakage rate is adequate since it is a small fraction of the calculated flow from a critical crack in a primary system pipe as discussed in Section 5.2.5.6. Safety limits and safety limit settings are discussed in Chapter 16.

5.2.5.5 Sensitivity and Response Time

Sensitivity, including sensitivity tests, and response times of the leak detection system components are discussed in Section 7.6. As indicated therein, the sensitivity and response time of the sump leak detection system is adequate to detect an increase in unidentified leakage inside the containment of 1 gpm in less than 1 hour. The airborne radiation monitors are capable of detecting a 1 gpm coolant leak in 1 hour in a containment atmosphere free from airborne radioactivity (zero background). However, during normal plant operation (i.e. operation within technical specification limits), the normal levels of containment airborne activity coupled with normal changes in coolant activity concentrations may mask an increase in coolant leakage to the extent that its detection, by this method alone, is not possible within 1 hour.

5.2.5.6 Crack Length and Through-Wall Flow

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

a. Critical Crack Length

Both the GE and BMI test results indicate that theoretical fracture mechanics formulas do not predict critical crack length, but that satisfactory empirical expressions may be developed to fit test results. A simple equation which fits the data in the range of normal design stresses (for carbon steel pipe) is:

$$L_c = \frac{15000\,D}{h} \qquad \text{(EQ. 5.2-1)}$$

where:

L_c = critical crack length (in)

D = mean pipe diameter (in)

h = nominal hoop stress (psi)

Data correlation for Equation (EQ. 5.2-1) is shown in Figure 5.2-13.

b. Crack Opening Displacement

The theory of elasticity predicts a crack opening displacement of:

$$W = \frac{2\,L\sigma}{E} \qquad \text{(EQ. 5.2-2)}$$

where:

W = crack opening displacement (in)

L = crack length (in)

σ = applied nominal stress (psi)

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 nominal stress (σ) approaches the failure stress (σ_f). A suitable correction factor for plasticity effects is:

$$C = \sec\left(\frac{\pi\sigma}{2\sigma_f}\right) \qquad \text{(EQ. 5.2-3)}$$

The crack opening area is given by,

$$A = \frac{C\pi WL}{4} = \frac{\pi L^2\sigma}{2E}\sec\left(\frac{\pi\sigma}{2\sigma_f}\right) \qquad \text{(EQ. 5.2-4)}$$

For a given crack length L:

$$\sigma_f = 15000\ D/L$$

c. 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 in^2 (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). Results are tabulated as follows:

Nominal Pipe Size (Sch 80) (in)	Avg. Wall Thickness (in)	5 gpm Crack Length (in)		Critical Crack Length (in)	
		Steam Line	Wtr Line	Steam Line	Wtr Line
4	0.337	7.2	4.9	9.7	9.6
12	0.687	8.5	4.8	19.7	19.8
24	1.218	8.6	4.6	34.8	34.8

The ratios of crack length (L) to the critical length (L_c) as a function of nominal pipe size are:

Nominal Pipe Size (Sch 80) (in)	Ratio/L/L 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 gallons per minute will precede crack instability. Measured crack opening displacements for the BMI experiments were in the range of 0.1-0.2 inches at the time of incipient rupture, corresponding to leaks of the order of 1 in^2 in size for plain carbon steel piping. For austenitic stainless steel piping, ever larger leaks are expected to precede crack instability, although there is 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. A good mathematical model which is supported by test data is not available for the circumferential crack. Therefore, 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 5.2-12 shows general relationships between crack length, leak rate, stress, and line size, using the mathematical model described above. The asterisks denote conditions at which the crack opening displacement is 0.1 inch, at which time instability is imminent. 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 greater than the 5 gpm unidentified leakage rate limit.

5.2.5.7 Margins of Safety

The margins of safety for a detectable flaw to reach critical size are presented in Section 5.2.5.6. Figure 5.2-12 shows general relationships between crack length, leak rate, stress, and line size using the mathematical model presented in Section 5.2.5.6.

5.2.5.8 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 the reactor system process barrier located in the primary containment and reactor enclosure (Table 5.2-7). The instrumentation can be set to provide alarms at established leakage rate limits and isolate an affected system when necessary. The alarm setpoints are determined

analytically or, where appropriate, based on measurements of appropriate parameters made during startup and preoperational tests.

The primary containment unidentified leakage rate limit of 5 gpm 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.

5.2.5.9 Differentiation Between Identified and Unidentified Leakage

Section 5.2.5.2 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 Section 5.2.5.2.

5.2.5.10 Sensitivity and Operability Tests

Testability of the leak detection system is discussed in Section 7.6.

5.2.5.11 Testing and Calibration

The leak detection system is preoperationally tested in accordance with the requirements of Chapter 14 and is periodically tested in accordance with the requirements of Chapter 16.

5.2.6 REFERENCES

5.2-1 R.Linford, "Analytical Methods of Plant Transient Evaluations 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," NEDO-10899, (June 1973).

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

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

5.2-5 "Investigation and Evaluation of Cracking in Austenitic Stainless Steel Piping of Boiling Water Reactor Plants," NUREG 75-067, (October 1975).

5.2-6 "Behavior of Iodine in Reactor Water during Plant Shutdown and Startup," NEDO-10585, (August 1972).

5.2-7 "Qualification of the One-dimensional Core Transient Model for BWRs", NEDO-24154, (October 1978)

5.2-8 Document 8031-M246B-19, "Preservice Inspection Program Plan for the LGS Unit 1 Reactor Pressure Vessel"

5.2-9 Document 8031-M246AQA-59, "Preservice Inspection Program Plan for the LGS Unit 1 Nuclear Piping Systems"

5.2-10 Program Document ML-008 Limerick Generating Station Units 1 and 2, “IST Program Plan” and “IST Basis Document”

5.2-11 ER-LG-330-1001 through 1006, Limerick ISI Program Documents

5.2-12 Deleted

5.2-13 Document 8031-M246B-129, "LGS Units 1 and 2 Reactor Pressure Vessel Examination Plan for Inservice Inspection"

5.2-14 DELETE (Document 8031-P-502, "LGS Unit 1 and Common Plant First 10 year Interval Inservice Inspection Examination Plan")

5.2-15 Document 8031-P-504, "LGS Unit 2 Preservice Inspection Program"

5.2-16 Document 8031-M-246B-227, "LGS Unit 2 Reactor Pressure Vessel Preservice Inspection Examination Plan"

5.2-17 Document 8031-P-505, "LGS Unit 2 Preservice Inspection Examination Plan for Nuclear Piping Systems"

5.2-18 Document 8031-P-507, "LGS Unit 2 (Baseline) Testing Plan for Safety-Related Pumps and Valves"

5.2-19 Letter from PECo to the NRC, "Response to NRC Generic Letter 88-01, 'NRC Position on IGSCC in BWR Austenitic Stainless Steel Piping,' for Limerick Generating Station," dated August 2, 1988.

5.2-20 Letter from PECo to the NRC, "Limerick Generating Station, Units 1 and 2, Revised Response to Generic Letter 88-01, 'NRC Position on IGSCC in BWR Austenitic Stainless Steel Piping,'" dated April 28, 1989.

5.2-21 Letter from PECo to the NRC, "Limerick Generating Station, Units 1 and 2, Revised Response to Generic Letter 88-01, 'NRC Position on IGSCC in BWR Austenitic Stainless Steel Piping,'" dated May, 1989.

5.2-22 Letter from the NRC to PECo, "Request for Additional Information on Response to Generic Letter 88-01 on IGSCC in Materials (TAC Nos. 69143 and 69144)," dated July 10, 1989.

5.2-23 Letter from PECo to the NRC, "Limerick Generating Station, Units 1 and 2, NRC Request for Additional Information on Generic Letter 88-01, 'NRC Position on IGSCC in BWR Austenitic Stainless Steel Piping,'" dated September 11, 1989.

5.2-24 Letter from the NRC to PECo, "Generic Letter 88-01 (TAC Nos. 69143 and 69144)," dated March 6, 1990.

5.2-25 Letter from PECo to the NRC, "Limerick Generating Station, Units 1 and 2, Response to NRC Letter Dated March 6, 1990, Results of NRC Review of Submittals Responding to NRC Generic Letter 88-01, 'NRC Position on IGSCC in BWR Austenitic Stainless Steel Piping,'" dated June 8, 1990.

5.2-26 Letter from the NRC to PECo, "Generic Letter 88-01 (TAC Nos. 69143 and 69144)," dated October 22, 1990.

5.2-27 Letter from PECo to the NRC, "Limerick Generating Station, Units 1 and 2, Technical Specifications Change Request," dated December 21, 1990.

5.2-28 Letter from the NRC to PECo, "Generic Letter 88-01, Limerick Generating Station, Units 1 and 2 (TAC Nos. 77293 and 79294)," dated March 5, 1991.

5.2-29 Power Rerate Safety Analysis Report for Limerick Generating Station Units 1 and 2, NEDC-22225P, GE, September 1993.

5.2-30 Deleted

5.2-31 Deleted

5.2-32 “Limerick Generating Station, Units 1 and 2, SRV Setpoint Tolerance Relaxation Licensing Report,” NEDC-32645P (December 1998).

5.2-33 Letter from NRC to Exelon, "Limerick Generating Station, Units 1 and 2 - Issuance of Amendment RE: Generic Letter 88-01 Requirements (TAC Nos. MB9884 and MB9885)," dated April 20, 2004, LAR No. 171 (Unit 1) and 133 (Unit 2).

5.2-34 “Migration to TRACG04 / PANAC11 from TRACG02 / PANAC10 for TRACG AOO and ATWS Overpressure Transients,” NEDE-32906P Supplement 3-A, Revision 1, April 2010

Table 5.2-1

APPLICABLE CODE CASES FOR REACTOR COOLANT
PRESSURE BOUNDARY COMPONENTS

1332-5	Requirements for Steel Forgings, Section III and VIII
1361-1	Socket Welds, Section III
1441-1	Waiving of 3 S_m Limit for Section III Construction
1464	Requirements for Stamping, Section III
1492	Postweld Heat Treatment Sections I, III, and VIII
1516-2	Welding of Seats in Valves for Section III Applications

Table 5.2-2

NUCLEAR SYSTEM SAFETY/RELIEF VALVE SETPOINTS

NO. OF VALVES[1]	SPRING SET PRESSURE (psig)	ASME RATED CAPACITY AT 103% REFERENCE PRESSURE OF 1090 psig (lb/hr each)
4	1170	870,000
5	1180	870,000
5	1190	870,000

[1] Five of the SRVs serve in the automatic depressurization function.

Table 5.2-3

REACTOR COOLANT PRESSURE BOUNDARY MATERIALS

COMPONENT	FORM	MATERIAL	SPECIFICATION (ASTM/ASME)
12" 900# TESTABLE CHECK VALVE (E21-F006/E11-F041)			
Body	Cast	Steel	SA352 LCB
Disc	Cast	Steel	SA352 LCB
Cover	Forged	Steel	SA350 LF1
Bearing cover	Forged	Steel	SA350 LF1
Stud	Bar	Alloy steel	SA540 B23 Class 5
Nut	Bar	Steel	SA540 B23 Class 5
MAIN STEAM FLOW ELEMENT B21-N051 thru N054			
Inst nozzle	Forging	Carbon steel	SA105
Pipe spool	Pipe	Carbon steel	SA105 Grade B
SLCS VALVE HV-048-F006A/F006B			
Body	Bar	304 SST	QQ-S-763 (Cond A)
Fitting	Bar	304 SST	QQ-S-763 (Cond A)
Trigger body	Bar	304 SST	QQ-S-763 (Cond A)
Ram	Bar	17-4PH Cond H900	
Q-ring	Liquid	ARP 568-116	

Table 5.2-3 (Cont'd)

COMPONENT	FORM	MATERIAL	SPECIFICATION (ASTM/ASME)
RCIC PUMP E51-C001			
Barrel, outer	Forged	Carbon steel	ASTM, A105 Grade II
Nozzle, disch	Forged	Carbon steel	ASTM, A105 Grade II
Nozzle, suction	Forged	Carbon steel	ASTM, A105 Grade II
Elbow, suction	Forged	Carbon steel	ASTM, A105 Grade II
Connection branches, vents & drains	Forged	Carbon steel	ASTM, A105 Grade II
Flange, bypass	Forged	Carbon steel	ASTM, A105 Grade I
Cover, drive end	Forged	Carbon steel	ASTM, A105 Grade II
Cover, thrust end	Forged	Carbon steel	ASTM, A105 Grade II
Pipe, bypass	Cold drawn	Carbon steel	ASTM, A106 Grade B
Bolting	Rolled	Alloy steel	ASTM, A193-B7
Nuts	Rolled	Alloy steel	ASTM, A194-2H
Piping-seal	Cold drawn	Carbon steel	ASTM, A106 Grade B
Circulation union pipe 6000#	Forged	Carbon steel	ASTM, A181 Grade II
Tee, pipe 6000#	Forged	Carbon steel	ASTM, A181 Grade II
Coupling, pipe	Forged	Carbon steel	ASTM, A181 Grade II
Nipple, pipe	Cold drawn	Carbon steel	ASTM, A106 Grade B
Separator, cyclone	Cast	Carbon steel	ASTM, A216 Grade WCB
CONTROL ROD	Pipe	Austenitic stainless steel	SA312 Type 304L
DRIVE HOUSING	Welds	Stainless steel	SFA5.9 Type 308 or 316 SFA5.4 Type 308 or 316
	Forging	Stainless steel	SA182 Type 304
INCORE HOUSING	Tube	Austenitic stainless steel	SA213 Type 304
	Weld	Stainless steel	SFA5.9 Type 308 or SFA5.4
	Forging	Stainless steel	SA182 Type 304
Reactor vessel	Rolled plate	Low alloy steel	SA533 Grade B
Heads, shells	Forgings		SA508 Class 2
	Welds	Low alloy steel	SFA5.5
Closure flange	Forged ring	Low alloy steel	SA508 Class 2
	Welds	Low alloy steel	SFA5.5
Nozzles	Forged shapes	Low alloy steel	SA508 Class 2
	Welds	Low alloy steel	SFA5.5
Cladding	Weld overlay	Austenitic stainless steel	SFA5.9 or SFA5.4 Type 309 with carbon content on final

Table 5.2-3 (Cont'd)

COMPONENT	FORM	MATERIAL	SPECIFICATION (ASTM/ASME)
Control rod	Pipe	Austenitic stainless steel	SA312 Type 304
Drive housing	Forgings	Stainless steel	SA182 Type 304
	Welds	Stainless steel	SFA5.9 or SFA5.4 Type 308
Incore	Tube	Austenitic stainless steel	SA213 Type 304
Housings	Welds	Stainless steel	SFA5.9 or 5.4 Type 308
MAIN STEAM ISOLATION VALVE B21-F022/F028			
Valve body	Cast	Carbon steel	ASTM A216 WCB
Cover	Forged steel	Steel	ASTM A105 Grade II
Poppet	Forged	Steel	ASTM A105 Grade II
Valve stem	Bar	Stainless steel	ASME/ASTM SA/A564 Grade 630
Body bolts	Bar	Steel	ASTM A193 Grade B7
Hex nuts	Bar	Steel	ASTM A194 Grade 2H
MAIN STEAM SAFETY RELIEF VALVE (B21-F013)			
Main Valve Body	Forging	Carbon Steel	ASME SA105
Base	Forging	Inconel 600	ASME SB564
Disc	Forging	Stainless Steel	ASME SA182
Seat	Bar or Forging	Carbon Steel	ASME SA105 OR SA696, Grade C
Studs	Bar	Alloy Steel	ASME SA193, Grade B7 and Grade B16
Nuts	Bar	Alloy Steel	ASME SA194, Grade 7
Pilot body	Bar	Inconel 600	ASME SB166, Grade 600
RECIRCULATION PUMP B32-C001			
Pump case	Cast	Stainless steel	SA351 Grade CF8M
Pump cover	Cast	Stainless steel	SA351 Grade CF8M
Seal flange	Forging	Stainless steel	SA351 Grade CF8M
Studs cover/case	Bar	Alloy steel	SA540 Grade B23 Class 4
Cap screw seal flg	Bar	Alloy steel	SA540 Grade B23 Class 4
Drive mount bottom flg	Forging	Carbon steel	A216 WCB
Coil heat exch	Tube	Stainless steel	SA213 Type 304 SMLS.
Fitting heat exch	Forging	Stainless steel	SA351 Grade CF8
	Forging	Stainless steel	SA479 Type 304
CONTROL ROD DRIVE	Forging	Austenitic stainless steel	ASME SA182 Type 304
	Pipe	Austenitic stainless steel	ASME SA312 Type 316

Table 5.2-3 (Cont'd)

COMPONENT	FORM	MATERIAL	SPECIFICATION (ASTM/ASME)
12" TESTABLE CHECK VALVE (E11-F050)			
Body	Cast	Stainless steel	SA351 CF8M
Disc	Cast	Stainless steel	SA351 CF8M
Cover	Forging	Stainless steel	SA182 F316
Bearing cover	Forging	Stainless steel	SA182 F316
Studs	Bar	Alloy steel	SA540 B23 Class 5
Hex nuts	Bar	Steel	SA540 B23 Class 5
12" 900# VALVE (E11-F015)			
Body	Cast	Stainless steel	SA351 CF8M
Bonnet	Cast	Stainless steel	SA351 CF8M
Disc	Cast	Stainless steel	SA351 CF8M
Stud	Bar	Alloy steel	SA193 B7
Nut	Bar	Carbon steel	SA194 2H
12" 900# GATE VALVE (E21-F005)			
Body	Cast	Carbon steel	SA352 LCB
Bonnet	Cast	Carbon steel	SA352 LCB
Disc	Cast	Carbon steel	SA352 LCB
Stud	Bar	Alloy steel	SA193 B7
Nut	Bar	Steel	SA194 2H
RECIRCULATION GATE VALVE B32-F023/F031			
Body	Cast	Stainless steel	SA351 Grade CF8M
Bonnet	Cast	Stainless steel	SA351 Grade CF8M
Disc	Cast	Stainless steel	SA351 Grade CF3A
Stud	Bar	Alloy steel	A193 Grade B7
Nut	Bar	Steel	ASTM 194 Grade 2H

Table 5.2-3 (Cont'd)

COMPONENT	FORM	MATERIAL	SPECIFICATION (ASTM/ASME)

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 used in the piping system:

Pipe	Carbon steel	SA106 Grade B, SA333 Grade 6, SA155 KCF70
Fittings	Carbon steel	SA105 Grade II, SA350 Grade LF1, SA234 Grade B, SA420 Grade WPL1
Bolting	Carbon steel	SA193 Grade B7, SA194 Grades 7 and 2H

For those systems or portions of systems, except for the reactor recirculation system, that require austenitic stainless steel, the following materials and specifications are used in the piping system:

Pipe	Austenitic stainless steel	SA376 Type 304, SA312 Type 304, SA358 Type 304
Flanges	Austenitic stainless steel	SA182 Grade F316
Bolting	Austenitic stainless steel	SA193 Grade B7, SA194 Grades 7 and 2H
Fittings	Austenitic stainless steel	SA182 Grade F304, SA403 Grades WP304, 304W

For the recirculation system (inside containment) the following materials and specifications are used in the piping system:

Pipe	Austenitic stainless steel	B50 YP166 Type A1
Fittings	Austenitic stainless steel	B50 YP193 Type B2

Table 5.2-4

NOMINAL BWR WATER CHEMISTRY FOR NORMAL OPERATION

	CONCENTRATIONS - PARTS PER BILLION (ppb)				CONDUCTIVITY	
	Iron	Copper	Chloride	Oxygen	(μmho/cm @ 25°C)	pH @ 25°C
1. Condensate (1)[1]	15-30	3-5	≤20	20-50	~0.1	~7
2. Condensate cleanup effluent (2)[1]	<5.0	<1	~0.2	20-50	<0.1	~7
3. Feedwater (3)[1]	<5.0	<1	~0.2	20-50	<0.1	~7
4. Reactor water (4)[1]						
(a) Normal operation	10-50	<20	<20	100-300 See note [2]	0.2-0.5	~7
(b) Shutdown	-	-	<20	See note [2]	<1	~7
(c) Hot standby	--	<20	See note [2]	<1	~7	
(d) Depressurized	-	-	<20	8000	<2	6-6.5
5. Steam (5)[1]	0	0	0	10000-30000	~0.1	-
6. CRD cooling water (6)[1]	50-500	-	<20	≤50	≤0.1	~7

[1] The numerals in parentheses refer to locations delineated in Figure 5.2-9.

[2] See Section 5.2.3.2.2.

Table 5.2-5

SYSTEMS THAT MAY INITIATE DURING OVERPRESSURE EVENT

SYSTEM	INITIATING/TRIP SIGNALS(S)[1]
RPS	Reactor trips "Off" on high flux
RCIC	"On" when reactor water level at L2 "Off" when reactor water level at L8
HPCI	"On" when reactor water level at L2 "Off" when reactor water level at L8
Recirculation system	"Off" when reactor water level at L2 "Off" when reactor pressure at 1125 psig
RWCU	"Off" when reactor water level at L2

[1] Vessel level trip settings are shown in Figure 5.3-2.

Table 5.2-6

WATER SAMPLE LOCATIONS

SAMPLE ORIGIN	SENSOR LOCATION	INDICATOR LOCATION	RECORDER LOCATION
Reactor water recirculation loop	Sample Line	Sample Station	_
RWCU System inlet	Sample Line	Sample Station	Control room
RWCU system outlets	Sample Line	Sample Station	Control room
Stored condensate CRD System	Sample Line	Sample Station	Control room
RHR heat exchanger Outlets	Sample Line	Sample Station	_
Main steam from reactor	Sample Line	Sample Station	_
Feedwater to reactor	Sample Line	Sample Station	_

Table 5.2-7

LEAK DETECTION METHODS USED AND SYSTEM ALARMS AND INTERLOCKS PROVIDED

SOURCE OF LEAKAGE		VARIABLE MONITORED[5]												
		A	A	A	A/I	A/I	A/I	A/I	A/I	A/I	A	A/I	A	A
SYSTEM[1]	LOCATION[5]	PC HIGH TEMPERATURE	PC HIGH PRESSURE	PC SUMP HIGH FLOW RATE	EQUIPMENT AREA HIGH T & ΔT	RWCU HIGH FLOW	PIPE ROUTING AREA HIGH T & ΔT	HIGH FLOW RATE[4]	RWCU HIGH ΔFLOW	REACTOR LOW WATER LEVEL	HIGH FISSION PRODUCT RADIOACTIVITY	RE FLOOR DRAIN SUMP MONITORING[6]	COMPARTMENT WATER LEVEL[6]	PC AIR COOLER CONDENSATE DRAIN FLOW
MAIN STEAM	PC RE	X 	X 	X 	 X[2]		 X[2]	X X		X X	X 	 X		X
RHR	PC RE	X 	X 	X 	 X[2]					X X	X 	 X	 X	X
RCIC STEAM	PC RE	X 	X 	X 	 X		 X	X X		X X	X 	 X		X
RCIC WATER	PC RE											 X	 X	
HPCI STEAM	PC RE	X 	X 	X 	 X		 X	X X		X X	X 	 X		X
HPCI WATER	PC RE											 X	 X	
RWCU WATER	PC RE RE	X Hot Cold	X 	X 	 X 	X X X			X X X	X X X	X 	 X X		X
FEEDWATER	PC RE	X 	X 	X 							X 	 X		X
CORE SPRAY	PC RE	X 	X 	X 								 X	 X	X

Table 5.2-7 (Cont'd)

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(1) All systems within the drywell share a common leak detection system.

(2) Isolation occurs on high temperature in the main steam tunnel and in the vicinity of the main steam lines within the turbine enclosure.

(3) Alarm only.

(4) Leakage downstream of the flow element will cause isolation of the steam line.

(5) The following notations are used:

A - alarm
I - isolation
PC - primary containment
RE - reactor enclosure

(6) Plant drainage system instrumentation.

Table 5.2-8

SEQUENCE OF EVENTS

(REDY, ODYN, and TRACG)

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Table 5.2-8 (Cont'd)

SEQUENCE OF EVENTS

(3 SRV Out-Of-Service)*

TIME-SEC	EVENT
0	Initiate closure of all MSIVs.
0.3	MSIVs reach 90% open and initiate reactor scram. However, hypothetical failure of this position scram is assumed in this analysis.
1.64	Neutron flux reaches the high APRM flux scram setpoint and initiates reactor scram.
2.35	Recirculation pump drive motors trip due to high vessel pressure.
2.75	Steam line pressure reaches the relief valve pressure setpoint and valves start to open.
3.01	All SRVs opened.
3.0	The MSIVs completely closed.
4.41	Vessel bottom pressure at its peak value.

* General Electric performed a re-analysis to increase the allowable SRV as-found setpoint tolerance to ± 3% of the SRV setpoint (ref. 3.9-26). The results of this analysis require that the number of SRVs allowed out-of-service, be limited to 2. An assessment of the data given in Table 5.2-8 indicates that the effect from the re-analysis on the time sequence is not significant. The change to ± 3% of the SRV setpoint tolerance is effective after 2R05 for Unit 2, and after 1R08 for Unit 1.

Table 5.2-9

RCPB OPERATING THERMAL CYCLES

NORMAL, UPSET, AND TESTING CONDITIONS	NUMBER OF CYCLES
1. Bolt-up(1)	123
2. Design hydrostatic test	130
3. Startup (100°F/hr heatup rate)(2)	117
4. Daily reduction to 75% power(1)	10,000
5. Weekly reduction to 50% power(1)	2,000
6. Control rod pattern change(1)	400
7. Loss of feedwater heaters (80 cycles total)	80
8. OBE event	1(4)
9. Scram:	
a. Turbine-generator trip, feedwater on, isolation valves stay open	40
b. Other scrams	140
10. Reduction to 0% power, hot standby, shutdown (100°F/hr cooldown rate)(2)	111
11. Unbolt	123
12. Natural circulation startup	3
13. Loss of AC power natural circulation restart	5
EMERGENCY CONDITIONS	
14. Scram:	
a. Reactor overpressure with delay scram, feedwater stays on, isolation valves stay open	1(3)

Table 5.2-9 (Cont'd)

	NUMBER OF CYCLES
b. Loss of feedwater pumps, isolation valves closed	5
c. Automatic blowdown	1[3]
d. Turbine trip with single SRV blowdown	8
15. Improper start of cold recirculation loop	1[3]
16. Sudden start of pump in cold recirculation loop	1[3]
17. Improper startup with reactor drain shutoff followed by turbine roll and increase to rated power	1[3]
FAULTED CONDITION	
18. SSE (at rated operating conditions)	1[3]
19. Pipe rupture and blowdown	1[3]

(1) Applies to RPV only

(2) Bulk average vessel coolant temperature change in any 1 hour period, excluding flooding

(3) The 40 year encounter probability of these one-cycle events is $<10^{-1}$ for emergency and $<10^{-3}$ for faulted events

(4) Includes 10 maximum load cycles per event – 1 event = 10 cycles

Table 5.2-10

MAIN STEAM SWEEPOLET MATERIAL DATA FROM OTHER BWRs

PROJECT A

Applicable Code: ASME Section III, 1974 Edition, S74 Addendum

Vendor: Bonney Forge Division, Gulf Western Manufacturing

Material Vendor: Sharon Steel

Material Spec: SA105N

Heat No.: 631218 (Sharon Steel)

	C	Mn	Si	P	S
Chemical Composition (wt%)	0.28	0.98	0.22	0.014	0.015

Heat Treatment: Normalize 1650°F (12 hr) - Air Cooled

Charpy V-Notch Impact Toughness (Longitudinal)

Test Temperature: +70°F

Energy (ft-lb) 68.2, 83.5, 76.0

Lat. Exp. (mil) 64, 71, 69

% Shear: 80, 80, 80

PROJECT B

Applicable Code: ASME Section III, 1974 Edition, S74 Addendum

Vendor: Bonney Forge Division, Gulf Western Manufacturing

Material Vendor: Sharon Steel

Material Spec: SA105N

Heat No.: 630614 (Sharon Steel)

Table 5.2-10 (Cont'd)

	C	Mn	Si	P	S
Chemical Composition (wt%)	0.26	0.86	0.16	0.022	0.017

Heat Treatment: Normalize 1650°F (4 hr) - Air Cooled

Charpy V-Notch Impact Toughness (Longitudinal)

Test Temperature:	+70°F	
Energy (ft-lb)	76.6, 74.9, 62.0	107.7, 108.5, 109.3
Lat. Exp. (mil)	68, 69, 63	75, 84, 85
% Shear:	80, 90, 80	100, 100, 100

Table 5.2-11

MSIV BONNET COVER MATERIAL

UNIT 1

Applicable Code:	1968 ASME Nuclear Pump and Valve Code				
Valve Vendor:	Atwood & Morrill Company				
Material Vendor:	Cann & Saul Steel Company				
Material Specification:	ASTM A105, Grade 2				
Heat No.	219222				
	C	Mn	Si	P	S
Chemical Composition (wt%)	0.30	0.68	0.19	0.009	0.014
Heat Treatment:	Normalize 1650°F (12 hr) - Air Cooled				

UNIT 2

Applicable Code:	1968 ASME Nuclear Pump and Valve Code				
Valve Vendor:	Atwood & Morrill Company				
Material Vendor:	Cann & Saul Steel Company				
Material Specification:	ASTM A105, Grade 2				
Heat No.	221398/69D018				
	C	Mn	Si	P	S
Chemical Composition (wt%)					
(221398)	0.28	0.79	0.23	0.010	0.020
(69D018)	0.34	0.82	0.18	0.009	0.024
Heat Treatment:	Normalize 1650°F (12 hr) - Air Cooled				

MSIVs modified with nose guided poppet kits have bonnets made from SA105 forgings which have equivalent properties.

Table 5.2-12

MSIV BODY DATA

UNIT 1

Applicable Code: ASME Section III, W68, Draft Pump & Valve Code

Valve Vendor: Atwood & Morrill Company

Material Vendor: Quaker Alloy Casting Company

Material Spec: ASTM A216 Grade WCB

Heat Number: F8304-1

	C	Mn	Si	P	S
Chemical Composition (wt%):	0.26	0.90	0.30	0.019	0.012

Heat Treatment: Normalize 1700°F (7 hr 10 min) air cool

+Temperature 1340°F (7 hr) air cool

+Postweld heat treatment/stress relieve
140°F/1170°F (5 hr 10 min) air cool

UNIT 2

Applicable Code: ASME Section III, W68, Draft Pump & Valve Code

Valve Vendor: Atwood & Morrill Company

Material Vendor: Quaker Alloy Casting Company

Material Spec: ASTM A216 Grade WCB

Heat Number: F3034

	C	Mn	Si	P	S
Chemical Composition (wt%):	0.25	0.63	0.38	0.016	0.014

Heat Treatment: Normalized @ 1690°F - 1770°F, 8 hr 5 min

+Temperature 1360°F, 6 hr 40 min.

+Postweld heat treatment/stress relieve
1100°F - 1125°F, 6 hr 45 min.

Table 5.2-13

MSIV BODY DATA FROM OTHER BWRs

PROJECT A

Applicable Code: ASME Section III, 1974

Valve Vendor: Atwood & Morrill Company

Material Vendor: Quaker Alloy Casting Company

Material Spec: ASME SA216 Grade WCB

Heat Number: F6406

	C	Mn	Si	P	S
Chemical Composition (wt%):	0.23	0.89	0.53	0.019	0.012

Heat Treatment: 1680°F/1710°F (5 hr, 30 min) air cool

+Temperature 1350°F (5 hr, 30 min) air cool

+Postweld 1200°F (6 hr) air cool

Charpy V-Notch Impact Toughness

Test Temperature: +60°F

Energy (ft-lb): 32, 31, 34

Exp. (mil): 33, 32, 31

% Shear: 40, 40, 40

PROJECT B

Applicable Code: ASME Section III, 1974

Valve Vendor: Atwood & Morrill Company

Material Vendor: Atwood & Morrill Company

Material Spec: ASME SA216 Grade WCB

Heat Number: 35

Table 5.2-13 (Cont'd)

	C	Mn	Si	P	S
Chemical Composition (wt%):	0.24	0.82	0.46	0.022	0.013

Heat Treatment: 1650°F/1800°F (8 hr), air cool to 400°F

+Temperature 1150°F/1250°F (8 hr), air cool

+Postweld 1095°F/1195°F (18 hr) furnace cool to 800°F (100°F/hr) air cool

Charpy V-Notch Impact Toughness

Test Temperature:	+60°F
Energy (ft-lb)	31.5, 37.5, 39.5
Exp. (mil):	33, 41, 40
% Shear:	10, 10, 10

PROJECT C

Applicable Code:	ASME Section III, 1974 with Summer 1975 Addenda
Valve Vendor:	Atwood & Morrill Company
Material Vendor:	Quaker Alloy Casting Company
Material Spec:	ASME SA216 Grade WCB
Heat Number:	F3547

	C	Mn	Si	P	S
Chemical Composition (wt%):	0.23	0.88	0.38	0.016	0.015

Heat Treatment: 1700°F/1725°F (6 hr, 20 min) air cool

+Temperature 1345°F (6 hr, 45 min) air cool

+Postweld 1200°F/1225°F (6 hr, 20 min) air cool

Table 5.2-13 (Cont'd)

Charpy V-Notch Impact Toughness

Test Temperature:	+60°F
Energy (ft-lb):	66, 56, 54
Exp. (mil):	53, 50, 53
% Shear:	40, 40, 40

PROJECT D

Applicable Code: ASME Section III, 1971 with Summer 1973 Addenda

Valve Vendor: Rockwell International

Material Vendor: Rockwell International

Material Spec: ASME SA216 Grade WCC

Heat Number: 1750262

	C	Mn	Si	P	S	Al
Chemical Composition (wt%):	0.21	1.19	0.43	0.011	0.009	0.043

Heat Treatment: 1700°F (10 hr) normalize

+1225°F (7.5 hr) temperature

+1100°F (6 hr) postweld

Charpy V-Notch Impact Toughness

Test Temperature:	+40°F
Energy (ft-lb):	29.0, 33.0 35.0
Exp. (mil):	25.0, 26.0, 30.0
% Shear:	15, 15, 15

Table 5.2-13 (Cont'd)

PROJECT E

Applicable Code: ASME Section III, 1971 with Summer 1973 Addenda

Valve Vendor: Rockwell International

Material Vendor: Rockwell International

Material Spec: ASME SA216 Grade WCC

Heat Number: 3760171

	C	Mn	Si	P	S	Al
Chemical Composition (wt%):	0.17	1.09	0.50	0.008	0.011	0.060

Heat Treatment: 1700°F (8 hr) normalize

+1275°F (8 hr) temperature

+1100°F (6 hr) postweld

Charpy V-Notch Impact Toughness

Test Temperature: +40°F

Energy (ft-lb): 35, 38, 29

Exp. (mil): 32, 36, 29

% Shear: 20, 20, 20

PROJECT F

Applicable Code: ASME Section III, 1974

Valve Vendor: Atwood & Morrill Company

Material Vendor: Quaker Alloy Casting Company

Material Spec: ASME SA216 Grade WCB

Heat Number: F7516

Table 5.2-13 (Cont'd)

	C	Mn	Si	P	S
Chemical Composition (wt%):	0.25	0.78	0.53	0.018	0.013

Heat Treatment: 1690°F/1710°F (6 hr, 5 min) air cool

+Temperature 1350°F/1360°F (6 hr) air cool

+Postweld 1200°F (6 hr, 5 min) air cool

Charpy V-Notch Impact Toughness

Test Temperature:	+60°F
Energy (ft-lb):	30, 24, 34
Exp. (mil):	37, 27, 33
% Shear:	40, 40, 40

5.3 REACTOR VESSEL

5.3.1 REACTOR VESSEL MATERIALS

5.3.1.1 Materials Specifications

The materials used in the RPV and appurtenances are shown in Table 5.2-3 together with the applicable specifications.

5.3.1.2 Special Processes Used for Manufacturing and Fabrication

The RPV is primarily constructed from low alloy, high strength steel plate and forgings. Plates are ordered to ASME SA533, Grade B, Class 1, and forgings to ASME SA508, 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 ASME SA540, Grade B23, or Grade B24. Welding electrodes are low hydrogen-type ordered to ASME SFA5.5.

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

All fabrication of the RPV 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 in accordance with ASME Sections 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. Postweld 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 requirements of ASME Section III, Paragraph N-624 including Summer 1975 Addenda. In addition, all welds are given a supplemental ultrasonic examination.

The materials, fabrication procedures, and testing methods used in the construction of BWR RPVs meet or exceed requirements of ASME Section III, Class I vessels.

5.3.1.3 Special Methods for Nondestructive Examination

The materials and welds on the RPV 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 manual techniques. The ultrasonic examination method, including calibration, instrumentation, scanning sensitivity, and coverage is based on the requirements imposed by ASME Section XI, Appendix I. Acceptance standards are equivalent to, or more restrictive than, those required by ASME 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, Control of Ferrite Content in Stainless Steel Weld Metal

Controls on stainless steel welding are discussed in Section 5.2.3.4.2.1.

5.3.1.4.1.2 Regulatory Guide 1.34, Control of Electroslag Weld Properties

Electroslag welding is not employed for the RPV fabrication.

5.3.1.4.1.3 Regulatory Guide 1.43, Control of Stainless Steel Weld Cladding of Low Alloy Steel Components

This guide applies to welding of cladding to low alloy steels made to coarse grain practice. LGS vessel plate and nozzle forgings are made to fine grain practice and a low heat input process is used. Other components are not clad. Therefore, the guide is not applicable.

5.3.1.4.1.4 Regulatory Guide 1.44, Control of the Use of Sensitized Stainless Steel

Controls to avoid severe sensitization are discussed in Section 5.2.3.4.1.1.

5.3.1.4.1.5 Regulatory Guide 1.50, Control of Preheat Temperature for Welding Low Alloy Steel

Preheat controls are discussed in Section 5.2.3.3.2.1.

5.3.1.4.1.6 Regulatory Guide 1.71, Welder Qualification for Areas of Limited Accessibility

Welder qualification for areas of limited accessibility is discussed in Section 5.2.3.4.2.3.

5.3.1.4.1.7 Regulatory Guide 1.99, Effects of Residual Elements on Predicted Radiation Damage to Reactor Pressure Vessel Materials

Predictions for changes in transition temperature and upper-shelf energy are made in accordance with the guidelines of Regulatory Guide 1.99 (Rev 2).

5.3.1.5 Fracture Toughness

This section is supplemented by Sections 5.3.1.7 and 5.3.1.8 in discussing the compliance to the intent of 10CFR50, Appendix G.

5.3.1.5.1 Assessment of 10CFR50, Appendix G

A major condition necessary for full compliance to 10CFR50, Appendix G is satisfaction of the requirements of the Summer 1972 Addenda to ASME Section III. This is not possible with components that were purchased to earlier code requirements. For the extent of compliance see Table 5.3-1.

Ferritic materials complying with 10CFR50, Appendix G must have both drop weight tests and Charpy V-notch tests with the CVN specimens oriented transverse to the maximum material working direction to establish the RT_{NDT}. The CVN tests must be evaluated against both an absorbed energy and a lateral expansion criteria. The maximum acceptable RT_{NDT} must be determined in accordance with the analytical procedures of ASME Section XI, Appendix G. 10CFR50, Appendix G, requires an initial minimum of 75 ft-lb upper-shelf CVN energy for beltline material. It also requires at least 45 ft-lb CVN energy and 25 mils lateral expansion for bolting material at the lower of the preload or lowest service temperature.

By comparison, materials for the LGS Units 1 and 2 reactor vessels are qualified by drop weight tests and/or in most cases longitudinally oriented CVN tests (both not required), confirming that the material NDTT is at least 60°F below the lowest service temperature. When the longitudinal CVN test was applied, a 30 ft-lb energy level was used in defining the NDTT. There was no upper-shelf CVN energy requirement on the LGS Units 1 and 2 beltline material. The bolting material was qualified to a 30 ft-lb CVN energy requirement at 60°F below the minimum preload temperature.

To determine operating limits in accordance with 10CFR50, Appendix G, estimates of the beltline material RT_{NDT} and the highest RT_{NDT} of all other material were made, as explained in Section 5.3.1.5.3. The method for developing these operating limits is also described therein.

5.3.1.5.2 Method of Compliance

The method of compliance is based on the last paragraph on page 19013 of the July 17, 1973 Federal Register.

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 ensure that a margin of safety against a nonductile failure of this vessel is very nearly the same as a vessel built to the Summer 1972 Addenda.

The specific temperature limits for all modes of plant operation when the core is critical are based on 10CFR50, Appendix G.

5.3.1.5.3 Methods of Obtaining Operating Limits Based on Fracture Toughness

Operating limits that define minimum metal temperatures versus reactor pressure during normal heatup and cooldown, and during inservice hydrostatic testing, are established using the methods of ASME Section XI, Appendix G, 1995 Edition in conjunction with Code Case N-640 (Reference 5.3-11 and 5.3-12). The results are shown in Figures 5.3-4 (LGS Unit 1) and 5.3-5 (LGS Unit 2).

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 were evaluated and the operating limit curves are based on the limiting location. The bolt-up limits for the flange and adjacent shell regions are based on a minimum metal temperature of RT_{NDT} + 60°F. The maximum through-wall temperature gradient from continuous heating and cooling at 100°F per hour was considered. The safety factors applied were as specified in ASME 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 the vessels are designed and manufactured. These toughness test data, CVN and/or drop-weight NDT are analyzed to permit compliance with the intent of 10CFR50, 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 CVNs, 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 drop-weight testing was permitted; therefore, in many cases both tests were not performed as is currently required. To substantiate the design adequacy, toughness property correlations are derived for the vessel materials to give a conservative estimate of RT_{NDT}, compliant with the intent of Appendix G criteria.

These toughness correlations vary, depending on the specific material analyzed, and are derived from the results of Bulletin 217, "Properties of Heavy Section Nuclear Reactor Steels," and from toughness data from the LGS Unit 1 and 2 vessels and other reactors. In the case of vessel plate material (SA533, Grade B, Class 1), the predicted limiting toughness property is either NDT or transverse CVN 50 ft-lb temperature minus 60°F. NDT values are available for all beltline and some other LGS 1 and 2 vessel plates. Where NDT results are missing, NDT is estimated as the longitudinal CVN 35 ft-lb transition temperature. The transverse CVN 50 ft-lb 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-lb transition temperature by adding 2°F per ft-lb to the test temperature. If the actual data equals or exceeds 50 ft-lb, the test temperature is used. Once the longitudinal 50 ft-lb temperature is derived, an additional 30°F is added to account for orientation effects and to estimate the transverse CVN 50 ft-lb temperature minus 60°F, estimated in the preceding manner.

For forgings (SA508, Class 2), the predicted limiting property is the same as for vessel plates. CVN and drop-weight values are available for the vessel flange, closure head flange, and feedwater and LPCI nozzle materials for LGS Units 1 and 2. RT_{NDT} is estimated in the same way as for vessel plate.

For the vessel weld metal the predicted limiting property is the CVN 50 ft-lb 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-lb 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 material the RT_{NDT} is assumed the same as for the base material, since ASME Code weld procedure qualification test requirements and postweld heat treatment indicates this assumption is valid.

Closure bolting material (SA540, Grade B24) toughness test requirements for LGS Units 1 and 2 are for 30 ft-lb at 60°F below the bolt-up temperature. Current Appendix G requirements are for 45 ft-lb and 25 MLE at the preload or lowest service temperature, including bolt-up. All LGS Units 1 and 2 closure stud materials meet current requirements at +10°F.

Using this general approach, an initial RT_{NDT} of 20°F is established for the core beltline region for LGS Unit 1 and 40°F for LGS Unit 2.

The effect of the main closure flange discontinuity is considered by adding 60°F and 90°F to the RT_{NDT} to establish the minimum temperature for bolt-up and pressurization respectively. The minimum bolt-up temperature of 80°F for LGS Unit 1, which is shown in Figure 5.3-4 is based on an initial RT_{NDT} of +20°F for the shell plate which connects to the closure flange. The minimum bolt-up temperature of +70°F for LGS Unit 2, which is shown in Figure 5.3-5, is based on an initial RT_{NDT} of +10°F for the closure flange forgings. A flange region flaw size less than 0.24 inch critical flaw depth can be detected at the outside surface of the flange to shell and head junctions where stresses due to bolt-up are most limiting.

Because the toughness testing in strict compliance with 10CFR50, Appendix G was not required at the time of vessel procurement, the effect of the reactor vessel discontinuities is considered by adjusting the results of a BWR/6 reactor discontinuity analysis to the LGS reactors. The BWR/6 analysis performed in accordance with 10CFR50, Appendix G includes the margin of safety implicit in the Appendix G requirement. The adjustment is made by increasing the minimum temperatures required by the difference between LGS and BWR/6 feedwater nozzle forging RT_{NDT}s. The adjustment is based on an RT_{NDT} of 40°F for Unit 1 and an RT_{NDT} of 40°F for Unit 2.

The reactor vessel closure studs have a minimum Charpy impact energy of 48 ft-lb and a 27 MLE at 10°F for LGS Unit 1. The studs for LGS Unit 2 have a minimum Charpy impact energy of 25 MLE and 46 ft-lb at 10°F. The lowest service temperature for bolt-up of LGS Unit 2 is 10°F. Charpy test results are discussed in Sections 5.3.1.7 and 5.3.1.8.

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 original surveillance program were selected to represent materials used in the reactor beltline region. The specimens were 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 were heat treated in a manner that simulates the actual heat treatment performed on the core region shell plates of the completed vessel.

The original surveillance program included three capsule holders per reactor vessel.

Information on the specimen arrangement is given in Table 5.3-12, referenced in Section 5.3.1.10.

A set of out-of-reactor baseline CVN specimens is provided with the surveillance test specimens.

Charpy impact specimens for the original reactor vessel surveillance programs are of the longitudinal orientation consistent with the ASME requirements prior to the issue of the 1972 Addenda and ASTM E185-73. Based on GE experience, the amount of shift measured by these irradiated longitudinal test specimens is essentially the same as the shift in an equivalent transverse specimen.

For LGS Units 1 and 2, each set of surveillance specimens is loaded in 6 small capsules rather than one large capsule. Therefore, each capsule holder which contains all 6 small capsules can be considered to be the same as one surveillance capsule as defined in 10CFR50, Appendix H. Three capsule holders are included in each reactor vessel. Since the predicted adjusted increase in reference temperature of the beltline region, estimated at the time of design, was less than 100°F at EOL and the calculated peak neutron fluence is less than $5x10^{18}$ n/cm^2, the use of three capsule holders meets the requirements of 10CFR50, Appendix H, and ASTM E185-73.

For the extent of compliance of the original surveillance program to 10CFR50, Appendix H, see Table 5.3-2.

In 2003, the NRC approved LGS participation in the BWR Vessel and Internals Project (BWRVIP) Integrated Surveillance Program (ISP) as described in BWRVIP-78 and BWRVIP-86 (Reference 5.3-13). The NRC approved the ISP for the industry in Reference 5.3-13 and approved LGS participation in Reference 5.3-14. The ISP meets the requirements of 10 CFR 50 Appendix H and provides several advantages over the original program. The surveillance materials in many plant-specific programs do not represent the best match with the limiting vessel beltline materials since some were established prior to 10 CFR 50 Appendix H requirements. Also, the ISP allows for better caparison to unirradiated material data to determine actual shifts in toughness. Finally, for many plants, ISP data will be available sooner to factor into plant operations since there are more sources of data.

The current withdrawal schedule for both units is based on the latest NRC-approved revision of BWRVIP-86 (Reference 5.3-13). Based on this schedule, LGS is not scheduled to withdrawal any additional material specimens. Per Reference 5.3-15, the schedule is not changed due to MUR power uprate.

5.3.1.6.2 Neutron Flux and Fluence Calculations

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

5.3.1.6.3 Predicted Irradiation Effects on Vessel Beltline Materials

Estimated maximum changes in RT_{NDT} (initial reference temperature) and upper shelf fracture energy as a function of the EOL fluence at the ¼ depth of the vessel beltline materials are provided in Section 5.3.1.7. The predicted peak EOL maximum neutron irradiation fluences for the 110% power rerate condition at the ¼ of the vessel beltline are $1.3x10^{18}$ n/cm^2 after 32 EFPY (where an EFPY is based on the rerated power level). For conservative flux calculations, 251 inches is used as the inside diameter of the beltline region with a wall thickness of 6-3/16 inches. Transition temperature changes and variations in upper-shelf energy were calculated in accordance with the rules of Regulatory Guide 1.99 (Rev 2). Reference temperatures were established in accordance with 10CFR50, Appendix G and NB-2330 of the ASME Code. Per Reference 5.3-15, the rerate fluence bounds the requirements for operation at the MUR power uprate.

5.3.1.6.4 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 ASME Section III. A positive spring-loaded locking device is provided to retain the capsules in position throughout any anticipated event during the lifetime of the vessel.

5.3.1.6.5 Dosimetry Measurements

Each surveillance capsule contains iron and copper flux wires. These wires can be used to determine the relationship between reactor power and neutron fluence.

5.3.1.7 Vessel Beltline Plates and Welds

This section supplements Section 5.3.1.5 in discussing the compliance to the intent of 10CFR50, Appendix G.

5.3.1.7.1 Test Data

Available Charpy V-notch and drop-weight impact test data are presented in Tables 5.3-3 and 5.3-4. The sample test welds are prepared in accordance with the ASME Code and do not include base material from the beltline. There are two categories of belt-line welds identified "shop" welds and "field" welds. The shop welds represent vessel vertical seams which were made prior to shipment of preassembled ring segments to the LGS Unit 1 and Unit 2 plant site. The flux material for the submerged arc weld is LINDE 124. The field welds (i.e., girth welds) were made at the plant site. However, exact identification of weld materials used in the beltline girth weld seam is not available. Therefore, a conservative assumption is made to consider all electrodes which were released for field welding the vessel shells.

Figure 5.3-7 shows the vessel beltline layouts. They give plate heat numbers and locations, as well as weld seam locations and identifications.

5.3.1.7.2 Effects of Irradiation

Copper, nickel, and phosphorus values used to estimate the effects of irradiation on toughness are presented in Table 5.3-5.

Estimated starting (i.e., unirradiated) RT_{NDT} values for the beltline plate and weld materials are presented in Table 5.3-5. These values were calculated using the data in Tables 5.3-3 and 5.3-4 in accordance with ASME Section III, NB2300.

Estimated EOL RT_{NDT} values including the influence of power rerate and MUR power uprate for (for ¼ thickness location from the vessel ID) are also given in Table 5.3-5. The EOL RT_{NDT} values are estimated in accordance with Regulatory Guide 1.99 (Rev 2).

5.3.1.7.3 Upper-Shelf Toughness Testing

Charpy V-notch upper-shelf toughness testing was not required when the LGS vessels were manufactured. 10CFR50, Appendix G requires a minimum of 75 ft-lb transverse upper-shelf CVN energy for beltline material. BTP MTEB 5-2 indicates that 70 ft-lb is adequate for fluence levels less than $1x10^{19}$ n/cm^2.

All of the LGS Units 1 and 2 beltline plates were CVN impact- tested as longitudinal specimens at only one temperature, +40°F. For Unit 1, the lowest CVN value obtained for beltline plate was 45 ft-lb with 50% shear, and the highest was 104 ft-lb with 70% shear. For Unit 2, the lowest value obtained for beltline plate was 35 ft-lb with 30% shear, and the highest was 97 ft-lb with 50% shear. The 50% and 30% shear values suggest there is a considerable margin remaining before the upper-shelf (i.e., 100% shear) level is reached.

Table 5.3-6 summarizes the test certificates for representative LGS Units 1 and 2 beltline plate. Similar data are also documented for all other plates. Supporting data from representative plate materials in other BWR plants are provided in Table 5.3-7. Compatibility of the supporting data from other BWRs, Plants A through E, with respect to LGS is based on criteria such as similarity in material, fabrication, vendor source, welding procedure, etc. All listed plate materials were produced by Luken's Steel Co. These data show that plates with as low as 36 ft-lb (Plant E, Heat No. C9570-1) of absorbed energy with 30% shear at +40°F can have longitudinal upper-shelf energies in excess of 100 ft-lb.

BTP MTEB 5-2 states that longitudinal values should be reduced to 65% of the test value to estimate transverse upper-shelf. To account for the power rerate end-of-life fluence of 1.3×10^{18}, a further shift in upper-shelf toughness can be made using Regulatory Guide 1.99, resulting in a maximum reduction of approximately 12.5% for the highest Cu content of 0.12 wt% for Unit 1 and 15% for the highest Cu content of 0.15 wt% for Unit 2 as shown in Table 5.3-5. Using these conservative assumptions with a goal of achieving at least 50 ft-lb transverse toughness at EOL, the following equations are derived:

Unit 1: $50 = .65(L) - (.125)[.65(L)]$

Unit 2: $50 = .65(L) - (.15)[.65(L)]$

(where L is unirradiated longitudinal upper-shelf value.)

These equations predict a minimum required unirradiated longitudinal upper-shelf toughness requirement of 88 ft-lb for Unit 1 and 91 ft-lb for Unit 2. Table 5.3-7 indicates that toughness in excess of 91 ft-lb is to be expected for longitudinal upper- shelf of this material.

Although upper-shelf testing was not required for the beltline welds, Table 5.3-4 shows that the majority of the weld materials, both field and shop, meet the 75 ft-lb minimum upper-shelf requirement.

Weld materials that do not meet the 75 ft-lb minimum upper-shelf requirement were impact-tested exclusively at +10°F. The upper-shelf level (i.e., the absorbed energy corresponding with 100% shear) is not expected at this temperature. For Unit 1, Heat No. 07L857, Lot No. B101A27A evidenced the lowest toughness properties exhibiting a minimum of 28 ft-lbs and 20% shear at +10°F T_{CV}. For Unit 2, Heat No. 432A2671, Lot No. H019A27A evidenced the lowest toughness properties exhibiting a minimum of 31 ft-lbs and 30% shear at +10°F T_{CV}. Again, the 20% and 30% shear values indicate that there is considerable margin for improved properties at higher test temperatures.

For these welds where the minimum upper-shelf energy was not established, the Cu content does not exceed 0.05 wt.%. Based on power rerate EOL fluence of 1.3×10^{18} n/cm^2, Regulatory Guide 1.99 predicts a maximum decrease in upper-shelf energy of approximately 12% for weld material containing .05% Cu. With a goal of achieving at least 50 ft-lbs upper-shelf at EOL, the following equation is derived:

$$50 = (U.V.) - .12 (U.V.)$$

where:

U.V. = the unirradiated upper-shelf value for weld metal.

This equation predicts a minimum required unirradiated upper-shelf level of only 57 ft-lbs.

Further upper-shelf toughness data for representative welds, made by the same vendor as LGS Units 1 and 2, are given in Table 5.3-8. Tables 5.3-9 and 5.3-10 present the weld procedures typical for the data base in Table 5.3-8. These tables represent surveillance program weld procedures and some corresponding baseline data, along with other vessel material data which are considered representative of LGS Units 1 and 2 beltline welds. These data are in excess of 75 ft-lbs at the upper-shelf.

5.3.1.7.4 Upper Shelf Energy Equivalent Margin Analysis

10CFR50 Appendix G requires that 50 ft-lb upper shelf energy be maintained in the vessel beltline low alloy steel material throughout operation. It further requires, if 50 ft-lb USE cannot be demonstrated, that methods to show equivalent margin be provided. Since initial USE data is not available to demonstrate 50 ft-lb USE per NCR methods, an equivalent margin analysis was performed (Reference 5.3-6). This analysis has been approved by the NRC and confirms that, even in the absence of initial USE data, adequate margin of safety against fracture equivalent to 10CFR50 Appendix G requirements does exist. For the MUR power uprate, all materials that did not demonstrate 50 ft-lb USE were qualified in Reference 5.3-15 by the equivalent margin analysis methodology of Reference 5.3-6.

However, the equivalent margin analysis methodology defined in Reference 5.3-6 does not specifically identify that forging materials were included in the statistical calculations. Therefore, the LPCI nozzle forgings and the water level instrumentation (WLI) nozzles were assessed in a plant specific evaluation (Reference 5.3-16). The evaluation concluded that the LPCI nozzle forgings in the RPV meet the margin of safety against fracture equivalent to those required by Appendix G of Section XI of the ASME code. The WLI nozzles were not evaluated further because the forging material is less than 2.5 inches thick. The evaluation for the WLI nozzle in Reference 5.3-15, based on the plate material in the shell where the nozzles are located, is appropriate.

5.3.1.8 Nonbeltline Region and Ferritic Piping and Valves

This section supplements Section 5.3.1.5 in discussing the compliance to the intent of 10CFR50, Appendix G.

Table 5.3-11 lists the estimated reference temperature (RT_{NDT}) for various components in the LGS Units 1 and 2 vessel nonbeltline region. These values were derived in accordance with the intent of ASME Section III, Paragraph NB-2300.

5.3.1.9 Assessment to Appendix G, 1983 Revision

This section addresses the latest requirements of Appendix G, 1983 Revision.

a. The 1983 revision of Appendix G has redefined the adjusted reference temperature by stating "change from the 50 ft-lb level to the 30 ft-lb level of Charpy energy at which the transition temperature shift is to be measured as an indicator of radiation damage." This change does not impact the current reference temperature RT_{NDT} shift information provided in Table 5.3-5. Those shift values were estimated in accordance with Regulatory Guide 1.99 (Rev 2). The data base to establish the procedures for shift prediction in this regulatory guide was given in terms of the 30 ft-lb level; therefore, the predicted reference temperature shifts meet the latest Appendix G requirement.

b. The 1983 revision of Appendix G requires that reactor vessel beltline materials "must maintain upper-shelf energy throughout the life of the vessel of no less than 50 ft-lbs." Section 5.3.1.7.4 describes the equivalent margin analysis which was performed to demonstrate compliance with this requirement.

c. The 1983 revision of Appendix G changes the definition of the "adjacent regions" by stating "beltline region of reactor vessel means the region of the reactor vessel (shell material including welds, heat-affected zones, and plates or forgings) that directly surrounds the effective height of the active core and adjacent regions of the reactor vessel that are predicted to experience sufficient neutron radiation damage to be considered in the selection of the most limiting material with regard to radiation damage." The active fuel region for the LGS vessels is contained by shell courses 1 and 2. The design elevation of the fuel assemblies indicates there remains approximately 36 inches of vessel steel above, and 90 inches of vessel steel below, the fuel elevation before reaching any additional girth welds/shell courses. Therefore, the "adjacent regions" of vessel shell are considered to represent material from shell courses 1 and 2 as well. For this reason, no additional considerations of beltline plates or welds are warranted. Moreover, the effect of radiation on the low pressure coolant injection nozzles and their related welds has been included in the current Appendix G analysis. No other components are near enough to the fuel to be significantly affected by radiation.

5.3.1.10 RPV Surveillance Program

This section supplements Section 5.3.1.6 in discussing the compliance to the intent of 10CFR50, Appendix H.

5.3.1.10.1 LGS Unit 1 Surveillance Program

The base-plate and weld materials used to fabricate the surveillance test plate are identified in Table 5.3-5. The base metal from a core beltline plate Heat No. C7689-1 was used for surveillance test material. With respect to initial RT_{NDT} and percent of copper by weight, this material is considered equivalent to other beltline plates and its utilization for test plate fabrication is in compliance with current recommendations for selection of surveillance materials. The test plate weld, like the core beltline vertical weld seams, was made using both SMAW and SAW welding processes. The test plate weld procedure is presented in Table 5.3-10. For each of the two welding processes, only one heat of weld material was used. The SAW material Heat/Flux No. IP4218/3929-989, which was also used for beltline seams BE, BA, and BB (Figure 5.3-7), is considered suitable for surveillance monitoring because it represents the most limiting SAW material in terms of shift and predicted EOL RT_{NDT}. The SMAW material Heat/Lot No. 421A6811/F022A27A which was used for surveillance material was not used for production beltline welds; however, the weight percentages of copper and nickel which it contains (0.09% Cu and 0.81% Ni) are generally greater than those for actual belt-line material. Moreover, the unirradiated RT_{NDT} of this material is equivalent to the initial RT_{NDT} of the beltline weld materials. The Chicago Bridge and Iron weld procedure for test plate fabrication involves utilizing stick electrode to fuse backup bars and completing the major volume of the weld with SAW. This includes back-gouging of the backup bar to complete the back side of the weld. Therefore, the test plate weld metal is essentially SAW welded material. Table 5.3-5 indicates that all beltline materials, both plate and weld, are highly resistant to irradiated degradation of notch toughness.

5.3.1.10.2 LGS Unit 2 Surveillance Program

The base-plate and weld materials used to fabricate the surveillance test plate are identified in Table 5.3-5. The base metal used was plate from Heat No. C9569-2; this material was also used for one of the core beltline plates, and its use in RPV surveillance is consistent with the requirements at the time of vessel fabrication.

The test plate weld, like the core beltline vertical weld seams, was made using SMAW and SAW welding processes. The test plate weld procedure is presented in Table 5.3-10. The SAW material Heat No. 3P4000, Lot No. 3933 (both single and tandem wire) was also used for fabrication of beltline seams BA, BB, BC, BD, BE, and BF, and is considered suitable for surveillance monitoring.

Two heats of SMAW material were used for test plate fabrication. One weld material, Heat/Lot No. CTY538/A027A27A, was not used for production beltline welds; however, the weight percentages of Cu and Ni (0.03% Cu and 0.83% Ni) are equivalent to the concentrations of these elements for actual beltline material. The other SMAW material, Heat/Lot No. 03R728/L910A27A, was used in production beltline seam fabrication and is considered suitable for surveillance monitoring.

Table 5.3-5 indicates that all beltline materials, both plate and weld, are acceptably resistant to irradiation degradation of notch toughness.

For Units 1 and 2, Table 5.3-12 lists the actual number of specimens and their orientations in each surveillance capsule (including tensile specimens). The number and orientation of the Charpy impact specimens are consistent with the ASME requirements prior to the issuance of the Summer 1972 Addenda and ASTM 185-73.

Prior surveillance experience indicates the amount of radiation-induced shift in properties measured by longitudinally oriented specimens is applicable to equivalent transverse oriented specimens (Reference 5.3-2). Therefore, the shift when determined can be used for the transverse RT_{NDT} values for the beltline materials. Referring to Table 5.3-12, the longitudinal orientation of the base metal heat-affected zone specimens are such that they simulate beltline vertical seams in this manner.

5.3.1.11 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 ASME Section III, Class I, code requirements. The material for studs, nuts, and washers is SA540, Grade B23 or Grade B24. The maximum reported ultimate tensile stresses for the bolting material are 164,000 psi for Unit 1 and 169,000 psi for Unit 2 which are less than the 170,000 psi limitation in Regulatory Guide 1.65. Also, the Charpy impact test recommendations of paragraph IV.A.4 of 10CFR50, Appendix G were not specified in the vessel order since the order was placed prior to issuance of 10CFR50, Appendix G. However, impact data from the certified materials report shows that all bolting materials meet the Appendix G impact properties.

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

The basis for setting operational limits on pressure and temperature for normal, upset, and test conditions for the RPV is described in Section 5.3.1.5.

5.3.2.1.1 Temperature Limits for Bolt-Up

A minimum temperature of 10°F is required on LGS Unit 1 and 10°F on LGS Unit 2 for the closure studs. A sufficient number of studs can be tensioned at a temperature between 10°F and 80°F for Unit 1 and 70°F for Unit 2 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 minimum temperatures of 80°F (Unit 1) and 70°F (Unit 2) before they are stressed by the full intended bolt preload (all bolts tensioned). The fully preloaded bolt-up limits are shown on Figures 5.3-4 and 5.3-5. Per Reference 5.3-15, Figure 5.3-4 is applicable at the MUR power uprate condition.

5.3.2.1.2 Temperature Limit for Preoperational Tests and Inservice Inspection

Based on the NRC general revision to 10CFR50, Appendix G, Document No. [7590-1], paragraph IV.A.4, the preoperational system hydrostatic test at 1563 psig prior to fuel loading was performed at a minimum temperature of 100°F for Unit 1 without fuel in the reactor, and 100°F for Unit 2. These limits were established by the 40°F maximum RT_{NDT} of the reactor vessel materials.

The fracture toughness analysis for system pressure tests with fuel in the reactor yields the curves labeled A shown in Figures 5.3-4 and 5.3-5. The curves labeled "core beltline" are based on an initial RT_{NDT} of 20°F for Unit 1 and 40°F for Unit 2. The predicted shift in the RT_{NDT} based on the power rerate neutron fluence at ¼ of the vessel wall thickness, has been added to the belt-line curve to account for the effect of fast neutrons. For Unit 2, intermediate A' curves have been provided for 6.5 and 8.5 EFPY (Reference 5.3-7). Per Reference 5.3-15, Figure 5.3-4 is applicable at the MUR power uprate condition.

5.3.2.1.3 Operating Limits During Heatup, Cooldown, and Core Operation

The fracture toughness analysis is done for the normal heatup or cooldown rate of 100°F/hour. The temperature gradients and thermal-stress effects corresponding to this rate are included. The results of the analyses are a set of operating limits for non-nuclear heatup or cooldown shown as curves labeled B in Figures 5.3-4 and 5.3-5. Curves labeled C in these figures apply whenever the core is critical. The predicted shift in the RT_{NDT}, based on the power rerate neutron fluence at ¼ of the vessel wall thickness, has been added to the belt-line curve to account for the effect of fast neutrons. Per Reference 5.3-15, Figure 5.3-4 is applicable at the MUR power uprate condition.

5.3.2.1.4 Reactor Vessel Annealing

Inplace annealing of the reactor vessel because of radiation embrittlement is not anticipated to be necessary because the predicted value of adjusted reference temperature does not exceed 200°F (10CFR50, Appendix G), even for the power rerate condition or the MUR power condition.

5.3.2.2 Operating Procedures

By comparison of the pressure versus temperature limits in Section 5.3.2.1 with intended normal operating procedures for the most severe upset transient, it is shown that the limits are not exceeded during any foreseeable upset condition. Reactor operating procedures are established so that actual transients are not 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 yields a minimum fluid temperature of 250°F and a maximum power rerate pressure peak of 1233 psig. Scram automatically occurs with initiation of this event, prior to the reduction in fluid temperature, so the applicable operating limits are given by curve B in Figures 5.3-4 and 5.3-5. For a temperature of 250°F, the maximum allowable pressure exceeds 1233 psig for the intended margin against nonductile failure. The maximum transient pressure is therefore within the specified allowable limits.

5.3.3 REACTOR VESSEL INTEGRITY

The reactor vessels are fabricated for GE's Nuclear Energy Division by Chicago Bridge and Iron Company; and are subject to the requirements of GE's Quality Assurance program.

Measures are established to ensure that purchased material, equipment, and services associated with the reactor vessels and appurtenances conform to the requirements of the purchase documents. These measures include 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.

GE provides 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 is responsible for the first level inspection of his manufacturing, fabrication, and testing activities and GE is responsible for the first level of audit and surveillance inspection.

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

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 LGS are designed, fabricated, tested, inspected, and stamped in accordance with the ASME Section III, Class A including the Summer Addenda 1969. Design of the reactor vessel and its support system meets seismic Category I requirements.

The materials used in the RPV are shown in Table 5.2-3.

The cylindrical shell 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.

Inplace 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 irradiation in those areas is less than $1x10^{18}$ nvt with neutron energies in excess of 1 MeV.

Quality control methods used during the fabrication and assembly of the reactor vessel and appurtenances ensure that design specifications are met.

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

The BWR does not use borated water for reactivity control. This section is therefore not applicable.

5.3.3.1.2 Safety Design Basis

The design of the reactor vessel and appurtenances meets the following safety design bases:

a. The reactor vessel and appurtenance will withstand adverse combinations of loading and forces resulting from operation under abnormal and accident conditions

b. To minimize the possibility of brittle fracture of the nuclear system process barrier, the following are required:

    1. Impact properties at temperatures related to vessel operation are specified for materials used in the reactor vessel

    2. 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 ensure that RT_{NDT} temperature shifts are accounted for in reactor operation.

    3. 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 bases:

a. The reactor vessel has been designed for a useful life of 40 years.

b. 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.

c. Design of the reactor vessel and appurtenances allows for a suitable program of inspection and surveillance.

5.3.3.1.4 Reactor Vessel Design Data

The reactor vessel design pressure is 1250 psig and the design temperature is 575°F. The maximum installed test pressure is 1656 psig (1563 psig plus the ASME Code tolerance of 6%).

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 structure-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 CRD housings are inserted through the CRD 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 CRD, 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 Incore Neutron Flux Monitor Housings

Each incore neutron flux monitor housing is inserted through the incore penetrations in the bottom head and is welded to the inner surface of the bottom head.

An incore flux monitor guide tube is welded to the top of each housing and either a SRM/IRM drive unit or a LPRM is bolted to the seal/ring flange at the bottom of the housing (Sections 7.6 and 7.7).

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. Most of the reactor vessel insulation is of the stainless steel, reflective type. The insulation used for many of the large reactor vessel nozzles is fiberglass-type, in blanket form. The top head insulation framework is designed to seismic Category I requirements and is used as an anchor point for reactor vessel vent piping.

The insulation above the reactor vessel stabilizer brackets is close-fitting, freestanding 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. Some of the suspended insulation is removable to permit access for manual inspection in locations where remote inspection may not be feasible. 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 connected to the reactor vessel nozzles is designed to not exceed the allowable loads on any nozzle.

The vessel top head nozzles are provided with a flange with small groove facings. 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, the RHR LPCI inlet nozzles, and the core spray inlet nozzles all have thermal sleeves.

Nozzles connecting to stainless steel piping have safe ends made of stainless steel or Inconel (ASME Section III, SB-166). 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.

5.3.3.1.4.6 Materials and Inspections

The reactor vessel is designed and fabricated in accordance with the appropriate ASME Code as defined in Section 5.2.1. Table 5.2-3 defines the materials and specifications. Section 5.3.1.6 defines the compliance with reactor vessel material surveillance program requirements.

5.3.3.1.4.7 Reactor Vessel Schematic (BWR)

The reactor vessel schematic is contained in Figure 5.3-1. Trip system water levels are indicated as shown in Figure 5.3-2.

5.3.3.2 Materials of Construction

All materials used in the construction of the RPV conform to the requirements of ASME Section II materials. The vessel heads, shells, flanges, and nozzles are fabricated from low alloy steel plate and forgings purchased in accordance with ASME SA533, Grade B, Class 1 and ASME SA508, Class 2. Special requirements for the low alloy steel plate and forgings are discussed in Section

5.3.1.2. Cladding employed on the interior surfaces of the vessel consists of austenitic stainless steel weld overlay.

These materials are selected because they provide adequate strength, fracture toughness, fabricability, and compatibility with the BWR environment. Their suitability is demonstrated by long-term successful operating experience in reactor service.

5.3.3.3 Fabrication Methods

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

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. Postweld heat treatment of 1100°F minimum is applied to all low alloy steel welds.

All previous BWR pressure vessels employed similar fabrication methods. These vessels have operated for periods of up to 16 years and their service history is excellent.

The vessel fabricator, Chicago Bridge and Iron Co., has had extensive experience with GE reactor vessels dating back to 1966. Chicago Bridge and Iron Nuclear Co. was formed in 1972 from a merger agreement between Chicago Bridge and Iron Co. and GE and has continued as the primary supplier for GE domestic reactor vessels.

5.3.3.4 Inspection Requirements

All plate, forgings, and bolting 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 meet the acceptance requirements specified by ASME Section III. In addition, the pressure-retaining welds were ultrasonically examined in accordance with ASME Section XI requirements prior to shipping.

5.3.3.5 Shipment and Installation

The LGS reactor vessels were assembled at the site. Methods and procedures are discussed in the PSAR, Appendix G. Suitable measures were taken during installation to ensure that vessel integrity was maintained; for example, access controls were applied to personnel entering the vessel, weather protection was provided, and periodic cleanings were 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:

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

b. If the coolant temperature difference between the dome (inferred from PSAT) and the bottom head drain exceeds 145°F, the reactor recirculation pumps shall not be started, and neither reactor power nor recirculation pump flow shall be increased.

c. 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 a) ensures that the vessel closure, closure studs, vessel support skirt, and CRD 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 b) augments the item a. limit in further detail by ensuring that the vessel bottom head region is not 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 inleakage and/or low recirculation flow rate during startup or hot standby). The item c. 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 are maintained to ensure that the stress limits within the reactor vessel and its components are within the thermal limits to which the vessel is designed for normal operating conditions. To maintain the integrity of the vessel if these operational limits are exceeded, the reactor vessel is also 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 are included in the design conditions. Therefore, it is concluded that vessel integrity is maintained during the most severe postulated transients, since all such transients are evaluated in the design of the reactor vessel.

5.3.3.7 Inservice Surveillance

The vessel is examined once prior to startup to satisfy the preoperational requirements of ASME Section XI. Subsequent ISI is scheduled and performed in accordance with the requirements of 10CFR50.55a .

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 the thermal environment. Specimens of actual reactor beltline material are exposed in the reactor vessel and periodically withdrawn for impact testing. Operating procedures are modified in accordance with test results to ensure adequate brittle fracture control.

Material surveillance programs and ISI programs are in accordance with applicable ASME code requirements, and provide assurance that brittle fracture control and pressure vessel integrity are maintained throughout the service life of the RPV.

Inservice inspection and testing of the RCPB is discussed in detail in Section 5.2.4.

5.3.4 REFERENCES

5.3-1 "Metal Progress", pp. 35-39, (July 1978).

5.3-2 "Radiation Effects in BWR Pressure Vessel Steels", GE Licensing Topical Report, NEDO-21708.

5.3-3 Letter MFN-414-77, G.G. Sherwood (GE) to Edson G. Case (NRC), (October 17, 1977).

5.3-4 Letter, Robert B. Minogue (NRC) to G.G. Sherwood (GE), (February 14, 1978).

5.3-5 NEDO-21778A, Transient Pressure Rise Affecting Fracture Toughness for BWR's, GE Licensing Topical Report (December 1978).

5.3-6 NEDO-32205-A, Revision 1, 10CFR50 Appendix G Equivalent Margin Analysis for Low Upper Shelf Energy in BWR/2 Through BWR/6 Vessels.

5.3-7 Reference no longer used

5.3-8 General Electric Co. Report GE-NE-B1100786-01R1, “Surveillance Specimen Program Evaluation for Limerick Generating Station Unit 1”, dated December 1997.

5.3-9 General Electric Co. Report GE-NE-B1100786-02, “Surveillance Specimen Program Evaluation for Limerick Generating Station Unit 2”, dated June 1998.

5.3-10 “Limerick Generating Station, Units 1 and 2, SRV Setpoint Tolerance Relaxation Licensing Report”, NEDC-32645P (December 1998).

5.3-11 General Electric Co. Report, GE-NE-B11-00836-00-01 “Pressure – Temperature Curves for PECO Energy Company Limerick Unit 1,” Dated April 2000.

5.3-12 General Electric Co. Report, GE-NE-B11-00836-00-02 “Pressure – Temperature Curves for PECO Energy Company Limerick Unit 2,” Dated July 2000.

5.3-13 BWRVIP-86-A: “BWR Vessel and Internals Project, Updated BWR Integrated Surveillance Program (ISP),” Final Report, October 2002.

5.3-14 Letter from S. P. Wall (U.S. NRC) to J. L. Skolds (Exelon Generation Company, LLC), "Limerick Generating Station Units 1 and 2 - Issuance of Amendment Re: Revision to the Reactor Pressure Vessel Material Surveillance Program (TAC Nos. MB7003 and MB7004)," dated November 4, 2003."

5.3-15 Design Analysis LEAM-MUR-0043, Revision 2, T0301, "RPV Fracture Toughness Evaluation"

5.3-16 "Upper Shelf Energy Evaluation for LPCI Nozzle Forging Material," GEH-0000-0114-0580-R0, Revision, August 2010.

Table 5.3-1

APPENDIX G MATRIX FOR LGS
(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 Section III NB-2300	Yes	See Section 5.3.1.5.1.2 for discussion
III.B.1	Location and orientation of impact test specimens	Yes	See III A above
III.B.2	Materials used to prepare test specimens	No	Compliance except for CVN orientation and CVN upper shelf
IIII.B.3	Calibration of temperature, instrumentation, and Charpy test machines	No	Paragraph NB-2360 of the ASME Section III was not in existence at the time of purchase of the Unit 1 RPV. However the requirements of the 1971 edition of ASME Section III, Summer 1971 addenda are met. For the discussions of the GE interpretations of compliance and NRC acceptance, see References 5.3-3 and 5.3-4. The temperature instruments and Charpy test machines calibration data are retained until the next calibration. This is accordance with Regulatory Guide 1.88 (Section 1.8) and ANSI N45.2.9 (1974). Therefore, the instrument calibration data for Unit 1 are not currently available.
III.B.4	Qualification of testing personnel	No	No written procedures were in existence as now required by the regulation; however individuals were qualified by on-the-job training and past experience. For a discussion of the GE interpretation of compliance and NRC acceptance see References 5.3-3 and 5.3-4.
III.B.5	Test results recording and certification	Yes	See References 5.3-3 and 5.3-4.
III.C.1	Test condtions	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 welds were not necessarily made on the same heat as that of the base plate.

Table 5.3-1 (Cont'd)

(UNIT 1)

APPENDIX G PARA. NO.	TOPIC	COMPLY YES/NO OR N/A	ALTERNATE ACTIONS OR COMMENTS
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^{o}F added to the RT_{NDT} for the reactor vessel flanges. For feedwater nozzles, the results of the BWR/6 analysis are adjusted to LGS Unit 1 RT_{NDT} conditions.
IV.A.2.c	RPV metal temperature requirement when core is critical	No	Regulation change in process (See Reference 5.3-5)
IV.A.2.d	Minimum permissible temperature during hydro test	Yes	
IV.A.3	Materials for piping, pumps, and valves	No	See section 5.2.3.3.1
IV.A.4	Materials for bolting and other fasteners	Yes	Meet requirements for closure studs at 10^{o}F.
IV.B	Minimum upper-shelf energy for RPV beltline	No	See Section 5.3.1.7.4 and Reference 5.3-6 for discussion on equivalent margin analysis.
IV.C	Requirement for annealing when RT_{NDT} > 200^{o}F	NA	
V.A	Requirements for material surveillance program	-	See Table 5.3-2
V.B	Conditions for continued operation	Yes	Meet requirements of IV.A.2

Table 5.3-1 (Cont'd)

(UNIT 1)

APPENDIX G PARA. NO.	TOPIC	COMPLY YES/NO OR N/A	ALTERNATE ACTIONS OR COMMENTS
V.C	Alternate if V.B cannot be satisfied	NA	
V.D.	Requirement for RPV thermal annealing if V.C. cannot be met	NA	
V.E.	Reporting requirement for V.C. and V.D. COMPLY	NA	

Table 5.3-1 (Cont'd)

APPENDIX G MATRIX FOR LGS

(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 Section NB-2300	Yes	See Section 5.3.1.5.1.2 for discussion
III.B.1	Location and orientation of impact test specimens	Yes	See III A above
III.B.2	Materials used to prepare test specimens	No	Compliance except for CVN orientation and CVN upper shelf
IIII.B.3	Calibration of temperature, instrumentation, and Charpy test machines	No	Paragraph NB-2360 of the ASME Section III was not in existence at the time of purchase of the Unit 2 RPV. However the requirements of the 1971 edition of ASME Section III, Summer 1971 Addenda are met. For the discussions of the GE interpretations of compliance and NRC acceptance, see References 5.3-3 and 5.3-4. The temperature instruments and Charpy test machines calibration data are retained until the next recalibration. This is accordance with Regulatory Guide 1.88 (Section 1.8) and ANSI N45.2.9 (1974). Therefore, the instrument calibration data for Unit 2 are not currently available.
III.B.4	Qualification of testing personnel	No	No written procedures were in existence as now required by the regulation; however individuals were qualified by on-the-job training and past experience. For a discussion of the GE interpretation of compliance and NRC acceptance see References 5.3-3 and 5.3-4.
III.B.5	Test results recording and certification	Yes	See References 5.3-3 and 5.3-4.

Table 5.3-1 (Cont'd)

(UNIT 2)

APPENDIX G PARA. NO.	TOPIC	COMPLY YES/NO OR N/A	ALTERNATE ACTIONS OR COMMENTS
III.C.1	Test condtions	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 welds were not necessarily made on the same heat as that of the base plate.
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 added to the RT_{NDT} for the reactor vessel flanges. For feedwater nozzles, the results of the BWR/6 analysis were adjusted to LGS Unit 2 RT_{NDT} conditions.
IV.A.2.c	RPV metal temperature requirement when core is critical	No	Regulation change in process (See Reference 5.3-5)
IV.A.2.d	Minimum permissible temperature during hydro test	Yes	
IV.A.3	Materials for piping, pumps, and valves	No	See section 5.2.3.3.1
IV.A.4	Materials for bolting and other fasteners	Yes	See Section 5.2.3.3.1.1 for discussion.
IV.B	Minimum upper-shelf energy for RPV beltline	No	See Section 5.3.1.7.4 and Reference 5.3-6 for discussion on equivalent margin analysis.
IV.C	Requirement for annealing when RT_{NDT} > 200°F	NA	
V.A	Requirements for material surveillance program	-	See Table 5.3-2
V.B	Conditions for continued operation	Yes	Meet requirements of IV.A.2

Table 5.3-1 (Cont'd)

(UNIT 2)

APPENDIX G PARA. NO.	TOPIC	COMPLY YES/NO OR N/A	ALTERNATE ACTIONS OR COMMENTS
			EOL 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	NA	
V.A	Requirements for material surveillance program		See Table 5.3-2
V.B	Conditions for continued operation	Yes	Meet requirements of IV.A.2
V.C	Alternate if V.B cannot be satisfied	NA	
V.D.	Requirement for RPV thermal annealing if V.C. cannot be met	NA	
V.E.	Reporting requirement for V.C. and V.D.	NA	

Table 5.3-2

APPENDIX H MATRIX FOR LGS

APPENDIX H PARA. NO.	TOPIC	COMPLY YES/NO OR N/A	ALTERNATE ACTIONS OR COMMENTS
I	Introduction	NA	
II.A	Fluence $<10^{17}$ n/cm^2 - surveillance program not required	NA	
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 is taken from locations alongside the fracture test specimens (Section III.B of Appendix G)	No	Noncompliance in that specimens are not necessarily taken from alongside specimens required by Section III fo Appendix G and transverse CVNs are employed. However, representative materials are 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 the attachment of brackets to vessel cladding (Section 5.3.1.6.4).
II.C.3.a	Withdrawal schedule of capsules, $RT_{NDT} \leq 100°F$	Yes	Three capsules planned. Starting RT_{NDT} of limiting material is based on alternative action (see paragraph III.A of Appendix G).
II.C.3.b	Withdrawal schedule of capsules, $100°F < RT_{NDT} \leq 200°F$	NA	
II.C.3.c	Withdrawal schedule of capsules, $RT_{NDT} > 200°F$	NA	
III.A	Fracture toughness testing requirements of specimens	No	CVN tests only

Table 5.3-2 (Cont'd)

(UNIT 1)

APPENDIX H PARA. NO.	TOPIC	COMPLY YES/NO OR N/A	ALTERNATE ACTIONS OR COMMENTS
III.B	Method of determining adjusted reference temperature for base metal, heat affected zone and weld metal	No	II.B and II.C.1 above.
IV.A	Reporting requirements of test results	Yes	
IV.B	Requirement for dosimetry measurement	Yes	
IV.C	Reporting requirements of pressure/temperature limits	Yes	

Table 5.3-2 (Cont'd)

APPENDIX H MATRIX FOR LGS

(UNIT 2)

APPENDIX H PARA. NO.	TOPIC	COMPLY YES/NO OR N/A	ALTERNATE ACTIONS OR COMMENTS
I	Introduction	NA	
II.A	Fluence $<10^{17}$ n/cm^2 - surveillance program not required	NA	
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. (Section 5.3.1.10)
II.C.1	Surveillance specimen is taken from locations alongside the fracture test specimens (Section III.B of Appendix G)	No	Noncompliance in that specimens are not necessarily taken from alongside specimens required by Section III fo Appendix G and transverse CVNs are not employed. However, representative materials are 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 the attachment of brackets to vessel cladding (Section 5.3.1.6.4).
II.C.3.a	Withdrawal schedule of capsules, $RT_{NDT} \leq 100°F$	Yes	Three capsules planned. Starting RT_{NDT} of limiting material is based on alternative action (see paragraph III.A of Appendix G).
II.C.3.b	Withdrawal schedule of capsules, $100°F < RT_{NDT} \leq 200°F$	No	Material with , RT_{NDT} shift > 100° F is not limiting material; shift was predicted < 100° F at time of surveillance program design,
II.C.3.c	Withdrawal schedule of capsules, $RT_{NDT} > 200°F$	NA	
III.A	Fracture toughness testing requirements of specimens	No	CVN tests only

Table 5.3-2 (Cont'd)

(UNIT 2)

APPENDIX H PARA. NO.	TOPIC	COMPLY YES/NO OR N/A	ALTERNATE ACTIONS OR COMMENTS
III.B	Method of determining adjusted reference temperature for base metal, heat affected zone and weld metal	No	II.B and II.C.1 above.
IV.A	Reporting requirements of test results	Yes	
IV.B	Requirement for dosimetry measurement	Yes	
IV.C	Reporting requirements of pressure/temperature limits	Yes	

Table 5.3-3

LGS BELTLINE PLATE TOUGHNESS DATA

(UNIT 1)

SHELL COURSE	HEAT NO./ SLAB NO.	NDT (^oF)	ORIENTATION	CHARPY TEST TEMP (^oF)	ENERGY (FT-LB)			LAT. EXPANSION (MIL)			% SHEAR		
NO. 1													
I. D. 14-1	C7688-1	TOP –10	L	+40	84	78	58	62	48	64	40	50	50
		BOTTOM –10	L	+40	78	58	85	78	58	85	40	50	50
I.D. 14-3	C7688-2	TOP –10	L	+40	69	84	79	75	67	58	50	50	50
		BOTTOM –10	L	+40	104	90	86	66	72	78	70	70	70
I. D. 14-2	C7698-2	TOP –10	L	+40	77	88	73	75	66	52	50	50	70
		BOTTOM –10	L	+40	100	98	87	79	72	64	50	60	60
NO. 2													
I. D. 17-3	C7698-1	TOP –10	L	+40	82	84	84	61	63	61	50	50	50
		BOTTOM –10	L	+40	85	96	80	69	63	66	50	50	50
I. D. 17-1	C7689-1	TOP –10	L	+40	87	93	77	73	69	62	50	60	60
		BOTTOM –10	L	+40	75	86	81	61	71	78	50	60	60
I. D. 17-2	C7677-1	TOP –10	L	+40	71	71	61	52	48	56	40	40	40
		BOTTOM –10	L	+40	71	45	65	54	58	55	40	50	50

Table 5.3-3 (Cont'd)

LGS BELTLINE PLATE TOUGHNESS DATA

(UNIT 2)

SHELL COURSE	HEAT NO./ SLAB NO.	NDT (°F)	ORIENTATION	CHARPY TEST TEMP (°F)	ENERGY (FT-LB)			LAT. EXPANSION (MIL)			% SHEAR		
NO. 1													
I. D. 14-1	B3312-1	TOP –20	L	+40	73	69	78	53	53	61	50	50	50
		BOTTOM –20	L	+40	58	63	68	44	48	47	60	60	60
I.D. 14-3	C9621-2	TOP 0	L	+40	44	47	60	35	35	44	30	30	30
		BOTTOM –20	L	+40	77	89	79	60	69	60	30	30	30
I. D. 14-2	B3416-1	TOP +10	L	+40	51	42	55	32	39	48	40	40	50
		BOTTOM 0	L	+40	61	35	37	45	32	36	50	30	30
NO. 2													
I. D. 17-3	C9526-2	TOP –40	L	+40	71	74	87	57	52	50	50	50	50
		BOTTOM –30	L	+40	83	84	97	65	58	62	50	50	50
I. D. 17-1	C9569-2	TOP –20	L	+40	68	62	68	45	50	47	30	30	30
		BOTTOM –30	L	+40	83	87	66	62	50	64	40	40	40
I. D. 17-2	C9526-1	TOP –40	L	+40	65	60	66	47	49	55	30	30	30
		BOTTOM –40	L	+40	73	63	89	68	50	56	40	40	40

(1) Longitudinal or transverse

Table 5.3-4

BELTLINE WELD TOUGHNESS DATA

(UNIT 1)

BELTLINE SHOP WELD TOUGHNESS

				CHARPY IMPACT TOUGHNESS									
IDENTITY	PROCESS	HEAT NO.	FLUX LOT	CV TEMP (°F)	ABSORBED ENERGY (ft-lb)			LATERAL EXP. (mil)			% SHEAR		
Weld B-E	SMAW	411A3531	H004A27A	+10 -20	60 41	60 68	68 48	51 39	52 53	54 41	60 35	50 35	60 25
Welds, B-A B-D, B-E, B-F	SMAW	06L165	F017A27A	+10 -20	60 46	61 53	62 32	40 34	52 39	46 24	70 25	60 25	70 25
Welds B-A B-D, B-E, B-F	SMAW	662A746	H013A27A	+10 -20	35 89	38 82	47 95	35 69	31 64	43 68	50 45	50 40	50 65
Welds B-A, B-B, B-C	SAW	3P4000	3932-989	+10	97	95	88	85	82	64	80	80	70
Weld B-F	SAW	S3986	Run #934	+10	46	51	49	38	44	43	40	40	40
Welds B-A, B-B, B-E	SAW	1P4218	3929-989	+10	98 94	100 91	102 90	72 58	65 66	83 77	82 98	65 95	83 95
Surveillance Test Plate Weld	SAW	421A6811	F022A27A	+10	80	85	91	64	73	72	70	75	75
N-17 Nozzle	SMAW	07L669	K004A27A			73 (Note 2)							
N-17 Nozzle	SMAW	411A3531	H004A27A			73 (Note 2)							
N-17 Nozzle	SMAW	401Z9711	A022A27A			73 (Note 2)							
N-17 Nozzle	SMAW	662A746	H013A27A			73 (Note 2)							
N-17 Nozzle	SMAW	S3986	RUN #934			73 (Note 2)							

Table 5.3-4 (Cont'd)

UNIT 1

BELTLINE FIELD WELD TOUGHNESS DATA

Test No.	983
Trade Name:	Atom Arc 8018NM
Diameter Size:	1/8 in 1,400 lb
Lot No.:	B101A27A
Heat No.:	07L857

CHEMICAL TEST RESULTS

Carbon	0.060
Manganese	1.20
Nickel	0.97
Silicon	0.42
Molybdenum	0.55
Copper	0.03
Phosphorus	0.012
Sulfur	0.017

MECHANICAL TEST RESULTS

Test Specimen PW ht @ 1100°F to 1150°F for 62½ hr

TENSILE PROPERTIES

Specimen Type: 0.505 in
UTS: 89,600 psi
YKP: 76,200 psi
Elongation in 2 in: 30%
Red of Area: 71.7%

IMPACT PROPERTIES

Specimen Type: Charpy V-Notch
Test Temp: +10°F
Energy (ft-lb): 28, 36, 39
Lateral Expansion (mil): 27, 41, 45
% Shear: 20, 40, 50

OTHER TESTS

Concentricity: 4%
Moisture @ 1800°F: 0.18%

Table 5.3-4 (Cont'd)

(UNIT 1)

		MECHANICAL TESTS
Test No.:	38	Test Specimen PW @ 1100°F to 1150°F for 62½ hr
Trade Name:	Atom Arc 8018NM	
Diameter Size:	5/32 in 6,750 lb	
Lot No.:	C115A27A	
Heat No.:	402C4371	TENSILE PROPERTIES
		Specimen Type: 0.505 in UTS: 94,000 psi YLP: 87,000 psi Elongation in 2 in: 26% Red of Area: 71.3%
CHEMICAL TEST RESULTS		
Carbon	0.033	
Manganese	1.22	
Nickel	0.92	
Silicon	0.49	
Molybdenum	0.57	
Copper	0.02	IMPACT PROPERTIES
Phosphorus	0.009	
Sulfur	0.014	Specimen Type Charpy V-Notch Test Temp: +10°F Energy (ft-lb): 82, 81, 92 Lateral Expansion (mil): 62, 61, 66 % Shear Area: 80, 70, 70
		OTHER TESTS
		Concentricity: 5% Moisture @ 1800°F: 0.18%

Table 5.3-4 (Cont'd)

(UNIT 1)

MECHANICAL TESTS

Test No.:	WO #11-D	Heat Treatment 1100°F to 1150°F for 62½ hr
Type Electrode:	E8018NM	
Trade Name:	Atom Arc 8018NM	
Electrode Diameter:	3/16 in	
Lot No.:	H004A27A	
Heat No.:	411A3531	

CHEMICAL TEST RESULTS

Carbon	0.066
Manganese	1.13
Nickel	0.96
Silicon	0.51
Molybdenum	0.47
Copper	0.02
Phosphorus	0.018
Sulfur	0.017

TENSILE PROPERTIES

Specimen Type: 0.505 in
UTS: 84,500 psi
YLP: 71,500 psi
Elongation in 2 in: 29%
Red of Area: 72.5%

IMPACT PROPERTIES

Specimen Type: Charpy V-Notch
Test Temperature: -20°F
Energy (ft-lb): 41, 68, 48
Lateral Expansion (mil): 39, 53, 41
% Shear: 35, 35, 25

Table 5.3-4 (Cont'd)

(UNIT 1)

		MECHANICAL TESTS
Test No.:	27	Test Specimen PW ht @ 1100°F to 1150°F for 62½ hr
Trade Name:	Atom Arc 8018NM	
Diameter Size:	7/32 in 13,800 lb	
Lot No.:	C109A27A	TENSILE PROPERTIES
Heat No.:	09M057	Specimen Type: 0.505 in UTS: 94,500 psi YLP: 85,000 psi Elongation in 2 in: 27% Red of Area: 69.8%
CHEMICAL TEST RESULTS		
Carbon	0.063	
Manganese	1.18	
Nickel	0.89	
Silicon	0.47	
Molybdenum	0.53	
Copper	0.03	IMPACT PROPERTIES
Phosphorus	0.009	
Sulfur	0.021	Specimen Type: Charpy V-Notch Test Temp: +10°F Energy (ft-lb): 43, 43, 44 Lateral Expansion (mil): 40, 41, 41 % Shear: 50, 60, 50
		OTHER TESTS
		Concentricity: 4% Moisture @ 1800°F: 0.18%

Table 5.3-4 (Cont'd)

(UNIT 1)

		MECHANICAL TESTS
Test No.:	346	Stress relieved 50 hr @ 1150°F
Trade Name:	Atom Arc 8018NM	
Diameter Size:	3/16 in 7,950 lb	
Lot No.:	J417B27AF	TENSILE PROPERTIES
Heat No.:	412P3611	UTS: 87,500 psi YLP: 75,000 psi Elongation in 2 in: 28% Red of Area: 71.2%

CHEMICAL TEST RESULTS

Carbon	0.07
Manganese	1.10
Chromium	0.03
Nickel	0.93
Silicon	0.36
Molybdenum	0.47
Copper	0.03
Phosphorus	0.016
Sulfur	0.019
Vanadium	0.02
Aluminum	<0.01

IMPACT PROPERTIES

See page 5.3-38 for impact values

OTHER TESTS

Concentricity: 3%
Moisture @ 1800°F: 0.2%

Table 5.3-4 (Cont'd)

(UNIT 1)

DROP-WEIGHT TESTS	SPECIMEN	TEST TEMPERATURE (°F)	RESULTS
MATERIAL: 8018NM	1	-90	Break
LOT: J417B27AF	2	-80	Break
HEAT: 412P3611	3	-70	No Break
	4	-70	No Break

NDT TEMPERATURE = -80°F

CVN IMPACT TESTS

SPECIMEN	TEST TEMPERATURE (°F)	ENERGY (ft-lb)	LATERAL EXP. (mil)	% SHEAR
1	-100	8	6	3
2	-100	12	10	5
3	-80	15	13	10
4	-80	16	14	10
5	-80	19	15	10
6	-20	52	41	30
7	-20	65	54	50
8	-20	69	53	45
9	+40	100	80	90
10	+40	103	68	80
11	+72	133	91	90
12	+72	138	92	90
13	+130	136	89	100
14	+130	137	95	100
15	+130	146	97	100

T_{cv} = -20°F

REFERENCE TEMPERATURE

Material	T_{NDT} (Drop-Weight)	T_{CV} (Charpy V-Notch)	RT_{NDT} (References)
Weld Metal	-80°F	-20°F	-80°F

Table 5.3-4 (Cont'd)

(UNIT 1)

Test No.:	46	MECHANICAL TEST RESULTS
Trade Name:	Atom Arc 8018NM	Test Specimen PW ht @ 1100°F to 1150°F for 62½ hr
Diameter Size:	3/16 in 7,900 lb	
Lot No.: Heat No.:	C118A27A 03M014	TENSILE PROPERTIES
		Specimen Type: 0.505 in UTS: 92,500 psi YLP: 82,500 psi

CHEMICAL TEST RESULTS

Carbon	0.041
Manganese	1.23
Nickel	0.94
Silicon	0.53
Molybdenum	0.58
Copper	0.01
Phosphorus	0.012
Sulfur	0.015

Elongation in 2 in: 26%
Red of Area: 69.5%

IMPACT PROPERTIES

Specimen Type: Charpy V-Notch
Test Temp: +10°F
Energy (ft-lb): 42, 44, 47
Lateral Expansion (mil): 37, 37, 51
% Shear: 40, 40, 40

OTHER TESTS

Concentricity: 5%
Moisture @ 1800°F: 0.16%

Table 5.3-4 (Cont'd)

(UNIT 1)

Test No.:	242
Trade Name:	Atom Arc 8018NM
Diameter Size:	1/8 in
	2,100 lb
Lot No.:	S411B27AD
Heat No.:	L83355

CHEMICAL TEST RESULTS

Carbon	0.07
Manganese	1.25
Chromium	0.03
Nickel	1.08
Silicon	0.38
Molybdenum	0.53
Copper	0.03
Phosphorus	0.017
Sulfur	0.018
Vanadium	0.02
Aluminum	<0.01

MECHANICAL TEST RESULTS

Stress relieved 50 hr @1150^{o}F

TENSILE PROPERTIES

UTS: 87,600 psi
YLP: 77,900 psi
Elongation in 2 in: 25%
Red of Area: 71.4%

IMPACT PROPERTIES

See page 5.3-41 for impact values

OTHER TESTS

Concentricity: 4%
Moisture @ 1800^{o}F: 0.2%

Table 5.3-4 (Cont'd)

(UNIT 1)

DROP-WEIGHT TESTS	SPECIMEN	TEST TEMPERATURE (^{o}F)	RESULTS
MATERIAL:8018NM	1	-90	Break
LOT: S411B27AD	2	-80	No Break
HEAT: L83355	3	-80	No Break

NDT TEMPERATURE = -90^{o}F

CVN IMPACT TESTS

SPECIMEN	TEST TEMPERATURE (^{o}F)	ENERGY (ft-lb)	LATERAL EXP. (mil)	% SHEAR
1	-105	7	6	5
2	-105	8	7	5
3	-90	19	11	8
4	-90	21	11	10
5	-90	21	13	10
6	-30	27	25	25
7	-30	30	24	25
8	-30	34	29	25
9	-20	31	26	30
10	-20	36	29	30
11	-20	45	37	30
12	-10	51	39	40
13	-10	52	37	40
14	-10	63	52	50
15	+40	112	83	80
16	+40	126	79	80
17	+130	150	91	100
18	+130	154	83	100
19	+130	154	83	100

T_{cv} = -10^{o}F

REFERENCE TEMPERATURE

Material	T_{NDT} (Drop-Weight)	T_{CV} (Charpy V-Notch)	RT_{NDT} (References)
Weld Metal	-90^{o}F	-10^{o}F	-70^{o}F

Table 5.3-4 (Cont'd)

(UNIT 1)

		MECHANICAL TEST RESULTS
Test No.:	374	Stress relieved 50 hr @ 1150^{o}F
Trade Name:	Atom Arc 8018NM	
Diameter Size:	5/32 in 2,000 lb	
Lot No.:	J424B27AE	TENSILE PROPERTIES
Heat No.:	640892	
		UTS: 90,000 psi YLP: 76,500 psi Elongation in 2 in: 27% Red of Area: 71%

CHEMICAL TEST RESULTS

Carbon	0.08
Manganese	1.20
Chromium	0.04
Nickel	1.00
Silicon	0.44
Molybdenum	0.55
Copper	0.09
Phosphorus	0.015
Sulfur	0.018
Vanadium	0.02
Aluminum	0.01

IMPACT PROPERTIES

See page 5.3-43 for impact values

OTHER TESTS

Concentricity: 3%
Moisture @ 1800^{o}F: 0.2%

Table 5.3-4 (Cont'd)

(UNIT 1)

DROP-WEIGHT TESTS	SPECIMEN	TEST TEMPERATURE (^{o}F)	RESULTS
MATERIAL: 8018NM	1	-70	Break
LOT: J424B27AE	2	-60	No Break
HEAT: 640892	3	-60	No Break

NDT TEMPERATURE = -70^{o}F

CVN IMPACT TESTS

SPECIMEN	TEST TEMPERATURE (^{o}F)	ENERGY (ft-lb)	LATERAL % EXP. (mil)	SHEAR
1	-108	14	3	3
2	-108	16	3	3
3	-70	15	8	5
4	-70	20	9	10
5	-70	27	15	10
6	-10	38	26	30
7	-10	42	31	30
8	-10	45	31	30
9	0	55	38	35
10	0	62	44	40
11	0	62	48	40
12	+40	56	42	50
13	+40	75	55	60
14	+130	118	87	100
15	+130	122	89	100
16	+130	130	82	100

T_{cv} = -10^{o}F

REFERENCE TEMPERATURE

Material	T_{NDT} (Drop-Weight)	T_{CV} (Charpy V-Notch)	RT_{NDT} (References)
Weld Metal	-70^{o}F	0^{o}F	-60^{o}F

Table 5.3-4 (Cont'd)

(UNIT 1)

RESULTS		MECHANICAL TEST
Test No.:	261	Stress relieved 50 hr @ 1150°F
Trade Name:	Atom Arc 8018NM	
Diameter Size:	7/32 in 2,400 lb	
Lot No.:	S419B27AG	TENSILE PROPERTIES
Heat No.:	401P6741	UTS: 85,000 psi YLP: 78,000 psi Elongation in 2 in: 30% Red of Area: 73%
CHEMICAL TEST RESULTS		
Carbon	0.06	
Manganese	1.16	
Chromium	0.03	
Nickel	0.92	
Silicon	0.34	
Molybdenum	0.47	
Copper	0.03	IMPACT PROPERTIES
Phosphorus	0.013	
Sulfur	0.014	See page 5.3-45 for impact values
Vanadium	0.02	
Aluminum	<0.01	
		OTHER TESTS
		Concentricity: 3% Moisture @ 1800°F: 0.2%

Table 5.3-4 (Cont'd)

(UNIT 1)

DROP-WEIGHT TESTS	SPECIMEN	TEST TEMPERATURE (°F)	RESULTS
MATERIAL: 8018NM	1	-70	Break
LOT: S419B27AG	2	-60	No Break
HEAT: 401P6741	3	-60	No Break
	NDT TEMPERATURE = -70°F		

CVN IMPACT TESTS

SPECIMEN	TEST TEMPERATURE (°F)	ENERGY (ft-lb)	LATERAL EXP. (mil)	% SHEAR
1	-90	13	8	5
2	-90	14	8	5
3	-70	11	12	10
4	-70	13	14	8
5	-70	16	16	15
6	-10	31	24	25
7	-10	44	30	30
8	-10	76	57	40
9	0	51	37	45
10	0	57	44	40
11	0	68	50	40
12	+40	83	61	50
13	+40	100	80	70
14	+130	136	93	100
15	+130	139	94	100
16	+130	146	94	100

T_{cv} = 0°F

REFERENCE TEMPERATURE

Material	T_{NDT} (Drop-Weight)	T_{CV} (Charpy V-Notch)	RT_{NDT} (References)
Weld Metal	-70°F	0°F	-60°F

Table 5.3-4 (Cont'd)

(UNIT 1)

Test No.:	CN-165
Electrode Specification:	WMS-444F, (Rev 1)
Electrode Type:	CBI 1NMM
Trade Name:	Raco 1NMM
Electrode Diameter:	3/32 in
Heat No.:	5P6756

MECHANICAL TEST RESULTS

Heat Treatment: 1150°F for 50 hr

TENSILE PROPERTIES

Specimen Type: 0.505 in
UTS: 90,500 psi
YLP: 84,000 psi
Elongation in 2 in: 25%
Red of Area: 64.1%

CHEMICAL TEST RESULTS

Carbon	0.13
Manganese	1.89
Chromium	0.08
Nickel	0.96
Silicon	0.07
Molybdenum	0.48
Copper	0.08
Phosphorus	0.008
Sulfur	0.012
Vanadium	0.006
Aluminum	0.02

IMPACT PROPERTIES

Specimen Type: Charpy V-Notch
Orientation: Perpendicular to weld direction
See page 5.3-47 for impact values

Table 5.3-4 (Cont'd)

(UNIT 1)

DROP-WEIGHT TESTS	SPECIMEN	TEST TEMPERATURE (°F)	RESULTS
MATERIAL: C BI 1NMM	1	-40	No Break
HEAT: 5P6756	2	-60	Break
	3	-50	No Break
	4	-50	No Break

NDT TEMPERATURE = -60°F

CVN IMPACT TESTS (@ 1/2T LOCATION)

SPECIMEN	TEST TEMPERATURE (°F)	ENERGY (ft-lb)	LATERAL EXP. (mil)	% SHEAR
1	-20	97	60	60
2	-20	115	75	75
3	-20	105	45	70
4	-20	107	74	65
5	-20	94	65	65
6	0	134	55	100
7	0	121	78	100
8	0	124	75	100

T_{cv} = 0°F

REFERENCE TEMPERATURE

Material	T_{NDT} (Drop-Weight)	T_{CV} (Charpy V-Notch)	RT_{NDT} (References)
Weld Metal	-60°F	0°F	-60°F

Table 5.3-4 (Cont'd)

BELTLINE WELD TOUGHNESS DATA

(UNIT 2)

BELTLINE SHOP WELD TOUGHNESS DATA[1]

WELD IDENTITY	PROCESS	HEAT NO.	FLUX LOT	NDT (°F)	CV TEMP (°F)	CHARPY IMPACT TOUGHNESS ABSORBED ENERGY (ft-lbs)			LATERAL EXP. mil			% SHEAR		
Seams BA,BB,BD,BE,BF	SMAW	432A2671	H019A27A	N/A	+10	31	31	33	30	32	30	30	30	30
					-20	64	42	32	53	34	31	40	35	30
Seams BA, BC	SMAW	03R728	L910A27A	N/A	+10	64	61	72	50	55	56	60	70	70
Seams BA,BB,BC,BD,BE,BF	SAW (SINGLE WIRE)	3P4000	3933	N/A	+10		92,92,95,82,80			81,62,60,72,66			80,80,80,75,80	
Seams BA,BB,BC,BD,BE,BF	SAW (TANDEM WIRE)	3P4000	3933	N/A	+10		90,86,91,87,79			78,73,58,76,71			95,90,98,90,80	
Seam BB	SMAW	401Z9711	A022A27A	N/A	+10	98	99	104	70	69	73	80	70	80
Seam BC					+10	35	38	47	35	31	43	50	50	50
	SMAW	662A746	H013A27A	N/A	-20	89	82	95	69	64	68	45	40	65
Seam BC	SMAW	402A0462	B023A27A	N/A	+10	75	77	86	60	70	60	60	60	62
Seams BD, BE	SMAW	09L853	A111A27A	N/A	+10	78	78	79	60	62	62	70	80	60
Seams BC,BD,BE,BF	SMAW	07L669	K004A27A	N/A	+10	50	50	54	44	44	46	50	50	54
					-20		49, 60, 55, 61, 54			41, 54, 46, 50, 49			40, 40, 35, 35, 35	
Seam AB	SMAW	07L857	B101A27A	N/A	+10	28	36	39	27	41	45	20	40	50
N-17 Nozzle	SMAW	432A2671	H019A27A				73 (Note 2)							
N-17 Nozzle	SMAW	09L853	A111A27A				73 (Note 2)							
N-17 Nozzle	SMAW	07L669	K004A27A				73 (Note 2)							
N-17 Nozzle	SMAW	C3L46C	J020A27A				73 (Note 2)							
N-17 Nozzle	SMAW	422B7201	L030A27A				73 (Note 2)							
N-17 Nozzle	Single/ Tandem	4P4784	3930				73 (Note 2)							

Table 5.3-4 (Cont'd)

(UNIT 2)

WELD IDENTITY	PROCESS	HEAT NO.	FLUX LOT	NDT (°F)	CHARPY IMPACT TOUGHNESS CV TEMP (°F)	ABSORBED ENERGY (ft-lbs)			LATERAL EXP. mil			% SHEAR		
Seam AB	SMAW	402C4371	C115A27A	N/A	+10	82	84	92	62	61	66	80	70	70
Seam AB	SMAW	03M014	C118A27A	N/A	+10	42	44	47	37	37	51	40	40	40
Seam AB	SMAW	411A3531	H004A27A	N/A	+10 -20	60 41	60 68	68 48	51 39	52 53	54 41	60 35	50 35	60 25
Seam AB	SMAW	09M057	C109A27A	N/A	+10	43	43	44	40	41	41	50	60	50
Seam AB	SMAW	L83355	S411B27AD	N/A	See page 5.3-50									
Seam AB	SMAW	640892	J424B27AE	N/A	See page 5.3-52									
Seam AB	SMAW	401P6741	S419B27AG	N/A	See page 5.3-53									
Seam AB	SMAW	412P3611	J417B27AF	N/A	See page 5.3-51									

Table 5.3-4 (Cont'd)

(UNIT 2)

DROP-WEIGHT TESTS	SPECIMEN	TEST TEMPERATURE °F	RESULTS
MATERIAL: 8018NM	1	-90	Break
LOT: S411B27AD	2	-80	No Break
HEAT: L83355	3	-80	No Break
	NDT TEMPERATURE = -90°F		

CVN IMPACT TESTS

SPECIMEN	TEST TEMPERATURE (°F)	ENERGY (ft-lb)	LATERAL EXP. (mil)	% SHEAR
1	-105	7	6	5
2	-105	8	7	5
3	-90	19	11	8
4	-90	21	11	10
5	-90	21	13	10
6	-30	27	25	25
7	-30	30	24	25
8	-30	34	29	25
9	-20	31	26	30
10	-20	36	29	30
11	-20	45	37	30
12	-10	51	39	40
13	-10	52	37	40
14	-10	63	52	50
15	+40	112	83	80
16	+40	126	79	80
17	+130	150	91	100
18	+130	154	83	100
19	+130	154	83	100
		T_{cv} = -10°F		

REFERENCE TEMPERATURE

Material	T_{NDT} (Drop-Weight)	T_{CV} (Charpy V-Notch)	RT_{NDT} (References)
Weld Metal	-90°F	-10°F	-70°F

Table 5.3-4 (Cont'd)

(UNIT 2)

DROP-WEIGHT TESTS	SPECIMEN	TEST TEMPERATURE °F	RESULTS
MATERIAL: 8018NM	1	-90	Break
LOT: J417B27AF	2	-80	Break
HEAT: 412P3611	3	-70	No Break
	4	-70	No Break
	NDT TEMPERATURE = -80°F		

CVN IMPACT TESTS

SPECIMEN	TEST TEMPERATURE (°F)	ENERGY (ft-lb)	LATERAL EXP. (mil)	% SHEAR
1	-100	8	6	3
2	-100	12	10	5
3	-80	15	13	10
4	-80	16	14	10
5	-80	19	15	10
6	-20	52	41	30
7	-20	65	54	50
8	-20	69	53	45
9	+40	100	80	90
10	+40	103	68	80
11	+72	133	91	90
12	+72	138	92	90
13	+130	136	89	100
14	+130	137	95	100
15	+130	146	97	100
		T_{cv} = -20°F		

REFERENCE TEMPERATURE

Material	T_{NDT} (Drop-Weight)	T_{CV} (Charpy V-Notch)	RT_{NDT} (References)
Weld Metal	-80°F	-20°F	-80°F

Table 5.3-4 (Cont'd)

(UNIT 2)

DROP-WEIGHT TESTS	SPECIMEN	TEST TEMPERATURE °F	RESULTS
MATERIAL: 8018NM	1	-70	Break
LOT: J424B27AE	2	-60	No Break
HEAT: 640892	3	-60	No Break
	NDT TEMPERATURE = -70°F		

CVN IMPACT TESTS

SPECIMEN	TEST TEMPERATURE (°F)	ENERGY (ft-lb)	LATERAL EXP. (mil)	% SHEAR
1	-108	14	3	3
2	-108	16	3	3
3	-70	15	8	5
4	-70	20	9	10
5	-70	27	15	10
6	-10	38	26	30
7	-10	42	31	30
8	-10	45	31	30
9	0	55	38	35
10	0	62	44	40
11	0	62	48	40
12	+40	56	42	50
13	+40	75	55	60
14	+130	118	87	100
15	+130	122	89	100
16	+130	130	82	100
		T_{cv} = -10°F		

REFERENCE TEMPERATURE

Material	T_{NDT} (Drop-Weight)	T_{CV} (Charpy V-Notch)	RT_{NDT} (References)
Weld Metal	-70°F	0°F	-60°F

Table 5.3-4 (Cont'd)

(UNIT 2)

DROP-WEIGHT TESTS	SPECIMEN	TEST TEMPERATURE °F	RESULTS
MATERIAL: 8018NM	1	-70	Break
LOT: S419B27AG	2	-60	No Break
HEAT: 401P6741	3	-60	No Break
	NDT TEMPERATURE = -70°F		

CVN IMPACT TESTS

SPECIMEN	TEST TEMPERATURE (°F)	ENERGY (ft-lb)	LATERAL EXP. (mil)	% SHEAR
1	-90	13	8	5
2	-90	14	8	5
3	-70	11	12	10
4	-70	13	14	8
5	-70	16	16	15
6	-10	31	24	25
7	-10	44	30	30
8	-10	76	57	40
9	0	51	37	45
10	0	57	44	40
11	0	68	50	40
12	+40	83	61	50
13	+40	100	80	70
14	+130	136	93	100
15	+130	139	94	100
16	+130	146	94	100
		T_{cv} = 0°F		

REFERENCE TEMPERATURE

Material	T_{NDT} (Drop-Weight)	T_{CV} (Charpy V-Notch)	RT_{NDT} (References)
Weld Metal	-70°F	0°F	-60°F

Table 5.3-4 (Cont'd)

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(1) This table is complemented by Table 5.3-5.

Note 2: An initial Upper Shelf Energy (USE) of 73 ft-lb has been applied for the N-17 nozzle welds. The Safety Evaluation Report for the Measurement Uncertainty Recapture Power Uprate, Section 3.9.3, states that the NRC reviewed the CMTR CVN data for LGS Units 1 and 2, LPCI nozzle (N-17) weld materials and concluded that it would be conservative to assume an initial USE value of 73 ft-lb for the LGS Units 1 and 2 SMAW materials instead of the initial USE values resulting from testing performed at temperatures lower than 40 degrees F.

Table 5.3-5

LGS BELTLINE PLATE WELDS AND FORGINGS EOL RT_{NDT} Based on MUR PU

(Unit 1)

(PEAK EOL FLUENCE = $1.3x10^{18}$ n/cm^2 @ ¼T)

A. Plates

I.D.	Heat	Wt % Cu	Wt % P	Wt % Ni	Wt % S	ASME NB-2300 Start RT_{NDT} (°F)	Regulatory Guide Guide 1.99 (Rev 2) Extrapolation RT_{NDT} (°F)	Estimated EOL RT_{NDT} (°F)
14-1	C7688-1	0.12	0.011	0.51	0.015	+10	72	82
14-2	C7698-2	0.11	0.010	0.48	0.014	+10	69	79
14-3	C7688-2	0.12	0.011	0.51	0.015	+10	72	82
17-1(1)	C7689-1	0.11	0.007	0.48	0.014	+10	69	79
17-2	C7677-1	0.11	0.016	0.50	0.016	+20	69	89(3)
17-3	C7698-1	0.11	0.010	0.48	0.014	+10	69	79

B. Welds

1 Shop Welds (i.e., Vertical Seams)

Heat/Lot	Seams Used In	Wt % Cu	Wt % P	Wt % Ni	Wt % S	ASME NB-2300 Start RT_{NDT} (°F)	Regulatory Guide 1.99 (Rev 2) Extrapolation RT_{NDT} (°F)	Estimated EOL RT_{NDT} (°F)
411A3531/H004A27A	BE	0.02	0.018	0.96	0.017	-50	26	-24
06L165/F017A27A	BA, BD BE, BF	0.03	0.021	0.99	0.017	-50	39	-11
662A746/H013A27A	BA, BD BE, BF	0.03	0.021	0.88	0.017	-20	39	+19
3P4000/(2) 3932-989	BC, BB BA	0.02	0.015	0.935	0.012	-50	26	-24
S3986/(2) Run #934	BF	0.058	0.019	0.949	0.016	-42	75	+33
1P 4218/(1)(2) 3929-989	BE, BA BD	0.058	0.010	0.865	0.011	-50	75	+25
421A6811/(1) F022A27A	Weld Test Plate	0.09	0.018	0.81	0.016	-50	114	+64

Table 5.3-5 (Cont'd)

LGS BELTLINE PLATE WELDS AND FORGINGS EOL RT_{NDT} Based on MUR PU

(Unit 1)

Heat/Lot	Wt % Cu	Wt % P	Wt % Ni	Wt % S	ASME NB-2300 Start RT_{NDT} (□F)	Regulatory Guide 1.99 (Rev 2) Extrapolation RT_{NDT} (°F)	Estimated EOL RT_{NDT} (°F)
2. Field Welds (i.e., Girth)							
07L857/B101A27A	0.03	0.012	0.97	0.017	-6	39	+33
402C4371/C115A27A	0.02	0.009	0.92	0.014	-50	26	-24
411A3531/H004A27A	0.02	0.018	0.96	0.017	-50	26	-24
09M057/C109A27A	0.03	0.009	0.89	0.021	-36	39	+3
412P3611/J417B27AF	0.03	0.016	0.93	0.019	-80	39	-41
03M014/C118A27A	0.01	0.012	0.94	0.015	-34	19	-15
L83355/S411B27AD	0.03	0.017	1.08	0.018	-70	39	-31
640892/J424B27AE	0.09	0.015	1.00	0.018	-60	114	+54[3]
401P6741/S419B27AG	0.03	0.013	0.92	0.014	-60	39	-21
5P6756	0.08	0.008	0.936	0.012	-60	101	+41
3. LPCI Nozzle Welds[4]							
07L669/K004A27A	0.03	0.014	1.02	0.016	-50	14	-36
401Z9711/A022A27A	0.02	0.021	0.83	0.017	-50	9	-41
411A3531/H004A27A							
662A746/H013A27A					Data Previously Provided Under "Shop" Welds		
3P4000/3932-989 S3986/Run #934							
C. Forgings[6]							
Water Level Instrumentation (SB166)	0.11		0.50		20	25	45
LPCI Nozzle forging:							
110L-1(45^{o}) & 110L-2(225^{o}) (Q2Q25W)	0.18		0.85		-6	48	42
110L-4(135^{o}) & 110L-3(315^{o}) (Q2Q35W)	0.18		0.78		-8	47	39

Table 5.3-5 (Cont'd)

LGS BELTLINE PLATE WELDS AND FORGINGS EOL RT_{NDT} Based on MUR PU

(Unit 2)

(PEAK EOL FLUENCE = 1.3×10^{18} n/cm^2 @ ¼T)

A. Plates

I.D.	Heat	Wt % Cu	Wt % P	Wt % Ni	Wt % S	ASME NB-2300 Start RT_{NDT} (°F)	Regulatory Guide 1.99 (Rev 2) Extrapolation RT_{NDT} (°F)	Estimated EOL RT_{NDT} (°F)
14-1	B3312-1	0.13	0.009	0.58	0.016	+10	77	87
14-2	B3416-1	0.14	0.009	0.65	0.015	+40	82	122(3)
14-3	C9621-2	0.15	0.006	0.60	0.020	+22	86	108
17-1(1)	C9569-2	0.11	0.009	0.51	0.018	+10	69	79
17-2	C9526-1	0.11	0.012	0.56	0.018	+10	69	79
17-3	C9526-2	0.11	0.012	0.56	0.018	+10	69	79

B. Welds

1. Shop Welds (i.e., Vertical Seams)

Heat/Lot	Seams Used In	Wt % Cu	Wt % P	Wt % Ni	Wt % S	ASME NB-2300 Start RT_{NDT} (°F)	Regulatory Guide 1.99 (Rev 2) Extrapolation RT_{NDT} (°F)	Estimated EOL RT_{NDT} (°F)
432A2671/H019A27A	BA,BB,BD BE,BF	0.04	0.019	1.08	0.014	-12	51	+39
03R728/L910A27A	BA,BC	0.03	0.020	0.92	0.016	-50	39	-11
3P4000/3933(1) Single and/or Tandem Wire	BA,BB,BC BD,BE,BF	0.02	0.014	0.935	0.012	-50	26	-24
401Z9711/A022A27A	BB	0.02	0.021	0.83	0.017	-50	26	-24
662A746/H013A27A	BC	0.03	0.021	0.88	0.017	-20	39	+19
402A0462/B023A27A	BC	0.02	0.021	0.90	0.018	-50	26	-24
07L669/K004A27A	BC,BD BE,BF	0.03	0.014	1.02	0.016	-50	39	-11
09L853/A111A27A	BD,BE	0.03	0.018	0.86	0.023	-50	39	-11

Table 5.3-5 (Cont'd)

LGS BELTLINE PLATE WELDS AND FORGINGS EOL RT_{NDT} Based on MUR PU

(Unit 2)

Heat/Lot	Wt % Cu	Wt % P	Wt % Ni	Wt % S	ASME NB-2300 Start RT_{NDT} (°F)	Regulatory Guide 1.99 (Rev 2) Extrapolation RT_{NDT} (°F)	Estimated EOL RT_{NDT} (°F)
2. Field Welds (i.e., Girth)							
07L857/B101A27A	0.03	0.012	0.97	0.017	-6	39	+33
402C4371/C115A27A	0.02	0.009	0.92	0.014	-50	26	-24
411A3531/H004A27A	0.02	0.018	0.96	0.017	-50	26	-24
09M057/C109A27A	0.03	0.009	0.89	0.021	-36	39	+3
412P3611/J417B27AF	0.03	0.016	0.93	0.019	-80	39	-41
03M014/C118A27A	0.01	0.012	0.94	0.015	-34	19	-15
L83355/S411B27AD	0.03	0.017	1.08	0.018	-70	39	-31
640892/J424B27AE	0.09	0.015	1.00	0.018	-60	114	+54[3]
401P6741/S419B27AG	0.03	0.013	0.92	0.014	-60	39	-21
3. LPCI Nozzle Welds[4]							
07L669/K004A27A	0.03	0.014	1.02	0.016	-50	14	-36
432A2671/H019A27A	0.04	0.019	1.08	0.014	-12	18	+6
C3L46C/J020A27A	0.02	0.019	0.87	0.017	-20	9	-11
422B7201/L030A27A	0.04	0.013	0.90	0.018	-40	18	-22
09L853/A111A27A	0.03	0.018	0.86	0.023	-50	14	-36
4P4784/3930[5]	0.06	0.012	0.87	0.013	-50	28	-22
4P4784/3930 Tandem Wire	0.06	0.012	0.87	0.013	-20	28	8
4. Surveillance Welds							
CTY538/A027A27A	0.03	0.020	0.83	0.018	-50	39	-11

Table 5.3-5 (Cont'd)

LGS BELTLINE PLATE AND FORGINGS WELDS EOL RT_{NDT} Based on MUR PU

Heat/Lot	Wt % Cu	Wt % P	Wt % Ni	Wt % S	ASME NB-2300 Start RT_{NDT} (°F)	Regulatory Guide 1.99 (Rev 2) Extrapolation RT_{NDT} (°F)	Estimated EOL RT_{NDT} (°F)
C. Forgings[7]							
LPCI Nozzle forging:							
892L-1 (Q2Q33W) (45°)	0.18		0.83		-20	47	27
892L-2 (Q2Q33W) (135°)	0.18		0.81		-6	47	41
892L-3 (Q2Q33W) (225°)	0.18		0.82		-4	47	43
892L-4 (Q2Q33W) (315°)	0.18		0.82		-20	47	27

(1) Surveillance Program Material.

(2) Submerged arc welding.

(3) The most limiting value.

(4) The shell plate and weld are subjected to fluence level in excess of 10^{17} n/cm^2; this information is given in Footnote (1) of Table 5.3-11.

(5) Single wire or tandem wire with submerged arc welding.

(6) The Unit 1 RPV design results in this component experiencing a predicted EOL fluence of 1.9×10^{17} n/cm^2 at ¼ of the thickness. This fluence, based on an assumed Cu content of 0.18% and a measured nickel content of 0.85%, yields an estimated EOL RT_{NDT} of +42°F. The EOL estimate is in accordance with Regulatory Guide 1.99 (Rev. 2).

(7) The Unit 2 RPV design results in this component experiencing a predicted EOL fluence of 1.9×10^{17} n/cm^2 at ¼ of the thickness. This fluence, based on an assumed Cu content of 0.18% and a measured nickel content of 0.82%, yields an estimated EOL RT_{NDT} of +43°F. The EOL estimate is in accordance with Regulatory Guide 1.99 (Rev. 2).

Table 5.3-6

LGS TYPICAL BELTLINE PLATE
(SURVEILLANCE PLATE)

(Unit 1)

Mill Order No: 27265-1

Requirements: SA533 Grade B
Class 1

Melt No: 7689

TEST RESULTS CHEMICAL
(Wt%)

Carbon	0.20
Manganese	1.33
Nickel	0.48
Silicon	0.23
Molybdenum	0.48
Phosphorous	0.007
Sulfur	0.014

MECHANICAL TEST RESULTS

Tensile Properties

UTS: 84,600 psi
85,100 psi

YLP: 63,900 psi
64,400 psi

% Elongation in 2 in: 26
28

Impact Properties

Specimen Type: Charpy V-Notch
Test Temp: +40°F
Energy (ft-lb): 87, 93, 77
75, 86, 81
Lateral Exp. (mil) 73, 69, 62
61, 71, 78
% Shear: 50, 60, 60
50, 60, 60

Drop-weight Test

TEST TEMP (°F)	TOP/BOTTOM RESULTS
+30	1 No Break
+20	1 No Break
+10	1 No Break
0	2 No Break
-10	1 Break
-20	1 Break
-30	1 Break

NDT=-10°F

Table 5.3-6 (Cont'd)

(Unit 1)

Test Location

Drop-Weights - Top and Bottom - Longitudinal
Bend - Top Middle - Transverse
Tensions - Top and Bottom - Transverse
Impacts - Top and Bottom - Longitudinal
Tests ¼T from rolled surface
No closer than "T" from quenched and tempered edge

Specification

ASME SA533 Grade B CL-1 Pressure Vessel Quality

Ultrasonic Testing

Per Procedure LS-UT-4

Heat Treatment

Procedure LS-102 Revision 5

Plates

Austenitized at 1650°F held ½ hr/in (min), and water quenched.
Tempered at 1260°F held ½ hr/in (min), and air cooled.
Stress relieved at 1075°F held 1 hr (min) and air cooled.
Test coupons then cut from plate.

Tests only

Stress relieved @ 1150°F held 50 hr and furnace cooled to below 600°F, then air cooled.

Maximum Heating Rate	Maximum Cooling Rate
100°F/hr	100°F/hr

Mechanical Property Requirements

Tensile:	80/100,000 psi
Yield:	50 min 0.2% offset
Elong:	18% in 2 in (min)
Impacts	30 ft-lb @ +40°F. Lateral Expansion and % Shear Fracture for information only.
Drop-Weights:	Type P-3 specimens with a NDT temperature no higher than +40°F.
Grain-Size:	Final Plate Grain-Size #5 or finer determined on a fully heat treated test coupon.

Table 5.3-6 (Cont'd)

LGS TYPICAL BELTLINE PLATE
(SURVEILLANCE PLATE)

(Unit 2)

Mill Order No: 51090-2

Requirements: SA533 Grade B
Class 1

Melt No: 0569

TEST RESULTS CHEMICAL
(Wt%)

Carbon	0.20
Manganese	1.26
Nickel	0.51
Silicon	0.23
Molybdenum	0.49
Phosphorous	0.009
Sulfur	0.018

MECHANICAL TEST RESULTS

Tensile Properties

UTS: 85,600 psi
88,500 psi

YLP: 65,100 psi
68,100 psi

% Elongation in 2 in: 26
24

Impact Properties

Specimen Type: Charpy V-Notch
Test Temp: +40°F
Energy (ft-lb): 68, 62, 68
83, 87, 66
Lateral Exp. (mil): 45, 50, 47
62, 50, 64

% Shear: 30, 30, 30
40, 40, 40

Drop-weight Test

TEST TEMP (°F)	TOP RESULTS	BOTTOM RESULTS
+10	No Break	No Break
0	No Break	No Break
-10	2	No Break -
-20	Break	2 No Break
-30	-	Break
	NDT = -20°F	NDT = -30°F

Table 5.3-6 (Cont'd)

(Unit 2)

Test Location

Drop-Weights - Top and Bottom - Longitudinal
Bend - Top Middle - Transverse
Tensions - Top and Bottom - Transverse
Impacts - Top and Bottom - Longitudinal
Tests ¼T from rolled surface
No closer than "T" from quenched and tempered edge

Specification

ASME SA533 Grade B CL-1 Pressure Vessel Quality

Ultrasonic Testing

Per Procedure LS-UT-4

Heat Treatment

Procedure LS-102 Revision 5

Plates

Austenitized at 1650°F held ½ hr/in (min), and water quenched.
Tempered at 1260°F held ½ hr/in (min), and air cooled.
Stress relieved at 1150°F held 1 hr (min) and air cooled.
Test coupons then cut from plate.

Tests only

Stress relieved @ 1150°F held 50 hr and furnace cooled to below 600°F, then air cooled.

Maximum Heating Rate	Maximum Cooling Rate
100°F/hr	100°F/hr

Mechanical Property Requirements

Tensile:	80/100,000 psi
Yield:	50 min 0.2% offset
Elong:	18% in 2 in (min)
Impacts:	30 ft-lb @ +40°F. Lateral Expansion and % Shear Fracture for information only.
Drop-Weights:	Type P-3 specimens with a NDT temperature no higher than +40°F.
Grain-Size	Final Plate Grain-Size #5 or finer determined on a fully heat treated test coupon.

Table 5.3-7

SA533 GRADE B, CLASS 1 PLATE TOUGHNESS DATA BASE INCLUDING UPPER-SHELF
(VENDOR - LUKENS STEEL CO.)

Heat No.	NDT (^oF)	Charpy Temp (^oF)	Orientation (L or T)	Energy (ft-lb)	Lateral Expansion (mil)	% Shear
I. PLANT A						
C5978-1	+10	-40	L	7.0, 7.0, 11.0	5, 3, 7	0, 0, 0
		+10		25.0, 33.0, 30.0	23, 25, 23	10, 10, 10
		+40		53.0, 48.0, 48.0	40, 35, 36	20, 20, 20
		+110		118.0, 116.0, 109.0	79, 76, 74	80, 80, 80
		+160		123.0, 136.0, 136.0	82, 84, 84	90, 95, 95
C5978-2	-10	-40	L	22.0, 24.0, 24,0	17, 18, 19	0, 0, 0
		+10		49.0, 46.0, 42.0	38, 36, 33	25, 25, 25
		+40		62.0, 60.0, 41.0	46, 44, 34	35, 35, 35
		+110		98.0, 90.0, 100.0	73, 67, 75	80, 70, 80
		+160		119.0, 120.0, 118.0	88, 86, 82	99, 100, 100
C5979-1	-10	-40	L	9.0, 11.0, 19.0	5, 7, 13	0, 0, 0
		+10		61.0, 57.0, 43.0	45, 41, 32	30, 30, 20
		+40		73.0, 92.0, 65.0	51, 63, 43	35, 45, 35
		+110		117.0, 116.0, 100.0	78, 76, 68	80, 80, 70
		+116		134.0, 136.0, 134.0	87, 86, 85	99, 99, 100
C6345-1	-40	-80	L	8, 6	4, 4	0, 0
		-40		29, 15, 23	21, 13, 16	5, 0, 1
		+10		109, 88, 77	76, 58, 56	50, 35, 35
		+40		103, 96, 122	68, 65, 77	45, 40, 60
		+110		147, 147	84, 82	100, 100
		+160		151, 165	87, 94	100, 100
C6318-1	-20	-40	L	25, 17, 14	18, 11, 9	1, 0, 0
		+10		80, 66, 72	57, 47, 50	35, 30, 30
		+40		85, 95, 112	64, 68, 75	40, 40, 50
		+110		126, 145, 117	81, 89, 76	90, 100, 90
		+160		140, 140, 139	86, 89, 88	100, 100, 100
C6345-2	-40	-80	L	10, 12	7, 9	0, 0
		-40		32, 16, 49	20, 11, 33	10, 0, 20
		+10		93, 94, 67	69, 66, 48	40, 40, 25
		+40		109, 125, 128	70, 72, 82	50, 60, 70
		+110		127, 165	81, 82	85, 100
		+160		153, 161	86, 88	100, 100

Table 5.3-7 (Cont'd)

Heat No.	NDT (^{o}F)	Charpy Temp (^{o}F)	Orientation (L or T)	Lateral Energy (ft-lb)	Expansion (mil)	% Shear
C5979-2	-10	-80	L	8.0, 11.0	5, 9	0, 0
		-40		28.0, 17.0, 18.0	20, 16, 14	15, 10, 10
		+10		64.0, 63.0, 49.0	44, 44, 34	30, 30, 20
		+40		72.0, 76.0, 79.0	47, 49, 50	35, 35, 35
		+110		107.0, 102.0	77, 74	85, 80
		+160		134.0, 141.0	79, 83	100, 100
C5996-1	-10	-40	L	12.0, 53.0, 12.0	10, 20, 11	0, 15, 0
		+10		65.0, 60.0, 77.0	46, 42, 54	30, 30, 40
		+40		88.0, 113.0, 78.0	56, 70, 54	40, 60, 40
		+110		111.0, 126.0, 134.0	74, 85, 83	80, 90, 90
		+160		146.0, 148.0, 143.0	86, 89, 86	100, 100, 100
C6318-1	-40	T	7.5	5.0	1	
		-10		32.5, 31.0	27.0, 28.5	5, 5
		+10		41.5, 42.0	36.0, 37.0	25, 25
		+40		31.5, 60, 49, 63	26.0, 44.0, 34.0,	35, 30, 40, 40 49.0
		+61		70, 71.0	57.2, 57.5	50, 50
		+120		95	70.5	99
		+200		100.0, 91.0, 90.0	75.0, 63.5, 69.5	99, 100, 100
A5333-1	-10	-40	L	21, 13, 11	17, 11, 9	5, 0, 0
		+10		56, 67, 53	41, 47, 40	20, 30, 20
		+40		82, 100, 84	56, 70, 60	40, 50, 40
		+110		126, 120, 133	87, 81, 84	80, 80, 80
		+160		155, 155, 145	92, 90, 89	100, 100, 100
B-0078-1	-10	-40	L	10, 14, 25	10, 13, 21	0, 0, 5
		+10		73, 49, 70	54, 39, 53	40, 30, 40
		+40		94, 100, 100	65, 68, 70	60, 60, 60
		+110		118, 128, 140	82, 86, 89	90, 90, 100
		+160		151, 136, 143	90, 84, 88	100, 100, 100
C6123-2	-10	-80	L	11, 8	9, 7	0, 0
		-40		28, 38, 10	22, 28, 9	10, 10, 0
		+10		77, 60, 73	56, 45, 53	40, 35, 40
		+40		113, 108, 122	73, 71, 75	65, 60, 75
		+110		120, 149	89, 91	90, 100
		+160		149, 151	91, 93	100, 100

Table 5.3-7 (Cont'd)

Heat No.	NDT (^oF)	Charpy Temp (^oF)	Orientation (L or T)	Lateral Energy (ft-lb)	Expansion (mil)	% Shear
C5987-1	-10	-40	L	19, 13, 14	14, 10, 13	0, 0, 0
		+10		63, 55, 35	47, 41, 26	35, 35, 30
		+40		80, 99, 87	59, 68, 61	50, 70, 55
		+110		122, 134, 122	84, 86, 84	100, 100, 100
		+160		122, 134, 127	86, 86, 84	100, 100, 100
C5987-2	-10	-40	L	15.0, 8.0, 10.0	7, 7, 7	0, 0, 0
		+10		76.0, 79.0, 51.0	54, 59, 57	35, 35, 30
		+40		57.0, 76.0, 75.0	42, 57, 54	30, 35, 35
		+110		106.0, 102.0, 113.0	72, 68, 76	80, 80, 80
		+160		140.0, 133.0, 138.0	87, 81, 84	100, 100, 100
C6003-2	-10	-40	L	10.0, 7.0, 8.0	7, 4, 3	0. 0. 0
		+10		37.0, 31.0, 51.0	28, 22, 37	20, 20, 30
		+40		65.0, 49.0, 50.0	44, 34, 36	35, 30, 30
		+110		81.0, 95.0, 82.0	60, 67, 61	60, 70, 60
		+160		121.0, 107.0, 120.0	82, 78, 87	100, 99, 100
C5996-2	-10	-80	L	7.0, 10.0	5, 8	0, 0
		-40		18.0, 25.0, 25.0	14, 20, 19	10, 10, 10
		+10		62.0, 71.0, 66.0	45, 50, 47	20, 30, 25
		+40		81.0, 100.0, 91.0	52, 71, 64	35, 50, 40
		+110		124.0, 130.0	83, 88	90, 90
		+160		149.0, 151.0	89, 91	100, 100
II. PLANT B						
C4882-1	-60	-80	L	16,0, 14.0	15, 11	0, 0
		-40		41.0, 32.0, 54.0	28, 22, 38	20, 20, 25
		+10		75.0, 68.0, 48.0	52, 49, 35	30, 30, 25
		+40		83.0, 95.0, 100.0	59, 65, 70	45, 50, 60
		+110		104.0, 116.0	75, 82	85, 90
		+160		130.0, 131.0	86, 84	100, 100
C4882-2	-40	-80	L	10.0, 7.0	10, 8	0, 0
		-40		46.0, 43.0, 30.0	36, 35, 26	10, 10, 10
		+10		75.0, 50.0, 66.0	58, 44, 52	30, 20, 20
		+40		90.0, 80.0, 88.0	71, 62, 63	40, 35, 40
		+110		120.0, 107.0	84, 82	85, 80
		+160		137.0, 129.0	94, 90	100, 100

Table 5.3-7 (Cont'd)

Heat No.	NDT (^oF)	Charpy Temp (^oF)	Orientation (L or T)	Energy (ft-lb)	Lateral Expansion (mil)	% Shear
C4882-2		-80	T	19.0	15.5	1
		-50		41.0, 25.0, 37.5	29.0, 16.5, 26.0	20, 10, 20
		-30		41.0, 40.0	30.0, 29.5	20, 20
		+10		61.0, 68.0	45.0, 49.5	30, 30
		+39		77.0, 71.0	54.0, 54.5	50, 30
		+70		91.0, 71.0	62.0, 53.0	75, 40
		+121		113.0	79.0	95
		+200		115.5, 113.5	82.5, 74.0	95, 95
III. PLANT C						
C9481-1	-30	+40	L	74, 74, 81	61, 58, 60	50, 50, 50
		+40		103, 61, 85	48, 66, 72	40, 50, 50
C9481-1		-40	T	17.0	15.0	5
		+10		23.5, 22.0	21.0, 20.5	10, 10
		+25		36.0	31.0	20-25
		+40		45.0, 35.0, 42.0	42.0, 34.2, 38.0	30-35, 30, 30-35
		+51		40.5	35.0	30
		+70		51.0, 50.0	44.5, 42.5	40, 40
		+93		71.0	58.5	70
		+120		93.0	69.5	90-95
		+200		93.5, 100.0, 93.0	74.0, 72.0, 69.0	95, 95, 95
IV. PLANT D						
C4574-2	-30	-80	L	8.0, 16.0	6, 13	0, 0
		-40		34.0, 32.0, 27.0	25, 24, 20	10, 10, 5
		+10		48.0, 49.0, 60.0	36, 37, 43	15, 15, 20
		+40		76.0, 63.0, 69.0	56, 47, 51	30, 20, 25
		+110		98.0, 103.0	72, 76	95, 95
		+160		121.0, 119.0	85, 82	100, 100
C4574-2		-20	T	22.0	17.5	1
		+10		32.0, 35.0	22.5, 27.5	5, 5
		+40		50.0, 52.5	35.5, 41.5	10, 10
		+65		64.0, 55.0	47.0, 42.5	30, 30
		+102		75.0	60.5	50
		+119		108.0, 88	75.0, 66.0	100, 85
		+201		112.5	83.5	100
		+202		108.5	79.0	100

Table 5.3-7 (Cont'd)

Heat No.	NDT (°F)	Charpy Temp (°F)	Orientation (L or T)	Energy (ft-lb)	Lateral Expansion (mil)	% Shear
V. PLANT E						
C9533-2	0/10	-50	L	7-12	7-4	1-1
		-30		9-7	7-4	1-1
		-20		9-14[(1)]/19-40[(1)]	11-7[(1)]/5-30[(1)]	10-10/20-20
		+10		34-26/45-47	28-23/36-35	20-20/30-30
		+70		48-64/70-76	48-40/57-60	40-40/40-40
		+100		82-62/82-71	53-65/63-61	60-60/60-60
		+130		72-68/92-86	60-57/72-69	60-60/80-80
		+212		88-81/110-99	75-73/87-81	99-99/99-99
C9570-2 (top)	-40	-70	L	6-4	5-4	1-1
		-50		10-13	20-20	10-10
		-20		14-27	23-14	20-20
		+10		30-42	29-39	30-30
		+70		70-64	56-54	40-40
		+100		76-79	64-63	60-60
		+212		106-108	66-81	90-90
C9570-2 (Bottom)	-50	-70	L	5-7	3-5	1-1
		-40		12-12	16-18	10-10
		-20		36-45	30-36	20-20
		+10		35-46	29-34	30-30
		+40		48-56	47-49	50-50
		+70		79-81	74-75	70-70
		+100		90-110	70-82	80-80
		+130		114-112	86-85	99-99
9570-1	-20 -10	-20	L	11-8[(1)]/13-12[(1)]	8-10[(1)]/12-12[(1)]	1-1/10-10
		-10		20-21/15-14	20-20/15-14	10-10/10-10
		+10		19-19/43-43	18-18/34-33	10-10/30-30
		+40		36-44	37-41	30-30
		+72		60-62/69-70	52-50/56-56	40-40/40-40
		+100		72-75/79-80	65-62/64-65	50-50/60-60
		+130		84-83/94-88	69-71/74-71	70-70/80-80
		+212		96-110/105-103	82-78/81-80	90-90/90-90

(1) Top/Bottom

Table 5.3-8

UPPER-SHELF TOUGHNESS FOR BELTLINE WELDS

Heat No./Flux Lot	NDT (°F)	Charpy Temp (°F)	Wire (S or T)[1]	Energy (ft-lb)			Lateral Expansion (mil)			% Shear		
I. PLANT A												
INMM ELECTRODE (TRADE NAME - TECHALLOY)												
LINDE 124 FLUX, SUBMERGED ARC												
POSTWELD 1150°F FOR 50 HR TYPICAL												
KN203/0171	-80	-130	S	7	6		7	7		5	5	
		-80		34	18	22	32	16	21	40	35	40
		-20		68	70	62	61	57	56	80	70	75
		+10		75	72		64	64		90	90	
		+40		94	82		81	71	100	95		
		+212		94	92	86	76	80	80	100	100	100
	-80	-130	T	7	5		6	5		5	5	
		-100		25	16		24	19		10	10	
		-80		24	22	25	21	19	25	25	20	30
		-20		48	49	54	44	42	46	45	45	60
		-10		59	54	54	48	49	46	60	45	60
		+10		78	67		65	56		95	80	
		+40		80	79		68	68		95	95	
		+212		86	89	87	87	86	85	100	100	100
E8018-G WELD ELECTRODE, SHIELDED METAL ARC												
(TRADE NAME - ATOM ARC 8018 NM)												
POSTWELD 1150°F FOR 50 HR TYPICAL												
640967/D502B27AF	-80	-105		13	14		4	4		5	5	
		-80		16	22	28	11	13	18	10	12	15
		-20		58	76	86	18	42	56	15	50	50
		+40		102	106	119	69	74	86	90	80	90
		+72		119	127			86	82		90	90
		+130		130	140	150	90	92	80	100	100	100

Table 5.3-8 (Cont'd)

Heat No./Flux Lot	NDT (°F)	Charpy Temp (°F)	Wire (S or T)[1]	Energy (ft-lb)			Lateral Expansion (mil)			% Shear		
II. PLANT B												
INMM ELECTRODE (TRADE NAME - RACO) LINDE 124 FLUX, SUBMERGED ARC POSTWELD 1150°F FOR 50 HR TYPICAL												
5P7397/[2]	-70	-70	T	22	16	36	22	18	28	5	5	5
0342		-10		58	68	61	54	50	47	25	20	20
		+10		76	73	75	60	65	60	30	45	50
		+10		75	69		58	56		35	35	
		+40		91	84		75	63		80	85	
		+70		79	75	77	73	63	74	90	95	95
		+212		84	81	87	69	67	75	100	100	100
	-70	-70	S	20	34	27	16	32	22	5	5	5
		-10		54	50	59	47	47	53	25	20	20
		+10		65	59	69	60	56	65	50	25	75
		+10		70	75		56	61		45	55	
		+40		71	78		65	68		75	90	
		+70		92	101	94	82	65	69	95	95	100
		+212		100	95	96	88	58	82	100	100	100
E8018-G WELD ELECTRODE, SHIELDED METAL ARC (TRADE NAME - E8018 NM) POSTWELD 1150°F FOR 50 HR TYPICAL												
401P2871/H430B27AF	-50	-90		7	10		7	11		3	5	
		-70		15	16	16	14	15	16	8	8	10
		-20		66	76	64	58	61	58	15	15	15
		-20		63	81		50	61		15	15	
		-10		27	39	54	25	35	46	35	35	35
		0		27	50	56	25	42	46	40	45	45
		+10		75	76	107	60	62	74	60	50	80
		+40		90	100		71	76		70	80	
		+130		130	140	142	91	94	93	100	100	100
402P3162/H426B27AE	-70	-70		11	7	14	9	6	8	5	5	5
		-40		33	52	32	27	42	22	10	15	10
		-20		65	62	37	52	48	30	20	10	20
		-20		52	55		36	38		15	15	
		-10		60	54	53	44	37	--	40	30	30
		+40		96	99		57	68		60	60	
		+212		119	122	124	93	90	68	100	100	100

Table 5.3-8 (Cont'd)

Heat No./Flux Lot	NDT (°F)	Charpy Temp (°F)	Wire (S or T)[1]	Energy (ft-lb)			Lateral Expansion (mil)			% Shear		
III.PLANT C												
INMM ELECTRODE (TRADE NAME - RACO) LINDE 124 FLUX, SUBMERGED ARC POSTWELD 1150°F FOR 50 HR TYPICAL												
02R486/J404B27AG	-70	-100		12	13		3	5		3	5	
		-90		16	17	19	6	8	7	8	8	10
		-30		17	30	31	15	24	23	15	20	20
		-20		41	42	44	33	34	35	30	30	30
		-10		52	64	66	39	45	46	40	40	40
		+40		84	87		63	68		60	60	
		+130		121	124	129	91	96	95	100	100	100
L83978/J414B27AD	-80	-100		10	12		6	7		4	5	
		-80		14	15	24	10	12	18	10	10	12
		-20		51	52	81	37	40	63	35	50	40
		-20		64	63	69	51	47	55	15	15	15
		-20		67	56		53	45		15	10	
		+40		120	123		72	73		80	80	
		+72		128	140		78	81		90	90	
		+130		148	156	168	90	81	87	100	100	100
5P7397/0156	-50	-70		25	21		18	15		5	5	
		-50		42	27	19	33	25	20	10	15	10
		+10		64	67	55	53	53	52	30	35	40
		+10		64	70		53	54		40	45	
		+40		91	84	85	78	68	79	85	90	95
		+212		103	92	94	59	66	59	100	100	100
3P4966/0342	-80	-80		51	27	9	45	25	12	5	5	5
		-20		71	66	54	57	57	45	30	25	20
		+10		85	84	71	68	72	61	70	80	65
		+10		83	76		67	64		65	55	
		+40		87	91		71	60		75	80	
		+70		100	101	97	82	89	71	90	95	90
		+212		108	111	108	66	84	86	100	100	100

Table 5.3-8 (Cont'd)

Heat No./Flux Lot	NDT (°F)	Charpy Temp (°F)	Wire (S or T)(1)	Energy (ft-lb)			Lateral Expansion (mil)			% Shear		
4P7465/0751	-60	-80		27	14		21	12		5	0	
		-70		48	43	26	42	36	22	15	15	5
		0		63	57	68	54	45	63	30	25	35
		+10		56	58	90	62	62	86	30	25	45
		+10		87	55		83	42		40	30	
		+40		67	97		71	90		45	50	
		+212		118	102	112	88	71	72	100	100	100
1P6484/0156	-20	-80		5	8		6	11		5	5	
		-60		22	16	12	23	13	10	10	10	10
		0		17	36	30	20	27	28	25	20	25
		+10		30	38	17	25	38	12	15	15	15
						34	38	28	30		15	20
		+30		34	46	42	29	37	45	25	50	35
		+40		72	60	72	54	47	49	50	45	50
		+212		93	81	83	65	66	69	100	100	100
5P5657/0931	-60	-80		39	39		27	37		5	5	
		-60		19	20	32	18	22	28	10	10	10
		0		51	55	58	50	50	63	30	30	55
		+10		69	69	66	61	65	59	50	50	40
		+10		62	57		60	63		60	40	
		+40		77	66		73	72		70	80	
		+212		88	91	85	86	75	83	100	100	100
E8018NM WELD ELECTRODE (TRADE NAME - ATOM ARC E8018NM) SHIELDED METAL ARC POSTWELD 1150°F FOR 50 HR TYPICAL												
492L4871/A421B27AE	-60	-108		10	11		5	4		4	4	
		-90		25	30	32	6	6	6	8	10	10
		-30		19	28	31	19	23	25	20	25	25
		-20		22	26	30	23	21	27	25	25	30
		-10		38	41	43	28	32	30	30	30	30
		0		50	51	57	36	38	40	30	40	45
		+40		135	137		84	80		90	80	
		+130		151	160	161	80	82	81	100	100	100

Table 5.3-8 (Cont'd)

Heat No./Flux Lot	NDT (°F)	Charpy Temp (°F)	Wire (S or T)[1]	Energy (ft-lb)			Lateral Expansion (mil)			% Shear		
422K8511/G313A27AD	-80	-90		14	17		15	16		5	5	
		-80		14	16	20	15	16	20	10	10	10
		-40		26	26	40	26	24	33	30	30	30
		-20		65	74	127	44	48	76	40	50	60
		+25		107	108		74	80		80	70	
		+40		125	125	140	84	89	82	100	100	90
		+50		153	143	156	95	81	91	90	80	90
		+68		153	143	165	85	96	91	100	100	100
640892/J424B27AE	-60	-108		14	16		3	3		3	3	
		-70		15	20	27	8	9	15	5	10	10
		-10		38	42	45	26	31	31	30	30	30
		0		55	62	62	38	44	48	35	40	40
		+40		56	75		42	55		50	60	
		+130		118	122	130	87	89	82	100	100	100
40150371/B504B27AE	-60	-60	42	45	23	35	36	20	5	5	5	
		-20		61	84	77	48	66	62	30	25	25
		-20		68	67		51	52		25	25	
		0		80	85	82	63	62	60	35	50	35
		+40		95	97		71	76		40	75	
		+70		111	107	109	87	85	77	80	90	80
		+212		122	114	130	92	92	69	100	100	100
402P3162/H426B27AE	-70	-70		11	7		9	6		5	5	
		-40		33	52	32	27	42	22	10	15	10
		-20		65	62	37	52	48	30	20	10	20
		-20		52	55		36	38		15	15	
		-10		60	54	68	44	37	53	40	30	30
		+40		96	99		57	68		60	60	
		+212		119	122	124	93	90	68	100	100	100

Table 5.3-8 (Cont'd)

Heat No./Flux Lot	NDT (°F)	Charpy Temp (°F)	Wire (S or T)[1]	Energy (ft-lb)			Lateral Expansion (mil)			% Shear		
401P2871/H430B27AE	-50	-90		7	10		7	11		3	5	
		-70		15	16	16	14	15	16	8	8	10
		-10		27	39	54	25	35	46	35	35	35
		0		27	50	56	25	42	46	40	45	45
		+10		75	76	107	60	62	74	60	50	80
		+40		90	100		71	76		70	80	
		+130		130	140	142	91	94	93	100	100	100
07R458/S403B27AG	-60	-70		9	9		7	7		5	5	
		-60		10	11	13	9	9	11	15	10	10
		0		59	61	70	51	52	58	50	50	60
		+40		99	101		77	78		80	75	
		+72		106	110		85	87		80	80	
		+130		129	131	132	81	78	81	100	100	100
03L048/B525B27AF	-60	-105		8	9		2	3		3	3	
		-80		10	16	19	7	10	11	10	10	10
		-20		31	50	65	22	37	50	30	30	30
		-10		36	53	58	34	43	45	40	40	40
		0		61	75	79	44	58	59	50	60	60
		+40		104	108		75	77		80	80	
		+130		122	123	126	89	83	91	100	100	100

1) Single or Tandem
(2) This material is in Plant B's vessel surveillance program.

Table 5.3-9

WELD PROCEDURE SPECIFICATION FOR VESSEL MATERIAL
REPRESENTATIVE OF LGS BELTLINE WELDS(1)

(Unit 1)

Reference Specifications

General WPS 800 Latest Revision
General WPS 820 Latest Revision

Procedure Qualification

No.	Position	Thickness Range (inches)
1890 (SMA)	V	3/16 to 8
1891 (SMA)	H	3/16 to 8
1892 (SMA)	OH, F	3/16 to 8
1893 (SA1)	F	3/16 to 8
2200 (SA2)	F	3/16 to 8

Postheat Treatment

Procedure qualified with 50 hours at 1150°F + 25°F/-50°F.

Postweld heat treatment of the weldment shall be in accordance with a Chicago Bridge and Iron approved procedure.

Base Metal

ASME SA533 Grade B Class 1 or SA508 Class 2
ASME Group No. P12B Subgroup 1

Shielding Gas: None

Backup Gas: None

Flux: Linde 124

Preheat Requirements

Minimum preheat of 300°F shall be applied uniformly to the full thickness of the weld joint and adjacent base material for a minimum distance of "T" or 6 inches, whichever is least, where "T" is the material thickness.

Maintain 300°F min preheat temperature until start of postweld heat treatment except for longitudinal and circumferential shell and head seams, preheat may be dropped to 250°F min 8 hours after completion of welding. All turnoff tabs and flux dams must be removed prior to dropping preheat below 300°F.

Table 5.3-9 (Cont'd)

Interpass Temperature Requirements

The interpass temperature shall not exceed 500°F maximum.

Filler Metal

Submerged Arc

Specification - N.A.
Classification - N.A.
Analysis - A3 (except Ni 0.50 to 1.25)
Usability - F6
Trade Name - CBI 1NMM (1% Nickel) or equal

Shielded Metal Arc

Specification - SFA5.5
Classification - E8018-G
Analysis - A3 (except Ni 0.50 to 1.25)
Usability - F4
Trade Name - Alloy Rods E8018NM

Electrical Characteristics

SMA - DCRP
Submerged Arc
  Tandem Wire
    Lead Wire - DCRP
    Trail Wire - AC
  Single Wire - DCRP

(1) This specification is extracted from the surveillance program of another BWR plant with similar beltline weld material.

Table 5.3-10

TYPICAL SURVEILLANCE PROGRAM WELD PROCEDURE FOR
LGS AND OTHER BWRs

Reference Specifications

General WPS 800 Latest Revision
General WPS 820 Latest Revision

Procedure Qualification

No.	Position	Thickness Range (in)
963 (TW)	F (Sub Arc) F,V,H (SMA)	4½ to 9.9
1261 (SW)	F (Sub Arc) F,V (SMA)	2¾ to 8

Postheat Treatment

Procedure qualified with 50 hours at 1150°F + 25°F/-50°F.

Postweld heat treatment of the weldment shall be in accordance with Chicago Bridge and Iron approved procedure.

Base Metal

ASME SA533 Grade B Class 1 or SA508 Class 2
ASME Group P12B Subgroup 1

Shielding Gas: None

Backup Gas: None

Flux: Linde 124

Table 5.3-10 (Cont'd)

Preheat Requirements

Minimum preheat of 300°F shall be applied uniformly to the full thickness of the weld joint and adjacent base material for a minimum distance of "T" or 6 inches, whichever is less where "T" is the material thickness.

Maintain preheat temperature until start of postweld heat treatment.

Interpass Temperature Requirements

The interpass temperature shall not exceed 500°F maximum.

Filler Metal

Submerged Arc

Specification - NA
Classification - NA
Analysis - A3 (except Ni 0.50 to 1.25)
Usability - F6
Trade Name - Adcom 1NMM (1% nickel) or equal

Shielded Metal Arc

Specification - SA316
Classification - E8018-G
Analysis - A3 (except Ni 0.50 to 1.25)
Usability - F4
Trade Name - Alloy Rods E8018NM

Electrical Characteristics

SMA - DCRP

Submerged Arc
Tandem Wire
Lead Wire - DCRP
Trail Wire - AC
Single Wire - DCRP

Table 5.3-10 (Cont'd)

GENERAL WELDING TECHNIQUE

Operation Description	Beads Layer	Weld Proc.	Electrod Size (in)	Current Type	Voltage (Amps)	(volts)	Travel
SMA	As Req'd	SMA	1/8	E8018NM	90-135	23-25	
			5/32		110-160	24-26	
			3/16		150-220	24-26	
			7/32		250-350	25-27	
			1/4		300-400	25-27	
Submerged Arc Single Wire (DCRP)	As Req'd	[1]		Adcom 1NMM or Equal	550-650	28-32	10-18
Tandem Wire	As Req'd	Lead	[1]		650-750	32-36	24 min
		Trail	[1]		550-650	34-37	
GROOVES[2]							
SMA	As Req'd	SMA	1/8	E8018NM	90-135	23-25	
			5/32		110-160	24-26	
			3/16		150-220	24-26	
			7/32		250-350	25-27	
			1/4		300-400	25-27	
Submerged Arc Single Wire (DCRP)	As Req'd		[1]	Adcom 1NMM or Equal	450-700	28-35	8-18

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[1] 5/32 or 3/16 inches
[2] See Figure 5.3-8

Table 5.3-11

ESTIMATED RT_{NDT} FOR COMPONENTS IN LGS
VESSEL NONBELTLINE REGION

	Component	Material	RT_{NDT}(°F)
I. Unit 1			
	1. Vessel Flange	SA508 Class 2	-20
	2. Top Head Flange	SA508 Class 2	+10
	3. Top Head Torus	SA533 Grade B, Class 1	+10
	4. Plate Connecting to Vessel Flange	SA533 Grade B, Class 1	+20
	5. Feedwater Nozzle	SA508 Class 2	-20
	6. Vessel Main Closure Stud(2)	SA540 Grade B 24	--
II. Unit 2			
	1. Vessel Flange	SA508 Class 2	+10
	2. Top Head Flange	SA508 Class 2	+10
	3. Top Head Torus	SA533 Grade B, Class 1	-20
	4. Plate Connecting to Vessel Flange	SA533 Grade B, Class 1	-16
	5. Feedwater Nozzle	SA508 Class 2	0
	6. Vessel Main Closure Stud(2)	SA540 Grade B 24	--

Table 5.3-11 (Cont'd)

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(1) Deleted

(2) This component meets the CVN test requirements of 45 ft-lb absorbed energy and 25 mil lateral expansion at +10°F.

(3) Deleted

Table 5.3-12

SURVEILLANCE CAPSULE[1]

Capsule	Tensile	Charpy V-Notch
No. 1 (Azimuth 300°)	3 Base Metal (BM) 4 Weld Metal (WM) 3 Heat Affected Zone (HAZ)	8 BM, Long. 8 WM 8 HAZ
No. 2 (Azimuth 120°)	3 BM 4 WM 3 HAZ	8 BM, Long. 8 WM 8 HAZ
No. 3 (Azimuth 30°)	3 BM 4 WM 3 HAZ	20 BM, Long. 16 WM 12 HAZ

(1) Each capsule also includes Fe and Cu flux wires .

5.4 COMPONENT AND SUBSYSTEM DESIGN

5.4.1 REACTOR RECIRCULATION PUMPS

5.4.1.1 Safety Design Bases

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

a. An adequate fuel barrier thermal margin is ensured during postulated transients.

b. A failure of piping integrity does not compromise the ability of the reactor vessel internals to provide a refloodable volume.

c. The system maintains 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:

a. The system provides sufficient flow to remove heat from the fuel.

b. The system provides for changing reactor power without control rod movement over the range of approximately 55% to 100% rated power.

c. The system design minimizes 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 (Figure 5.4-1 and drawing M-43). Each external loop contains one high capacity, variable-speed, motor-driven recirculation pump and two motor-operated gate valves (for pump maintenance). Each loop contains a flow-measuring system. The recirculation loops are part of the RCPB and are located inside the drywell structure. The jet pumps are reactor vessel internals. Their location and mechanical design are discussed in Section 3.9.5. However, certain operational characteristics of the jet pumps are discussed in this section. The important design and performance characteristics of the reactor recirculation system are shown in Table 5.4-1. The head, NPSH, flow, and efficiency curves are shown in Figure 5.4-3. An instrumentation and control description is provided in Section 7.1.2.

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 pumps 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 (Figure 5.4-4). The adequacy of the total flow to the core is discussed in Section 4.4.

The allowable heatup rate for the recirculation pump casing is the same as for the reactor vessel. If one loop is shut down, it 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 due to the resulting draining of reactor coolant, 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 back-seats 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 NPSH available beyond that provided by the pump location below the reactor vessel water level. If feedwater flow is less than the minimum value necessary for 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 NDTT limit. The vessel is heated by operating the recirculation pumps and/or by core decay heat.

Each recirculation pump is a single-stage, variable-speed, vertical, centrifugal pump equipped with mechanical shaft seal assemblies. The pump is capable of stable and satisfactory performance while operating continuously at any speed corresponding to a power supply frequency range of 11.5 Hz 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 that 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% probability factor (original design). Each individual seal in the cartridge is capable of sealing against pump operating pressure so that any one seal can adequately limit leakage if the other seal fails. A breakdown orfice is provided in the pump casing to reduce leakage if there is 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. It should be noted that the original seals have been replaced with an improved design prone to less leakage in the event of a seal failure.

Each recirculation pump motor is a vertical, variable-speed, ac electric motor that can drive the pump over a controlled range of 28% to 99.2% of rated pump speed. The motor is designed to operate continuously at any speed within the power supply frequency range of 15.7 Hz to 55.6 Hz from 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 ASD located outside the drywell supplies power to a recirculation pump motor. Each unit has two variable-frequency ac ASDs (one per recirculation pump motor). The pump motor is electrically connected to the ASD 4kV output generated by digitally controlled solid-state power electronics.

The ASD has no rotational inertia to add to the recirculation pump and motor to slow down the coast-down rate. However, per the GEH Transient Analysis Report (SDOC G-080-VC-00467 and SDOC G-080-VC-00470), it is determined that this lack of rotational inertia was either 1) not a factor in the transient, 2) had low consequence in the analysis or 3) was otherwise analyzed as not being a significant contributor to the event. The lack of additional coast-down inertia is not a concern for flow into the core.

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. Removable parts of the pump such as wear rings, impellers, bearings, etc, are designed for as long a life as practicable, 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 a refueling 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. Design codes and standards are discussed in detail in Section 3.2.

Snubbers located at the top of the motor and at the bottom of the pump casing are designed to resist seismic 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 the postulated pipe breaks 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 is in Section 3.6.

The recirculation system piping, valves, and pump casings are covered with fiberglass-type thermal insulation and stainless steel jacketing. This insulation is removable for the purpose of inservice inspection.

5.4.1.4 Safety Evaluation

Postulated reactor recirculation system malfunctions that could lead to damage to the fuel barrier are described and evaluated in Chapter 15. It is shown in Chapter 15 that none of the malfunctions could result in significant fuel damage. The recirculation system has sufficient flow coast-down 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 ECCS topical report (Reference 5.4-1). The ability to reflood the BWR core to the top of the jet pumps is shown schematically in Figure 5.4-5 and is discussed in Reference 5.4-1.

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-retaining components are chosen in accordance with applicable codes (Section 3.2). Use of the code design criteria ensures that a system designed, built, and operated within design limits has an extremely low probability of failure caused by any known failure mechanism.

GE purchase specifications require that the recirculation pump's first critical speed be not less than 130% of operating speed. Calculations were verified by GE design engineering.

GE purchase specifications require that the 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 is required to be such that the dynamic load capability at rated operating conditions is not exceeded during the SSE. Calculation submittal is required.

Pump overspeed occurs during the course of a postulated LOCA due to blowdown through the broken loop pump. Design studies determined that the overspeed is not sufficient to cause destruction of the motor, and consequently no provision is made to decouple the pump from the motor for such an event.

5.4.1.5 Inspection and Testing

Quality control methods were used during fabrication and assembly of the reactor recirculation system to ensure that design specifications are met. Inspection and testing were carried out as described in Section 3.9. The RCS was thoroughly cleaned and flushed before fuel was loaded initially.

During the preoperational test program, the reactor recirculation system was hydrostatically tested in accordance with the codes listed in Section 5.2.1. Preoperational tests on the reactor recirculation system also include checking the operation of the pumps, flow control system, and gate valves and are discussed in Chapter 14.

During the startup test program, horizontal and vertical motions of the reactor recirculation system piping and equipment were observed; supports were adjusted, if necessary, to ensure that components are free to move as designed. Nuclear system responses to recirculation pump trips at rated temperatures and pressure were evaluated during the startup tests, and plant power response to recirculation flow control was determined.

5.4.2 STEAM GENERATORS (PWR)

Section 5.4.2 is not applicable to LGS.

5.4.3 REACTOR COOLANT PIPING

The reactor coolant piping is discussed in Sections 3.9.3 and 5.4.1. The recirculation loops are shown in Figure 5.4-1 and drawing M-43, and the design characteristics are presented in Table 5.4-1.

5.4.4 MAIN STEAM LINE FLOW RESTRICTORS

5.4.4.1 Safety Design Bases

The main steam line flow restrictors are designed to:

a. Limit the loss of coolant from the reactor vessel following a steam line 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 MSIVs

b. Withstand the maximum pressure difference expected across the restrictor, following complete severance of a main steam line

c. Limit radioactive release outside of the drywell before MSIV closure

d. Provide a trip signal for MSIV closure

5.4.4.2 Description

A main steam line flow restrictor (Figure 5.4-6) is provided for each of the four main steam lines. The restrictor is a complete assembly welded into the main steam line, and it is located in the drywell.

If a main steam line break occurs outside the containment, the restrictor limits the coolant blowdown rate from the reactor vessel to the maximum (choke) flow of $7.08x10^6$ lb/hr at 1000 psig upstream pressure. The restrictor assembly consists of a venturi-type nozzle insert welded, in accordance with applicable code requirements, into the main steam line. 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 steam line 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 steam line inside diameter of approximately 0.55 results in a maximum pressure differential (unrecovered pressure) of about 11 psi at 105% of rated flow. This design limits the steam flow in a severed line to 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 MSIVs when the steam flow exceeds preselected operational limits.

5.4.4.3 Safety Evaluation

If a main steam line breaks outside the containment, the critical flow phenomenon would restrict the steam flow rate in the venturi throat to 200% of the rated value. Before 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 a steam line rupture accident (Chapter 15) shows that the core remains covered with water and that the amount of radioactive materials released to the environment through the main steam line break does not exceed the guideline values of published regulations.

Tests on a scale model determined the 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 about 0.2% moisture flowing at velocities of 150 ft/sec (steam piping ID) to 600 ft/sec (steam restrictor throat). ASTM A351 (Type 304) cast stainless steel is used for the 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 erodes 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 erode more rapidly than a smooth surface. Experience shows that a machined or a ground surface is sufficiently smooth and that no detrimental erosion occurs.

5.4.4.4 Inspection and Testing

Because the flow restrictor forms a permanent part of the main steam line piping and has no moving components, no testing program is planned. Only very slow erosion occurs with time, and such a slight enlargement has 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 600-900 ft/sec. However, calculations show that, even if the erosion rates are as high as 0.004 inch per year, after 40 years of operation the increase in restrictor choked flow rate would be no more than 5%. A 5% increase in the radiological dose calculated for the postulated main steam line break accident is not significant.

5.4.5 MAIN STEAM LINE ISOLATION SYSTEM

5.4.5.1 Safety Design Bases

The MSIVs, individually or collectively, perform the following functions:

a. Close the main steam lines within the time established by DBA analysis to limit the release of reactor coolant.

b. Close the main steam lines slowly enough that simultaneous closure of all steam lines does not induce transients that exceed the nuclear system design limits.

c. Close the main steam line when required despite single failure in either a valve or in the associated controls, to provide a high level of reliability for the safety function.

d. Use separate energy sources as the motive force to close independently the redundant isolation valves in the individual steam lines.

e. 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.

f. Close the steam lines, either during or after seismic loadings, to ensure isolation if the nuclear system is breached.

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

5.4.5.2 <u>Description</u>

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.

Figure 5.4-7 shows an MSIV. Each is a 26 inch Y-pattern, globe valve. The rated steam flow rate through each valve is 3.543×10^6 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 permit flow areas past the wide open poppet that are greater than the seat port area. The poppet travels approximately 90% of the valve stem travel to close the main seat port area; approximately the last 10% of valve stem travel closes the pilot valve. The gas (air for MSIVs outside containment, nitrogen for MSIVs inside containment, as described in Section 9.3) 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° angle permits the inlet and outlet passages to be streamlined; this minimizes pressure drop during normal steam flow and helps prevent blockage by debris. The pressure drop at 105% of rated flow is 7 psi maximum.

Attached to the upper end of the stem is a gas 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 seconds and 10 seconds.

The gas cylinder is supported on the valve bonnet by actuator support and spring guide shafts. Helical springs around the spring guide shafts close the valve on loss of pneumatic pressure to the gas cylinder. The motion of the spring seat member actuates switches in the 10% open, 90% open valve positions.

The valve is operated by pneumatic pressure and by the action of compressed springs. The control unit is attached to the gas cylinder. This unit contains three types of control valves (pneumatic, dc, 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 gas is supplied to the valves from the Instrument Air (outboard valves) and Primary Containment Instrument Gas (inboard valves) systems. A gas accumulator between the control valve and a check valve provides backup operating gas. The actuator and seals act as extensions of the accumulator and as such have specific leakage limits.

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 steam line ruptures downstream of the valve, the steam flow would quickly increase to 200% 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 (excluding exercise cycles) are estimated to be 50 cycles per year during the first year and 20 cycles per year thereafter.

In addition to the minimum wall thickness required by applicable codes, a 0.120 inch (minimum) corrosion allowance is added to provide for 40 years' service.

Design specification ambient conditions for normal plant operation are 150°F maximum temperature and 100% relative 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 MSIVs are designed to close under accident environmental conditions of 340°F for one minute at drywell design pressure. In addition, they are designed to remain closed under the following postaccident environment conditions:

a. For the balance of the first 3 hours; 340°F at a drywell pressure of 45 psig maximum

b. 320°F for an additional 3 hours at 45 psig maximum

c. 250°F for an additional 24 hours at 25 psig maximum

d. 200°F during the next 100 days at 20 psig maximum

To resist sufficiently the response motion from the SSE, the MSIV installations are designed as seismic Category I equipment. The valve assembly is manufactured to withstand the SSE forces applied at the mass center of the extended mass of the valve operator, assuming that 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 a percentage of the allowable yield stress for the material. The parts of the MSIVs that constitute a process fluid pressure boundary are designed, fabricated, inspected, and tested as required by the ASME Section III. Design codes and standards are discussed in detail in Section 3.2.

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 environment through process openings in the steam system, or they can 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 steam line break outside the containment is described in Chapter 15. The analysis shows that the fuel barrier is protected against loss of cooling if MSIV 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 seconds) of the MSIVs is also shown to be satisfactory in Chapter 15. The switches on the valves initiate reactor scram when specific conditions (extent of valve closure, number of pipelines included, and reactor power level) are exceeded (Section 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 is insignificant. No fuel damage results.

The ability of this 45°, Y-design globe valve to close in a few seconds after a steam line 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:

a. To verify its capability to close at settings between 3 seconds and 10 seconds, each valve is tested at a pressure of 1000 psig and no flow (response time for full closure is set before plant operation at 3 seconds minimum, 5 seconds maximum). The valve is stroked several times, and the closing time is recorded. The valve is closed by springs only and by the combination of air cylinder and springs. The closing time is slightly greater when closure is by springs only.

b. Leakage is measured with the valve seated and back-seated. 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. (Valve stem backseat feature and valve packing leak off feature has been eliminated on all of the MSIVs by modification.)

c. 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. The tests include radiographic, liquid penetrant, or magnetic particle examinations of casting, forgings, welds, hard-facings, and bolts.

d. The spring guides, the guiding of the spring seat member on support shafts, and rigid attachment of the seat member ensure 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.

Each valve is tested after installation as discussed in Chapter 14.

Two isolation valves provide redundancy in each steam line so that 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 MSIV 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 stresses or prevent the valve from closing as required.

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

5.4.5.4 Inspection and Testing

The MSIVs can be functionally tested for operability during plant operation. The test provisions are listed below. During refueling outages the MSIVs can be functionally tested, leak tested, and visually inspected.

During plant operation the MSIVs can be tested and exercised individually to the 90% open position, because the valves still pass rated steam flow when 90% open.

During prestartup tests following an extensive shutdown, the valves receive the same hydro tests (approximately 400 psi) that are imposed on the primary system.

The MSIVs 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 steam line flow restrictors in the remaining steam lines. Continued operation with one or both MSIV's closed on one Main Steam Line is permitted provided that reactor thermal power is maintained at or below 75% of rated (ref. 5.4-3).

Leakage from the valve stem packing can be detected 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 valve seats (pilot and poppet seats) is measured accurately as a part of the 10CFR50, Appendix J LLRT program during shutdown.

Leak rates are determined by typical Appendix J, Type C, leak rate procedures which use a combination of flow-in, flow-out, pressure decay, and pressure differential elimination across valves to determine valve leakage at its test pressure required by technical specifications.

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

5.4.6 REACTOR CORE ISOLATION COOLING SYSTEM

5.4.6.1 Design Bases

The RCIC system is a safety system that consists of a turbine, pump, piping, valves, accessories, and instrumentation designed to ensure 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:

a. The vessel is isolated and maintained in the hot standby condition.

b. The vessel is isolated, accompanied by loss of coolant flow from the reactor feedwater system.

c. A complete plant shutdown under conditions of loss of normal feedwater system is started before the reactor is depressurized to a level at which the shutdown cooling system can be placed into operation.

Following a reactor scram, steam generation continues at a reduced rate due to the core fission product decay heat. At this time the turbine bypass system diverts the steam to the main condenser, and the feedwater system supplies the makeup water required to maintain the reactor vessel inventory.

If 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 drops due to continued steam generation by decay heat. On reaching a predetermined low level, the RCIC system is initiated automatically. The system then functions to restore adequate vessel water levels. On reaching a predetermined high level, the RCIC steam admission valve (F045) automatically closes, resulting in turbine shutdown. RCIC will automatically restart if the level returns to the low level trip point. The turbine-driven pump supplies makeup water from the CST to the reactor vessel. If the CST level falls below a predetermined low level, an alternate source of water is automatically made available from the suppression pool. The turbine is driven with a portion of the decay heat steam from the reactor vessel, and exhausts to the suppression pool. Suppression pool water may not be of condensate quality and hence it is preferred that it only be used if sources of condensate quality water are not available.

During RCIC operation, the suppression pool acts as the heat sink for steam generated by reactor decay heat. This results in a rise in pool water temperature. The RHR system heat exchangers are used to maintain pool water temperature within acceptable limits by cooling the pool water.

For design basis events RCIC needs to operate for a maximum of six hours in order to fulfill its safety functions.

5.4.6.1.1 Residual Heat and Isolation

5.4.6.1.1.1 Residual Heat

The RCIC system is designed to initiate and discharge, within 55 seconds, a specified constant flow into the reactor vessel over a specified pressure range. The temperature of the RCIC water discharged into the reactor vessel varies between 40°F and 140°F. Station Blackout (SBO) procedures directs the operation of the RCIC System for Reactor Pressure Level Control for the SBO 4-hour coping duration with water supply from the Suppression Pool. During the SBO event, the Suppression Pool water may reach up to 180 degrees F. The mixture of the cool RCIC water and the hot steam accomplishes the following:

a. Quenches the steam

b. Removes reactor residual heat by reducing the heat level (enthalpy) due to the temperature difference between the steam and water

c. Replenishes reactor vessel inventory

The HPCI system performs the same function, thereby providing single failure protection. Both systems use different electrical power sources of high reliability, which permits operation with either onsite or offsite power. Additionally, the RHR system performs a residual heat removal function.

The RCIC system design includes interfaces with redundant leak detection devices:

a. A high pressure drop across a flow device in the steam supply line equivalent to 300% of the steady-state steam flow at 1197 psia

b. A high area temperature, using temperature switches as described in the leak detection system; high area temperature is alarmed in the control room

c. A high pressure between the turbine exhaust rupture diaphragms

d. Reactor low pressure

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

Other isolation bases are defined below. Again, HPCI provides redundancy for RCIC if RCIC becomes isolated, thus providing single failure protection.

5.4.6.1.1.2 Isolation

Isolation valve arrangements include the following (additional information about containment isolation is given in Section 6.2.4):

a. Two RCIC lines penetrate the primary containment and form a part of the RCPB. The first is the RCIC steam supply line, which branches off the B main steam line between the reactor vessel and the inboard MSIV. This line has two automatic motor-operated isolation valves F007 and F008, which are key-locked open. One is located inside and the other outside primary containment. An automatic MOV, F076, is provided in the bypass line around the outboard RCIC isolation valve. The isolation signals noted earlier close these valves. The isolation signal is automatic and bypasses the key-lock when the valves must be closed in the case of an RCIC line break. For other accidents, it is more desirable to have steam available for RCIC operation than to preclude its operation because of a containment automatic isolation valve closure signal. If the isolation valves were closed, operator action would be required to reopen the valves to avoid water hammer and thermal shock. An isolation signal is given for a large pipe break by detecting flow rates greater than 300% of the steady-state steam flow. For leakage with flow rates less than

300% of steady-state steam flow, an isolation signal is signaled by use of area temperature sensors provided by the leak detection system.

The RCIC pump discharge line is the other line that forms a part of the RCPB; however, it indirectly connects to the RPV. Outside containment, this line enters the feedwater line, which provides required isolation valves inside the primary containment. The RCIC system has an automatic MOV F013 outside primary containment for isolation. The above arrangements satisfy GDC 55 for RCPB lines penetrating containment.

b. The RCIC turbine exhaust line vacuum breaker system line has two automatic MOVs (F080, F084) and four check valves. This line runs between the suppression pool air space and the turbine exhaust line downstream of the exhaust line check valve. Isolation is automatic via a combination of low reactor pressure and high drywell pressure. This design satisfies GDC 56 for primary containment isolation.

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

c. The RCIC pump suction line, minimum flow pump discharge line, turbine gland seal system vacuum pump discharge, and turbine exhaust line all penetrate the primary containment and terminate below the suppression pool water level. The isolation valves (F031, F019, F002 and F060, respectively) for the lines are all outside primary containment and require remote manual operation, except for the minimum flow valves, which operate automatically. This arrangement satisfies GDC 56 for primary containment isolation.

5.4.6.1.2 Reliability, Operability, and Manual Operation

5.4.6.1.2.1 Reliability and Operability

The RCIC system as shown 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 is tested during both the startup and preoperational 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 periodically performed.

A design flow functional test of the RCIC system may be performed during normal plant operation by drawing suction from the CST and discharging through a full flow test return line to the condensate storage tank. The discharge valve to the feedwater line remains closed during the test, and reactor operation remains undisturbed. All components of the RCIC system are capable of undergoing individual functional testing during normal plant operation. The system automatically aligns from test to operating mode if system initiation is required, but operator action is needed in the following instances:

a. Auto/manual switch is in manual on the flow controller. This feature is provided for operator flexibility during system operation.

b. Steam inboard/outboard isolation valves closed. Closure of either or both of these valves requires operator action to properly sequence their opening. An alarm sounds when either of these valves leaves the fully open position.

c. Parts of the system have been bypassed or otherwise deliberately rendered inoperable. These conditions are automatically indicated in the control room at the system level. Capability for manual initiation of system level indication exists for items not readily automated.

See also Section 5.4.6.2.4.

Four RCIC lines have a low design pressure and could possibly be overpressurized: the turbine exhaust line, the turbine leak-off to the barometric condenser, the cooling water line from PCV-1F015 to the barometric condenser, and the vacuum pump discharge line. These lines are shown on drawings M-49 and M-50. The turbine exhaust line is protected from overpressurization by two rupture discs, PSE-1D001 and PSE-1D002. Pressure relief valves are provided to protect the other three lines from overpressurization.

5.4.6.1.2.2 Manual Operation

Provisions are included for remote manual startup, operation, and shutdown of the RCIC system.

As discussed in Section 5.4.7.1.1.5, the steam condensing mode of the RHR system has been removed from the plant.

See also Section 5.4.6.2.4.

5.4.6.1.3 Loss of Offsite Power

The RCIC system electrical power is derived from a highly reliable source that is maintained by either onsite or offsite power. See Section 5.4.6.1.1.

5.4.6.1.4 Physical Damage

The system is designed to the requirements shown in Table 3.2-1, which are commensurate with the safety importance of the system and its equipment. The RCIC system is located in a different area of the reactor enclosure and uses different divisional power (with separated electrical routings) than its redundant system (HPCI), as discussed in Sections 5.4.6.1.1 and 5.4.6.2.3.

5.4.6.1.5 Environment

The RCIC suction line that is exposed to the outdoor environment is provided with non-Class 1E heat tracing. Indication is provided in the control room if this heat tracing should become inoperative. Condensate level instrumentation for control room monitoring is located in the reactor enclosure and therefore is not exposed to the outdoor environment.

5.4.6.2 System Design

5.4.6.2.1 General

5.4.6.2.1.1 Description

A summary description of the RCIC system is presented in Section 5.4.6.1, which defines in general the system functions and components. The detailed description of the system, its components, and its operations is presented in the following sections.

5.4.6.2.1.2 Diagrams

The following diagrams are included for the RCIC system:

a. A P&ID (drawings M-49 and M-50) shows all components, piping, points where interface system and subsystems tie together, and instrumentation and controls associated with subsystem and component actuation.

b. A schematic process diagram (drawing E51-1020-G-002) shows temperature, pressures, and flows for RCIC operation and system process data hydraulic requirements.

5.4.6.2.1.3 Interlocks

The following electrical interlocks are provided:

a. There are four key-locked valves (F007, F008, F060, and F002) and two key-locked resets (the isolation resets).

b. Limit switches on valves F029 and F031 activate such that when both valves are fully open, F010 closes.

c. A limit switch on valve F060 activates when fully open and clears a permissive so valve F045 can open.

d. The limit switch on valve F045 actuates when F045 is partially open. The limit switch causes the valve to stop at the partially open position and also initiates a time delay relay. The valve remains in the partially open position until the time delay relay times out and activates the opening of the F045 valve. The time delay relay also initiates the ramp generator.

e. A limit switch on valve F045 activates when F045 is not fully closed, initiates startup ramp function and acts to lock out the following alarms for 15 seconds: RCIC pump low flow, RCIC low oil pressure, and RCIC vacuum tank low vacuum. This ramp resets each time F045 is closed.

f. The F045 limit switch activates when fully closed to permit valves F004, F005, F025, and F026 to open and causes valves F013 and F019 to close.

g. The turbine trip throttle valve (part of the turbine assembly) limit switch activates when fully closed and causes valves F013 and F019 to close.

h. The combined pressure switches at reactor low pressure and high drywell pressure, when activated, close valves F080 and F084.

i. High turbine exhaust pressure, low pump suction pressure, or an isolation signal, actuate to close the turbine trip throttle valve; when the signal is cleared, the trip throttle valve must be reset from the control room.

j. 124% overspeed trips both 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

k. An isolation signal closes valves F007, F008, F076, and other valves, directly or indirectly, as noted in items f and h above.

l. An initiation signal opens valves F010 (if closed), F013, and F046; starts the barometric condenser vacuum pump; and causes valve F022 to close, if open.

m. A high RCIC steam line drain pot level signal causes valve F054 to open. The valve recloses when the high level signal clears.

n. The combined signal of low flow plus pump discharge pressure opens and, with increased flow, closes valve F019. See also items e and f above.

o. The switches for reactor low pressure, high turbine exhaust diaphragm pressure, steam line high differential pressure, or high area temperature, when activated, close valves F007, F008 and F076.

p. Limit switches on valves F029 and F031 activate such that when either valve is open, F022 closes.

q. High water level in the reactor vessel (level 8) initiates closure of the F045 and FO46 valves.

5.4.6.2.2 Equipment and Component Description

5.4.6.2.2.1 Design Conditions

The RCIC system components are as follows:

a. One 100% capacity turbine and accessories

b. One 100% capacity pump assembly and accessories

c. Piping, valves, and instrumentation for the following:

    1. Steam supply to the turbine

    2. Turbine exhaust to the suppression pool

    3. Makeup supply from the CST to the pump suction

    4. Makeup supply from the suppression pool to the pump suction including pump suction strainers described in Section 6.2.2.2.

5. Deleted

6. Pump discharge to the feedwater line, a test line to the CST, a minimum flow bypass line to the suppression pool, and a coolant water supply to accessory equipment.

5.4.6.2.2.2 Applicable Codes and Classifications

The RCIC system components within the drywell up to and including the outer isolation valve are designed in accordance with the ASME Section III, Class 1. The RCIC system is also designed as seismic Category I equipment. The RCIC electrohydraulic system integrated with the turbine governing valve is a safety-grade design, specified for seismic Category I design.

Other RCIC system component classifications are given in Table 3.2-1.

5.4.6.2.3 System Reliability Considerations

To ensure that the RCIC operates 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. The instrumentation design for the RCIC system is such that no failure of a single initiating sensor either prevents or falsely starts the system.

To ensure HPCI or RCIC availability for the operational events noted previously, certain design considerations are used in the design of both systems:

a. Physical independence. The two systems are located in separate rooms in the secondary containment. Piping runs are separated, and the water delivered from each system enters the reactor vessel via different nozzles.

b. Prime mover diversity and independence. Prime mover independence is achieved by using separate steam lines to drive the HPCI and RCIC steam turbines. Additionally, separate divisions of electrical power are used for HPCI and RCIC.

c. Control independence. Control independence is obtained by using different battery systems to provide control power to each system. Separate detection initiation logics are also used for each system.

d. Environmental independence. The safety-related equipment in the RCIC compartment does not rely upon auxiliary system support.

e. Periodic testing. A design flow functional test of the RCIC system is performed during plant operation by taking suction from the CST and discharging through the full flow test return line back to the CST. The discharge valve to the reactor feedwater line remains closed during the test, and reactor operation is undisturbed. Control system design provides automatic alignment from test to operating mode if system initiation is required during testing.

f. General. Periodic inspections and maintenance of the turbine-pump units are conducted in accordance with manufacturers' instructions. Valve position indication and instrumentation alarms are provided in the control room.

5.4.6.2.4 System Operation

Manual actions required for the various modes of RCIC are discussed in system operating and test procedures.

5.4.6.2.4.1 Automatic Operation

Automatic startup of the RCIC system due to an initiation signal from reactor low water level requires no operator action. Preparation of the system for the standby mode and manual actions required during operation and shutdown are defined by system operating procedures.

5.4.6.2.4.2 Test Loop Operation

This operating mode is manually initiated by the operator. Operator action is required as defined by surveillance test procedures and system operating procedures.

During all modes of manual MOV operation using the handswitch from the main control room, a "dead zone" is present for a portion of the valve travel. The "dead zone" is present when the valve is not fully closed and green light only indication exists. If the valve is stopped in the "dead zone", operator action is required to restart the valve.

5.4.6.2.4.1 Deleted

5.4.6.2.4.2 Deleted

5.4.6.2.4.3 Steam Condensing (Hot Standby) Operation

As discussed in Section 5.4.7.1.1.5, all the components which make up the steam condensing mode of the RHR system with the exception of an isolated vent line off each of the 2 Unit 2 RHR heat exchangers have either been abandoned in place or physically removed from the plant. Therefore, the mode is no longer functional.

5.4.6.2.4.4 Limiting Single Failure

The most limiting single failure of the high pressure recovery systems, HPCI and RCIC, is the failure of HPCI. If the capacity of the RCIC system is adequate to maintain reactor water level, the operator follows the station operating procedures. However, if the RCIC capacity is inadequate, the station operating procedures still apply, but additionally the ADS may be initiated as described in Section 6.3.2.

5.4.6.3 Performance Evaluation

The analytical methods and assumptions used in evaluating RCIC system performance are presented in Chapter 15. The RCIC system provides the flows required from the analysis (drawing E51-1020-G-002) within a 55 second interval based on considerations noted in Section 5.4.6.2.3.

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 BOP/NSSS safety interfaces for the RCIC system are:

a. Preferred water supply from the CST.

b. All associated wire, cable, piping, sensors, and valves that are outside the NSSS scope of supply.

c. Air supply for testable check and solenoid-actuated valves.

5.4.7 RESIDUAL HEAT REMOVAL SYSTEM

5.4.7.1 Design Bases

The RHR system is comprised of four independent loops. Each loop contains a motor-driven pump, piping, valves, instrumentation, and controls. Each loop takes suction from the suppression pool and is capable of discharging water to the reactor for low pressure coolant injection vessel via a separate vessel nozzle or back to the suppression pool via a full flow test line. In addition, loops A and B have heat exchangers that are cooled by RHRSW. These two loops can also take suction from the reactor recirculation system suction or from the spent fuel pool during refueling and can discharge into the reactor recirculation system discharge for residual heat removal. One of these two loops can be aligned to cool the spent fuel pool. The pumps in loops C and D can be aligned via crossties for use as alternates to the pumps in loops A and B, respectively. During cold shutdown and refueling operation condition, this results in the availability of four shutdown cooling subsystems (A heat exchanger with A RHR pump; A heat exchanger with C RHR pump; B heat exchanger with B RHR pump; and B heat exchanger with D RHR pump).

The Fire Protection system can be cross-tied to Loop B (Unit 1) and Loop A (Unit 2) RHR to provide an alternate source of water for the containment (drywell) spray mode of RHR. This cross-tie can only be used when there is no other method of injection to containment spray. Also, the cross-ties can provide an alternate source of water that can be injected through the LPCI injection line.

As discussed in Section 5.4.7.1.1.5, all the components comprising steam condensing mode of the RHR system have ether been abandoned in place or physically removed from the plant. Therefore, the mode is no longer functional.

5.4.7.1.1 Functional Design Basis

Each of the RHR modes has it's own functional requirements. Each subsystem is discussed separately below.

5.4.7.1.1.1 Residual Heat Removal Mode (Shutdown Cooling Mode)

a. The functional design basis of the shutdown cooling mode is to have the capability to remove decay and sensible heat from the reactor primary system so that the reactor outlet temperature is reduced to 125°F, approximately 20 hours after the control rods have been inserted, and to permit refueling when the RHRSW temperature is 85°F, assuming that the core is "mature" and the RHR heat exchanger tubes are completely fouled (see Section 5.4.7.2.2 for exchanger design details). The capacity of the heat exchangers is such that the time to reduce the vessel outlet water temperature to 212°F results in a cooldown rate in excess of 100°F per hour with both loops in service. However, the flushing operation associated with shutdown prevents the attainment of 212°F coolant temperature at a continuous 100°F per hour rate.

Assuming 2 hours are used for flushing the system before operation, with all systems available the minimum time required to reduce vessel coolant temperature to 212°F is depicted by Figure 5.4-11.

b. The plant can be shut down using the capacity of a single RHR heat exchanger and related RHRSW system capability. Figure 5.4-12 shows the minimum time required to reduce vessel coolant temperature to 212°F using one RHR heat exchanger in the shutdown cooling mode and allowing 2 hours for flushing.

c. Each RHR heat exchanger can be aligned to one of two associated RHR pumps, constituting a shutdown cooling subsystem comprised of a heat exchanger, pump, and piping flow path. This results in the availability of four shutdown cooling subsystems since each subsystem can be considered operable for shutdown cooling if it can be aligned (remote or local) for removal of decay heat.

d. The RHR heat exchangers can be aligned in the RHR Alternate Decay Heat Removal (ADHR) method when flooded-up during refueling. The heat exchangers are aligned to the spent fuel pool skimmer surge tanks and flow is returned to the reactor vessel. The functional design basis is to maintain the reactor and fuel pool water temperatures below 140°F while providing greater maintenance flexibility. In addition, the FPCC system(s) and the RHR fuel pool assist mode can be used for decay heat removal. See Section 9.1.3.1 for further details.

5.4.7.1.1.2 <u>Low Pressure Coolant Injection Mode</u>

The functional design basis for the LPCI mode is to pump 10,000 gpm of water per loop, using the separate loop pumps from the suppression pool into the core region of the vessel when the vessel pressure is 20 psid over drywell pressure. Injection flow commences at 295 psid vessel pressure above drywell pressure.

The initiating signals are reactor vessel low water level (level 1) or high drywell pressure coincident with low reactor pressure. The pumps attain rated speed in less than 40 seconds after their initiation signal, and the injection valves are in their fully open position in less than 40 seconds after the initiation signal.

The LPCI mode is discussed in greater detail in Section 6.3.

5.4.7.1.1.3 <u>Suppression Pool Cooling Mode</u>

The functional design basis for the suppression pool cooling mode is that it has the capacity to maintain the suppression pool temperature immediately after a blowdown below 170°F when the reactor pressure is above 135 psig.

5.4.7.1.1.4 <u>Containment Spray Cooling Mode</u>

The functional design basis for the containment spray cooling mode is that there are two (redundant) means to spray into the drywell and suppression pool vapor space to maintain internal pressure below the design limits.

5.4.7.1.1.5 Reactor Steam Condensing Mode

The reactor steam condensing mode is no longer functional. With the exception of an isolated vent line off each Unit 2 heat exchanger, components required for reactor steam condensing mode have been either abandoned in place or physically removed from the plant.

5.4.7.1.2 Design Basis for Isolation of RHR System from Reactor Coolant System

In the absence of a valid LOCA signal, 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 Section 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 piping pressure rating (see Section 5.2.5 for an explanation of the LDS and the isolation signals).

The RHR pumps are protected against damage from a closed discharge valve by the automatic minimum flow valves in the recirculation lines, which open on low main line flow and close on high main line flow. The minimum flow valve opens at main line flows of less than approximately 15% of pump rated flow (1500 gpm); this allows flow to return to the suppression pool through the low resistance low flow bypass line which branches off the main line upstream of the flow element.

The minimum flow valve closes at main line flows greater than approximately 1500 gpm; this closes the low resistance low flow bypass to the suppression pool and forces the entire pump discharge flow through the main line.

Under certain preplanned valve test scenarios, it is necessary and permissible to temporarily operate the RHR pumps at shutoff without minimum flow protection. These preplanned tests are conducted under explicit administrative controls. Continuous RHR pump operation at shutoff would eventually result in pump damage.

The minimum flow valve, valve control meets IEEE 279 requirements on the ECCS network level.

The minimum flow line restricting orifice is Quality Group B (i.e. seismic Category I, ASME Section III). The piping is rated at 300 lb (ANSI Primary Rating).

5.4.7.1.3 Design Basis for Pressure Relief Capacity

The relief valves in the RHR system are sized on one of two bases:

a. Thermal relief

b. Valve leakage

Transients are treated by item a; item b results from an excessive leak past isolation valves. Valves F025, F029 and F030 are set at or below the design pressure of the associated piping.

An interlock prevents the opening of valves to the low pressure suction piping when the reactor pressure is above the shutdown range. The same interlock initiates valve closure on increasing reactor pressure.

An additional interlock is provided for valve F008 to prevent a fire-induced valve open signal from causing both valves to open simultaneously while the reactor pressure is greater than the design capabilities of the RHR low-pressure piping.

In addition, check valves in the discharge lines to the vessel prevent reverse flow from the reactor if the reactor pressure increases above the RHR system pressure. Relief valves in the discharge piping are sized to account for leakage past the check valves.

5.4.7.1.4 Design Basis with Respect to GDC 5

With the exception of the fuel pool structures as discussed below, the RHR system for one unit does not share equipment or structures with the other unit. The RHRSW and ESW systems that provide cooling water to the RHR heat exchangers, and pump motor oil coolers, respectively, are common to both units. During OPCON 5, the spent fuel pools can be cross-tied with RHR on the shutdown unit taking suction from the reactor cavity or fuel pool. However, as discussed in Sections 3.1, 9.1.3.2.3, 9.2.2, and 9.2.3, this cannot compromise the ability of the RHR system to perform its safety-related functions.

5.4.7.1.5 Design Basis for Reliability and Operability

The shutdown cooling mode of the RHR system can be controlled by the operator from the control room. The only operations required to be performed outside of the control room for a normal shutdown are manual operation of keep fill valves (condensate transfer) to prevent addition to the reactor, and local alignment of the C and D RHR pumps to the respective A and B RHR heat exchangers for operation of the C and D subsystems of shutdown cooling.

The use of the RHR ADHR method will require the installation of the RHR/FPC spool piece, the manual alignment of several valves, and adjustment of the fuel pool overflow weirs.

Two separate shutdown cooling heat exchanger loops are provided, each with two alignable RHR pumps, and although both heat exchanger loops may be employed for shutdown, the reactor coolant can be brought to 212°F in approximately 20 hours with only one heat exchanger loop in operation. Interties are provided between the suction and discharge lines of the RHR pump in the direct injection LPCI loop (C and D pumps) and the suction and discharge lines of the associated RHR pump in the heat exchanger loop (A and B pumps, respectively) to allow use of the C and D pumps in the shutdown cooling mode, thus providing greater maintenance flexibility. During the cold shutdown and refueling operation conditions, this results in the existence of four shutdown cooling subsystems, each comprised of one of the two heat exchangers with one of its two alignable RHR pumps (A heat exchanger with A RHR pump; A heat exchanger with C RHR pump; B heat exchanger with B RHR pump; and B heat exchanger with D RHR pump). With the exception of the shutdown suction and shutdown return lines, the entire RHR system is part of the ECCS and containment cooling systems and is therefore required to be designed with the 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 powered from both offsite and onsite emergency power for isolation and shutdown following a LOOP. If either of the two shutdown supply valves fails to operate, the design basis states that an operator is sent to open the valve by hand. If this is not feasible, the shutdown line is isolated using manual valve F077 and repairs are made to the shutdown valves so that they can be opened to supply shutdown suction to the RHR pumps. 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 (see Section 5.4.7.5 for a discussion of an alternate shutdown cooling flow path).

To increase the reliability of RHR shutdown cooling mode during refueling outages, the automatic isolation function of the RHR shutdown cooling mode supply and return valves is typically bypassed provided that automatic isolation is not required by the Technical Specifications or Technical Requirements Manual and the reactor cavity is flooded up. Manual isolation capability is retained.

The RHR ADHR block valve (051-1193 and 051-2193) will be normally locked open to provide additional assurance that the valve is not inadvertently closed. Closure of this valve has no impact on ESF functions but could delay the start of shutdown cooling mode.

A non-limiting single active failure is a loss of a shutdown cooling return valve or testable check valve. In this case the other RHR heat exchanger loop is used, or if two required operable shutdown cooling subsystems are on the same heat exchanger loop and manual actions to restore valve operability are not successful, cooling water flow may be returned to the vessel through the LPCI injection line.

5.4.7.1.6 Design Basis for Protection from Physical Damage

Evaluation of the RHR system with respect to the following areas is discussed in the indicated sections:

a.	Protection from wind and tornado effects	Section 3.3
b.	Flood design	Section 3.4
c.	Missile protection	Section 3.5
d.	Protection against dynamic effects associated with the postulated rupture of piping	Section 3.6
e.	Seismic events	Section 3.7
f.	Environmental design	Section 3.11
g.	Fire protection	Section 9.5

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 (drawing M-51). A description of the controls and instrumentation is presented in Section 7.3 and 7.4.

The RHR process diagram (drawing E11-1020-G-002) contains both the process diagram and process data. All of the sizing modes of the system are shown in the process data. The FCD for the RHR system is provided in Section 7.3.

Interlocks are provided to prevent:

a. Draining the vessel water to the suppression pool during shutdown.

b. Opening the vessel suction valves above the suction line design pressure or the discharge line design pressure, with the pumps at shutoff head.

c. Inadvertent opening of the drywell spray valves while in shutdown.

d. A pump start when the suction valve(s) is not open.

NOTE: The pump start interlock when the suction valve(s) is not open is defeated during operation of certain modes of RHR.

5.4.7.2.2 Equipment and Component Description

a. System pumps:

The RHR pumps are motor-driven deep-well pumps with mechanical seals and cyclone separators. The motors are air cooled by the ventilating system and by lube oil coolers. The pumps are sized on the basis of the LPCI mode (Mode A) and the minimum flow mode (Mode G) in drawing E11-1020-G-002. Design pressure for the pump suction is 220 psig, with a temperature range from 40°F to 360°F. Design pressure for the pump discharge is 500 psig. The bases for the design temperature and pressure are maximum shutdown cut-in pressures and temperature, minimum ambient temperature, and maximum shutoff head. The pump pressure vessel is carbon steel, and the shaft and impellers are stainless steel. Available NPSH is calculated according to Regulatory Guide 1.1 and is greater than the required NPSH for all operating modes. NPSH calculations for the LPCI mode are provided in Section 6.3.2.2.4.1.

The reliability of the RHR pumps at LGS is enhanced by their "deep draft" design, which is based on many years of experience in industrial and power plant application. While their use in nuclear power plant application is more recent, the same proven design principles have been used in successful industrial applications.

The RHR pumps are designed for the life of the plant (40 years) and are tested for operability assurance and performances as follows:

i. In-shop tests including: (1) hydrostatic tests of pressure-retaining parts at 150% of the design pressure; (2) performance tests while the pump is operated with flow to determine the total developed head at zero flow and design flow; and (3) NPSH requirements.

ii. After the pump is installed in the plant, it undergoes the: (1) system hydro tests; (2) functional tests; and (3) required periodic inservice inspection of approximately once every three months for approximately an hour during normal plant operation, and approximately one month of operation each year for shutdown.

iii. In addition, the pumps are designed for a postulated single operation of 100 days for one accident during the unit's 40 year life.

The following shows the maximum expected accumulated operating time for the life of the plant (40 years):

Mode of Operation	RHR (days)
In-Shop Test	1
Preoperation	3
Monthly Testing	9.2
Yearly Testing	1.7
Post-LOCA	365
Shutdown	2520
	2899.9*

Furthermore, Ingersoll-Rand pumps similar to the ones installed at LGS have been installed in operating plants and have accumulated significant successful operating experience.

* The above information provided on maximum accumulated RHR pump operating time for the life of the plant is historical and is based on original plant design conditions. The evaluation did not include any expected RHR pump operating time to maintain Suppression Pool temperature within specified operating limits during plant operation. Suppression Pool cooling operation to offset heat addition (such as HPCI runs or SRV leakage) up to 10% cumulative RHR runtime per unit per year is evaluated as acceptable use and is offset by shutdown usage (due to shorter outage lengths than the original evaluation).

A summary of experience of Ingersoll-Rand RHR pumps currently available to GE through 1981 is as follows:

			RHR (hours)
Hatch	RHR Pump	2A	864
		2B	1112
		2C	629
		2D	569
Chinshan 1	RHR Pump		100
Chinshan 2	RHR Pump		75

Maintenance was not required or performed during the running times listed above.

No problems have been reported on these pumps.

A comparison of the pumps cited for operating experience and the LGS pumps is as follows:

	HATCH	CHINSHAN	LGS

Column Length	RHR	241"	187"	138"
RPM	RHR	1800	1800	1200

Thus, it may be noted that the LGS pumps are conservatively represented by the operating experience in that they are shorter than the operating pumps and run at a slower speed. Both of these features are conservative because they make the pump less susceptible to vibration that causes degradation and wear. In this context, it should also be noted that the LGS pumps are much shorter than the deep draft pumps that have experienced vibration problems in the past.

b. Heat exchangers:

The RHR system heat exchangers are sized on the basis of the duty for the shutdown cooling mode (Mode E of the process data). All other uses of these exchangers require less cooling surface.

Flow rates are 10,000 gpm (rated) on the shell side and 9,000 gpm (rated) on the tube side (service water side). The rated inlet temperature is 125°F shell side and 85°F tube side for the shutdown cooling mode. The overall heat transfer coefficient is 225 Btu/hr-ft^2-°F. The exchangers contain 6,281 ft^2 (Unit 1 and Unit 2A) and 6,073.56 ft^2 (Unit 2B) of effective surface. The design temperature range of both shell and tube sides is 40°F to 470°F. The design pressure is 450 psig on both sides, and original design fouling factors are .0005 (Unit 1 & 2) shell side and 0.0018 (Unit 1), and 0.002 (Unit 2A) and 0.001 (Unit 2B) tube side. The construction materials are carbon steel for the pressure vessel with stainless steel tubes and stainless steel clad tube sheet. The condition of the heat exchangers is controlled by administrative and test procedures to assure that the required heat removal capability is maintained. Refer to Section 6.2 for heat removal requirements.

c. Valves:

All of the directional valves in the system are conventional gate, globe, butterflies and check valves designed for nuclear service. The injection valves and reactor coolant isolation 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. ECCS portions of the RHR system:

The ECCS portions of the RHR system include those sections described through Mode A-1 of drawing E11-1020-G-002.

The route includes suppression pool suction strainers, suction piping, RHR pumps, discharge piping, injection valves, and drywell piping into the vessel nozzles and core region of the reactor vessel.

Pool cooling components include pool suction strainers, suction piping, pumps, heat exchangers, and pool return lines. The suction strainers are described in Section 6.2.2.2.

Containment spray components 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 prevent the components and piping from inadvertent overpressure conditions. The relief valve setpoint, capacity, and method of collection are shown in Table 5.4-3.

5.4.7.2.4 Applicable Codes and Classifications

a. Piping, pumps, and valves:

1.	Process side	ASME III	Class 2
2.	Service water side	ASME III	Class 3

b. Heat exchangers

1.	Process side	ASME III (Refer to Table 3.2-3)	Class C
2.	Service water side	ASME VIII	Division 1

c. Electrical portions

1. IEEE 279

2. IEEE 308

5.4.7.2.5 Reliability Considerations

The ECCS portion of the RHR system includes the redundancy requirements of Section 5.4.7.1.5. Two redundant heat exchanger loops are provided to remove residual heat, each with two alignable RHR pumps powered from separate emergency buses. Either heat exchanger loop is capable of cooling down the reactor within a reasonable length of time. During cold shutdown and refueling operation conditions, this results in the availability of four shutdown cooling subsystems. The shutdown cooling suction line from the recirculation suction piping is common to all four shutdown cooling subsystems, and each heat exchanger and its associated discharge piping is common to its two alignable RHR pumps (A and C RHR pumps for the A heat exchanger, B and D RHR pumps for the B heat exchanger). When aligned in the shutdown cooling mode of operation, each heat exchanger and its discharge piping are passive components that are assumed not to fail.

5.4.7.2.6 Manual Action

a. Residual heat removal (shutdown cooling mode)

To align a loop of the RHR system for shutdown cooling, the minimum flow and suppression pool suction valves are closed to prevent inadvertent drainage of reactor water to the suppression pool. The RHRSW pump on the selected loop is started. A preoperational flush of the RHR suction and most of the discharge piping is performed to prewarm piping and reactor cavity water clarity by flushing reactor water through it. During flushing the reactor water is discharged to the suppression pool through the test return line in a controlled manner. After verifying that the discharge lines are full, the RHR pump is started. The operator controls the cooldown rate by regulating the reactor coolant flow through the heat exchanger by using the heat exchanger outlet bypass valve, the heat exchanger bypass valve, and the heat exchanger outlet valve in various combinations. The total flow can be throttled with the shutdown cooling return outboard isolation valve. All of these operations except initial alignment of a LPCI dedicated RHR pump (C or D pump) for shutdown cooling use can be performed from the control room.

The operator determines the cooldown rate by monitoring temperature change with time. When the process computer is available, the operator can assign the process computer points to use in the trend function. The computer then drives a strip-chart recorder with the data from the computed points. The operator can display temperature/time information graphically by calling up information on the process computer and manually plotting the temperature information versus time. The operator can monitor temperature directly on recorders that provide a permanent record of cooldown transient information.

The manual actions required for the most limiting failure are discussed in Section 5.4.7.1.5.

b. Steam condensing mode

As discussed in Section 5.4.7.1.1.5, all components comprising the steam condensing mode of the RHR system have either been abandoned in place or physically removed from the plant. Therefore, the mode is no longer functional.

5.4.7.3 Performance Evaluation

The thermal performance of the RHR heat exchangers is based on the residual heat generated at 20 hours 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 (95°F) less 10°F is used as the service water inlet temperature (i.e., 85°F). 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 generated due to the following:

a. 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

b. The fouling condition of the exchangers

c. Operator use of one or two cooling loops

d. Service water temperature

e. System flushing time

Since the exchangers are designed for the fouled condition with relatively high service water temperature, the heat exchangers have excess capability to cool when first cut-in at high vessel temperatures. The total flow and mix temperature must be controlled to avoid exceeding a 100°F per hour cooldown rate. See Section 5.4.7.1.1.1 for the minimum shutdown time to reach 212°F.

5.4.7.3.2 Shutdown with Most Limiting Failure

Shutdown under conditions of the most limiting failure is discussed in Section 5.4.7.1.1.1. 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 are used to generate data to verify the operational capabilities of each piece of equipment in the system: each instrument, each setpoint, each logic element, each pump, each heat exchanger, each valve, and each limit switch. In addition these programs 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. Logic elements are tested electrically; valves, pumps, controllers, and relief valves are tested mechanically; finally, the system is tested for total system performance against the design requirements as specified above using both the offsite power and standby emergency power. Preliminary heat exchanger performance can be evaluated by operating in the pool cooling mode, but a vessel shutdown is required for the final check due to the small temperature difference available with pool cooling.

5.4.7.5 Conformance to Regulatory Guide 1.139

Although the LGS design was completed before the issuance of this guide, which provides guidance on shutdown cooling mode design, the LGS design satisfies the intent of the guide, subject to the following clarifications:

a. Provisions of shutdown cooling, assuming the most limiting single active failure (loss of the common suction line due to valve failure to open), as discussed in paragraph C.1 of the regulatory guide, is accomplished by an alternate flow path. In the alternate method the RHR pump pumps from the suppression pool to the reactor vessel via the RHR heat exchanger. Flow returns from the vessel to the suppression pool via manually opened ADS valves. The LGS ADS valves are three-stage Target Rock valves. Their operability under the fluid conditions expected to occur during the alternate shutdown cooling mode has been demonstrated by generic testing described in topical report NEDE-24988-P/NEDO-24988.

The test results documented in the topical report verify the adequacy of the LGS valve operation and integrity under the expected liquid discharge conditions. The loads on the valve and piping induced by the liquid discharge were shown to be lower than the high pressure steam discharge loads for which the system is designed. The test results also provide flow capacity information, which shows that sufficient shutdown cooling flow is provided through 1 or 2 valves.

A safety-grade pneumatic supply is available for ADS valve actuation, as discussed in Section 9.3.1. A variation of this method is to pump to the vessel with a Core Spray Loop and operate the RHR system in the suppression pool cooling mode.

b. Regarding paragraph C.2.a of the regulatory guide, which discusses reactor high pressure interlocks and alarms, LGS conforms with the intent in that the two suction valves are interlocked with reactor pressure; two out of two low reactor pressure signals must be present to permit opening. On a high pressure signal, the valves close and the pump trips. A pump trip activates an "RHR system out of service" alarm that annunciates in the main control room. There is no high pressure alarm per se. Loss of power to the valve control logic causes suction valve closure and pump trip.

   An additional interlock is provided for valve F008 to prevent a fire-induced valve open signal from causing both valves to open simultaneously while the reactor pressure is greater than the design capabilities of the RHR low-pressure piping.

c. Pressure relief is not provided for a pressure transient during operation as discussed in paragraph C.3 of the regulatory guide; however, no rapid pressurization mechanism is known for a BWR. Pressure may increase gradually, and the automatic high pressure valve interlocks (closed on high pressure) are considered adequate for this case.

d. The RHR design is considered to conform to the pump protection discussion in paragraph C.4 of the regulatory guide, in that the pump motor is equipped with thermal overload protection and the stator and bearing temperatures are monitored on the plant computer. Cavitation can be detected by the vibration.

   Pump cavitation/NPSH protection is provided by automatic pump trip if any of the three suction valves leaves the fully open position. Automatic valve closure and pump trip occurs at low reactor water level (level 3), which protects the pump from inadequate NPSH.

e. On-line testing capability of isolation valve operability and interlock circuits, as discussed in paragraph C.5 of the regulatory guide, is not provided. However, the system is periodically tested as discussed in Chapter 16.

5.4.8 REACTOR WATER CLEANUP SYSTEM

The RWCU system is classified as a primary power generation system (not an engineered safeguard feature), a small part of which is part of the RCPB up to and including the second isolation valve. The other portions of the system are not part of the RCPB and are isolatable from the reactor. The RWCU system may be operated at any time during reactor operations, or it may be shut down if water quality is within the Technical Specification limits.

5.4.8.1 Design Basis

5.4.8.1.1 Safety Design Basis

The RCPB portion of the RWCU system meets the requirements of Regulatory Guide 1.26 and Regulatory Guide 1.29 (Section 3.2) to:

a. Prevent excessive loss of reactor coolant

b. Prevent the release of radioactive material from the reactor

c. Isolate the major portion of the RWCU system from the RCPB

5.4.8.1.2 Power Generation Design Basis

The RWCU system performs the following functions:

a. Removes solid and dissolved impurities from reactor coolant, maintains reactor water purity, and measures the reactor water conductivity in accordance with Regulatory Guide 1.56.

b. Discharges excess reactor water during startup, shutdown, and hot standby conditions to the main condenser, condensate storage tank (CST), or radwaste system.

c. Minimizes temperature gradients in the main recirculation piping and RPV during periods when the main recirculation pumps are unavailable.

d. Minimizes the RWCU system heat loss.

e. Enables the major portion of the RWCU system to be serviced during reactor operation.

f. Prevents the standby liquid reactivity control material from being removed by the RWCU system when required for shutdown.

g. Provide a means for monitoring the effectiveness of the Noble Metals Chemical Addition. An ECP monitor measures the electrochemical corrosion potential of the reactor water with respect to the Noble Metals treated piping. A Durability Monitor provides a source of tubing samples which can be analyzed to determine the surface density and wear rate of the deposition of the Noble Metals. A Data Acquisition System collects and stores the flow, temperature and ECP data from the Noble Metals Monitoring System.

5.4.8.2 System Description

The system takes its suction from the inlet of the "B" reactor main recirculation pump and from the RPV bottom head. The process fluid is circulated with the cleanup pump(s) through a regenerative and nonregenerative heat exchanger for cooling, through filter/ demineralizers for cleanup, and back through the regenerative heat exchanger for reheating. The processed water is returned to the RPV and/or the main condenser, CST, or radwaste (drawings M-44 and G31-1030-G-001).

The major equipment of the RWCU system is located outside the drywell. This equipment includes pumps, regenerative and nonregenerative heat exchangers, and filter/demineralizers with precoat equipment. Flow rate capacities for the major pieces of equipment are presented in Table 5.4-2.

The temperature of the filter/demineralizer units is limited by the resin operating temperature. Therefore the reactor coolant is cooled before being processed in the filter/demineralizer units. The regenerative heat exchanger transfers heat from the tube side (hot process inlet) to the shell side (cold process inlet). The shell side flow returns to the reactor. The nonregenerative heat exchanger cools the process further by transferring heat to the reactor enclosure closed cooling water system.

The filter/demineralizer units (drawing M-45) are pressure precoat-type filters using filter aid and powdered, mixed ion exchange resins. Spent resins are nonregenerative 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 limit resins from entering the reactor recirculation system if there is failure of a filter/demineralizer resin support, a strainer is installed on the filter/demineralizer unit. Each strainer and filter/demineralizer vessel has a control room alarm that is energized by high differential pressure. On further increase in differential pressure from the alarm point, the filter/demineralizer automatically isolates.

The backwash and precoat cycle for a filter/demineralizer unit is entirely automatic to prevent human operational errors, such as inadvertent opening of valves, that would initiate a backwash or contaminate the reactor water with resins. The filter/demineralizer piping configuration is arranged to ensure that transfers are complete and crud traps are eliminated. A bypass line is provided around the filter/demineralizer units.

If there is low flow or loss of flow in the system, 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 to ensure that the reactor coolant quality is within limits, and as a check of filter/ demineralizer effectiveness. High conductivity is annunciated in the control room. The control room alarm setpoints for conductivity at the inlet and outlet of the filter/demineralizers are 1.0 μmho/cm and 0.1 μmho/cm, respectively. The effluent setpoint indicates impending resin exhaustion and, therefore, forestalls breakthrough of solubles, usually chlorides. The influent sample point is also used as the normal source of reactor coolant grab samples. Reactor water chloride content and pH will be determined from grab samples taken and analyzed in accordance with approved plant procedures. Additional water quality limits and corrective actions to be taken will be specified in the Technical Specifications.

A Noble Metals Monitoring System sample bypasses the RWCU system from the Regenerative Heat Exchanger Tube Side Inlet Piping to the Shell Side Outlet. An ECP Monitor, measures the electrochemical corrosion potential of the reactor water with respect to the piping. A Durability Monitor provides a source of tubing samples which can be analyzed to determine the surface density and wear rate of the deposition of the Noble Metals. A Data Acquisition System collects and stores the flow, temperature and ECP data from the Noble Metals Monitoring System.

The suction line (RCPB portion) of the RWCU system contains two motor-operated isolation valves, which automatically close in response to signals from reactor low water level, leak detection system, actuation of the SLCS, and nonregenerative heat exchanger high outlet temperature. Section 7.6 describes the leak detection requirements, and they are summarized in Table 5.2-7. This isolation prevents the loss of reactor coolant and release of radioactive material from the reactor, prevents removal of liquid reactivity control material by the cleanup system if the SLCS is in operation, and prevents damage of the filter/demineralizer resins due to high temperature. The RCPB isolation valves may be remotely manually operated to isolate the system equipment for maintenance or servicing. The requirements for the RCPB are specified in Section 5.2.

A remote manually operated spring assisted check valve and a simple check valve in the RWCU return line to feedwater provides instantaneous reverse flow isolation. The spring assisted check valve is a containment isolation valve and is remote manually closed for long term leakage control as described in Section 6.2.4.

A motor operated globe valve in the return line to the reactor is provided for system flow control (throttling) and system isolation for maintenance.

The operation of the RWCU system is controlled from the control room. Resin changing operations, which include backwashing and precoating, are controlled from a local control panel. The time required to remove a unit from the line, backwash, and precoat is approximately 1 hour.

A FCD is provided in Section 7.7.

5.4.8.3 System Evaluation

The RWCU system in conjunction with the condensate cleanup system, and the FPCC system, maintains reactor water quality during all reactor operating modes (normal, hot standby, startup, shutdown, and refueling).

This type of "pressure precoat" cleanup system was first put into operation in 1971 and is used in all operating BWR plants that have been placed in operation since. Operating plant experience has shown that the RWCU system as designed in accordance with these criteria provides the required BWR water quality. The nonregenerative heat exchanger is sized to maintain the required process temperature for filter demineralization even when the cooling capacity of the regenerative heat exchanger is reduced due to bypassing a portion of the return flow to the main condenser, CST, or radwaste. The control requirements of the RCPB isolation valves are designed to the requirements of Section 7.3.1. The component design data (flow rates, pressure, and temperature) are presented in Table 5.4 2. All components are designed to the requirements listed in Section 3.2, according to the requirements of the P&IDs (drawings M-44 and M-45).

5.4.9 MAIN STEAM LINE AND FEEDWATER PIPING

The main steam piping is described in Section 10.3. The feedwater piping is described in Section 10.4.7. Additional design information concerning these lines is found in Sections 3.6, 3.9, and 5.2.

5.4.10 PRESSURIZER

Not applicable to BWRS

5.4.11 PRESSURIZER RELIEF DISCHARGE SYSTEM

Not applicable to BWRS

5.4.12 VALVES

5.4.12.1 Safety Design Bases

Line valves such as gate, globe, and check valves are located in the fluid systems to perform a mechanical function. Valves are components of the system pressure boundary and, having moving parts, are designed to operate efficiently to maintain the integrity of this boundary.

The valves operate under the internal pressure/temperature loading as well as the external loading experienced during the various system transient operating conditions. The design criteria, the design loading, and acceptability criteria are as discussed in Section 3.9.3 for ASME Class 1, 2, and 3 valves. Compliance with ASME Codes is discussed in Section 5.2.1.

5.4.12.2 Description

Line valves furnished are manufactured standard types, designed and constructed in accordance with the requirements of ASME Section III for Class 1, 2, and 3 valves. All materials exclusive of seals, packing, and wearing components are designed to endure the 40 year plant life under the environmental conditions applicable to the particular system when appropriate maintenance is periodically performed.

Power operators have been sized to operate successfully under the maximum differential pressure determined in the design specification.

5.4.12.3 Safety Evaluation

Line valves are shop tested by the manufacturer for performability. Pressure-retaining parts are subject to the testing and examination requirements of ASME Section III. To minimize internal and external leakage past seating surfaces, maximum allowable leakage rates are stated in the design specifications for the back-seat as well as the main seat for gate and globe valves.

Valve construction materials are compatible with the maximum anticipated radiation dose for the service life of the valves.

5.4.12.4 Inspection and Testing

Valves that serve as containment isolation valves and must remain closed or open during normal plant operation may be partially exercised during this period to ensure their operability at the time of an emergency or faulted conditions. Other valves, serving as a system block for throttling valves, may be exercised when appropriate.

Leakage from critical valve stems is monitored by the use of double-packed stuffing boxes with an intermediate lantern leak-off connection for detection and measurement of leakage rates.

Motors used with valve actuators are furnished in accordance with applicable industry standards. Each motor actuator is assembled, factory tested, and adjusted on the valve for proper operation, position and torque switch setting, position transmitter function (where applicable), and speed requirements. Valves are additionally tested to demonstrate adequate stem thrust (or torque) capability to open (or close) the valve within the specified time at specified differential pressure.

Tests verify that there is no mechanical damage to valve components during full stroking of the valve. Suppliers are required to furnish assurance of acceptability of the equipment for the intended service based on any combination of the following:

a. Test stand data

b. Prior field performance

c. Prototype testing

d. Engineering analysis

Preoperational and operational testing performed on the installed valves consists of total circuit checkout and performance tests to verify speed requirements for specified differential pressure.

5.4.13 SAFETY AND RELIEF VALVES

5.4.13.1 Safety Design Bases

Overpressure protection is provided at isolatable portions of systems in accordance with the rules set forth in ASME Section III, for Class 1, 2, and 3 components.

5.4.13.2 Description

Pressure relief valves are designed and constructed in accordance with the same code class as that of the line valves in the system.

Table 3.2-1 lists the applicable code classes for valves, and system design pressures and temperatures are given in Section 5.2.2.4.2.1. The design criteria, design loadings, and design procedures are described in Section 3.9.3.

5.4.13.3 Safety Evaluation

The use of pressure-relieving devices ensures that overpressure does not exceed 10% above the design pressure of the system. The number of relieving devices on a system or portion of a system is determined on this basis.

5.4.13.4 Inspection and Testing

No provisions are to be made for in-line testing of pressure relief valves. Certified set pressures and relieving capacities are stamped on the body of the valves by the manufacturer, and further examinations would necessitate removal of the component.

5.4.14 COMPONENT SUPPORTS

Support elements are provided for the components included in the RCPB and the connected systems.

5.4.14.1 Design Bases

Design loading combinations, design procedures, and acceptability criteria are described in Section 3.9.3. Stress analysis calculations for ASME Section III, Class 1, 2, and 3, and ANSI B31.1 piping conform to the requirements of the appropriate code.

The spacing and size of pipe support elements are based on piping stress analyses performed in accordance with the appropriate codes and as further described in Section 3.7.

Materials, fabrication, and inspection of pipe supporting elements for nuclear piping are in accordance with the USAS B31.7 "Nuclear Power Piping" code, the 1969 issue through the 1971 addenda. Pipe supporting elements for conventional steam and service piping are in accordance with the Code for Power Piping, ANSI B31.1 through the Winter 1974 addenda.

5.4.14.2 Description

The use of rigid-type supports, variable or constant spring-type supports, snubbers, and anchors is determined by stress analysis performed on the piping system. Component support elements are generally manufacturers' standard items.

5.4.14.3 Safety Evaluation

Design loadings used for the determination of component support systems included transient loading conditions expected by each component. Provisions are made to prevent damage to the piping system and the spring-type supports during initial deadweight loading due to hydrostatic testing.

5.4.14.4 Inspection and Testing

After completion of the installation of a support system, component support elements are visually examined to ensure that they are in correct adjustment to their cold setting positions. Verification of satisfactory component support performance is made in accordance with the requirements of Chapter 14.

5.4.15 REFERENCES

5.4-1 P.W. Ianni, "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, GE, Atomic Power Equipment Department (March, 1969).

5.4-3 “Evaluation of Limiting Transients with MSIV/TCV/TSV Closed for Peach Bottom and Limerick at Power,” General Electric Nuclear Energy Letter A096-0013 (J.L. Casillas to G.C. Storey), Sept. 23, 1996.

5.4-4 “GE14 Fuel Design Cycle-Independent Analyses for Limerick Generating Station, Units 1 and 2,” General Electric Nuclear Energy Report, GE-NE-L12-00884-00-01P, March 2001 (G-080-VC-00146).

Table 5.4-1

REACTOR RECIRCULATION SYSTEM DESIGN CHARACTERISTICS

EXTERNAL LOOPS

Number of Loops 2

SINGLE-LOOP PIPING DESCRIPTION	QUANTITY	APPROX LENGTH (feet)	NOMINAL SIZE (inches)
Pump suction line			
Straight pipe	-	0	28
Elbows	3	-	28
Gate valves	1	-	28
Discharge line			
Straight pipe	-	22	28
Elbows	1	-	28
Gate valves	1	-	28
Discharge manifold			
Pipe	-	38	22
Reducer cross	1	-	28x22
Contour nozzle	4	-	22x12
Caps	2	-	22
Concentric reducer	1	-	28x12
External risers			
Straight pipe	5	12/riser	12
Elbows	5	-	12

Design Pressure (psig)/Design Temperature (^{o}F)

Suction piping and valve up to and including pump suction nozzle	1250/575
Pump, discharge valves, and piping Between	1500/575
Piping after discharge blocking valve up to vessel	1500/575
Pump auxiliary piping and cooling water piping	150/212
Vessel bottom drain	1275/575

Table 5.4-1 (Cont'd)

Operation at Rated Conditions

Recirculation Pump

Flow, gpm	45,200
Flow, lb/hr	$17.1x10^6$
Total developed head, ft	710
Suction pressure (static), psia	1070
Required NPSH, ft	138
Water temperature (max), ^oF	537
Pump brake hp (min)	7050
Flow velocity at pump suction, fps	28

Jet Pumps

Number	20
Total jet pump flow, 10^6 lbs/hr	102.6
Throat I.D., in	8.18
Diffuser I.D., in	19.0
Nozzle I.D. (representative), in	3.14
Diffuser exit velocity, fps	15.4
Jet pump head, ft	81.5

Recirculation Block Valve, Discharge

Type	Gate
Actuator	Motor-operated
Material	Austenitic stainless steel
Valve size diameter, in	28

Recirculation Block Valve, Suction

Type	Gate
Actuator	Motor-operated
Material	Austenitic stainless steel
Valve size diameter, in	28

Pump Motor

Voltage rating	3920
Speed, rpm	1668
Meter rating, hp	7500
Phase	3
Frequency, Hz	56
Rotational inertial ($lb\text{-}ft^2$)	14,710

Table 5.4-1 (Cont'd)

Drive Motor and Power Supply	
Frequency Hz (at rated)	55.6
Frequency Hz (operating range)	15.7 - 55.6
Total Required Power to ASD	
kW/ASD Unit	6146
kW Total	12,293

Table 5.4-2

REACTOR WATER CLEANUP SYSTEM EQUIPMENT DESIGN DATA

System flow rate (lbs/hr)		
Normal operation "A" pump		133,000
Normal operation "B" plus "C" pump		133,000
Maximum operation (cold operation only)		180,000
MAIN CLEANUP RECIRCULATION PUMPS	"A" Pump	"B" & "C" Pumps
Number required	1	2
Capacity, % (each)	100	50
Design temperature, ^oF	582	575
Design pressure, psig	1400	1400
Discharge head at shutoff, ft	650	650
Required NPSH, ft	15.5	6.5
HEAT EXCHANGERS	Regenerative	Nonregenerative
Rated capacity, %	100	100
Shell side pressure, psig	1425	150
Shell side temperature, ^oF	575	370
Tube side pressure, psig	1425	1425
Tube side temperature, ^oF	575	575
FILTER/DEMINERALIZERS		
Number required	2	
Capacity, % each	50	
Design temperature, ^oF	150	
Design pressure, psig	1425	

Table 5.4-3
RHR SYSTEM RELIEF VALVE DATA

VALVE LOCATION	VALVE NO.	SETPOINT (psig)	CAPACITY (gpm)(1	METHOD OF COLLECTION(3)
Shutdown supply line (outside containment)	PSV-1F029	140	10	RW
Pump suction line	PSV-1F030 A,B,C&D	170	10	Suppression pool
Pump discharge line	PSV-1F025 A,B,C,D	420	10	RW
Heat exchanger (shell side)(2)	PSV-106 A,B	450	Thermal relief only	Suppression pool
Heat exchanger (tube side)(2)	PSV-105 A,B	450	Thermal relief only	RW
Thermal relief valve on shutdown cooling suction line (inside containment)	PSV-155	1200	2	RW
Thermal relief valve on head spray line, Unit 1 only (ABANDONED)	PSV-122	1200	2	RW

(1) Capacity is based on setpoint plus 10% accumulation
(2) GE supplied valves
(3) RW = liquid radwaste collection system