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5.0 REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

5.1 SUMMARY DESCRIPTION

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

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

   3. The reactor coolant system safety/relief valve piping.

This chapter, specifically <Section 5.4>, discusses various subsystems to the RCPB which are closely allied to it.

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

<Section 5.2.5> establishes the limits on nuclear system leakage inside the drywell so that appropriate action can be taken before the integrity of the nuclear system process barrier is impaired.

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

The reactor recirculation system provides coolant flow through the core. Adjustment of the core coolant flow rate changes reactor power output, thus providing a means of following plant load demand without adjusting control rods. The recirculation system is designed to provide a slow coastdown of flow so that fuel thermal limits cannot be exceeded as a result of recirculation system malfunctions. The arrangement of the recirculation system routing is such that a piping failure cannot compromise the integrity of the floodable inner volume of the reactor vessel.

The main steam line flow restrictors of the venturi-type are 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 reactor vessel water level remains above the top of the core during the time required for the main steam line isolation valves to close. This action protects the fuel barrier.

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

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

The residual heat removal (RHR) system includes a number of pumps and heat exchangers that can be used to cool the nuclear system under a variety of situations. During normal shutdown and reactor servicing, the RHR system removes residual and decay heat. The RHR system allows decay heat to be removed whenever the main heat sink (main condenser) is not available via the suppression pool cooling mode (e.g., hot standby). One mode of RHR operation allows the removal of heat from the primary containment following a postulated loss-of-coolant accident. Another operational mode of the RHR system is low pressure coolant injection (LPCI). LPCI operation is an engineered safety feature for use during a postulated loss-of-coolant accident. This operation is described in <Section 6.3>.

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

Design and performance characteristics of the reactor coolant system and its various components are found in <Table 5.4-1>.

5.1.1 SCHEMATIC FLOW DIAGRAM

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

5.1.2 PIPING AND INSTRUMENTATION DIAGRAM

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

a. The nuclear boiler system shown on <Figure 5.1-3> (partial diagrams of the nuclear boiler system are also shown on <Figure 5.2-11>).

b. The main steam system shown on <Figure 10.1-1>.

c. The feedwater system shown on <Figure 10.1-3>.

d. The recirculation system shown on <Figure 5.4-2>.

e. The reactor core isolation cooling system shown on <Figure 5.4-9>.

f. The residual heat removal system shown on <Figure 5.4-13>.

g. The reactor water cleanup system shown on <Figure 5.4-16>.

5.1.3 ELEVATION DRAWINGS

Elevation drawings showing the principal dimensions of the reactor coolant system in relation to the containment are presented in <Figure 5.1-4>.

5.2 INTEGRITY OF REACTOR COOLANT PRESSURE BOUNDARY

Contained herein are discussions of the measures employed to provide and maintain the integrity of the reactor coolant pressure boundary (RCPB) for the plant design lifetime.

5.2.1 COMPLIANCE WITH CODES AND CODE CASES

5.2.1.1 Compliance with 10 CFR 50.55a

<Table 3.2-1> shows compliance with the rules of <10 CFR 50>. Code edition, applicable addenda and component data are in accordance with <10 CFR 50.55a>.

5.2.1.2 Applicable Code Cases

The reactor pressure vessel and appurtenances, and the RCPB piping, pumps and valves, have been designed, fabricated and tested in accordance with the applicable edition of the ASME Code, including addenda that were mandatory at the order date for the applicable components <Table 3.2-7>. The reactor recirculation loop piping, residual heat removal (shutdown cooling line attached to reactor recirculation Loop B only), main feedwater inside drywell, reactor water clean up piping attached to the reactor recirculation piping and the 10” reactor core isolation and isolation cooling line inside drywell have been reanalyzed for the purposes of optimizing the suspension system. This reanalysis was performed using the 1983 edition with addenda through winter of 1984 of the ASME code. For purposes of standardization, all of the above components supplied by the General Electric Co. will be installed to the Winter 1975 Addenda of the 1974 Edition of the ASME Code, except the Summer 1976 Addenda of the 1974 Edition of the ASME Code is used for slotting of holes on the main steam piping guides (B21-G009). <10 CFR 50.55a> requires code case approval for Class 1 components. These code cases contain requirements or special rules which may be used for the construction of pressure-retaining components of Quality Group Classification A. The various ASME code cases that were applied to components in the RCPB are listed in <Table 5.2-1>.

During RF07, a weld overlay was applied to the feedwater nozzle to safe-end weld 1B13-N4C-KB. The overlay is designed as a full structural overlay in accordance with the recommendations of <NUREG-0313>, Revision 2 (forwarded by <Generic Letter 88-01>), ASME Code Case N-504, and Section XI of the ASME Boiler and Pressure Vessel Code, 1989 Edition (Paragraph IWB-3640). Examination of the final weld overlay was performed in accordance with ASME Code Case N-504 (modified for welding of P-1 and P-43 materials) and <NUREG-0313>, Revision 2 (modified as necessary for examination of Ni-Cr-Fe overlays). Pressure testing of the weld overlay repair was performed in accordance with ASME Section XI, 1989 Edition, no Addenda per ASME Code Case N-416-1 (Reference PNPP letter PY-CEI/NRR-1851L and NRC Safety Evaluation Response).

<Regulatory Guide 1.84> and <Regulatory Guide 1.85> provides a list of ASME Design and Fabrication Cases that have been generically approved by the Regulatory Staff. Code cases on this list may, for design purposes, be used until appropriately annulled. Annulled cases are considered “active” for equipment that has been contractually committed to fabrication prior to the annulment.

In reference to Code Case N-242-1 “Materials Certification,” this code case has been approved for use per <Regulatory Guide 1.85>, Revision 25 (May 1988), provided that all components and supports requiring the use of Paragraphs 1.0 through 4.0 of the Code Case be identified. <Table 5.2-1a> contains a listing of all applicable components and supports affected.

GE’s procedure for meeting the regulatory requirements is to obtain NRC approval for code cases applicable to Class 1 components only. NRC approval of ASME Class 2 and 3 code cases was not required at the time of the design of PNPP.

All Class 2 and 3 equipment has been designed to ASME code or ASME approved code cases. This provision together with the Quality Assurance programs provide adequate safety equipment functional assurances.

ASME Code Case N-411-1, referenced in <Regulatory Guide 1.84>, Revision 25 through 27 require five criteria that must be met prior to and after the use of the Code Case. This section delineates the criteria required, and mandates that these criteria must be met.

a. The Code Case damping shall be used completely and consistently, if used at all. For equipment other than piping, the damping values specified in <Regulatory Guide 1.61> “Damping Values for Seismic Design of Nuclear Power Plants” shall be used.

b. The damping values specified may be used only in those analyses in which current seismic spectra and procedures have been employed. Such use should be limited only to response spectral analyses (similar to that used in the study supporting its acceptance - Reference <NUREG/CR-3526>). The acceptance of the use with other types of dynamic analyses (e.g., time-history analysis or independent support motion method) is pending further justification.

c. When used for reconciliation work or for support optimization of existing designs, the effects of increased motion on existing clearances and on line mounted equipment shall be checked.

d. This Code Case is not appropriate for analyzing the dynamic response of piping systems using supports designed to dissipate energy by yielding (i.e., the design of which is covered by Code Case N420).

e. This Code Case is not applicable to piping in which stress corrosion cracking has occurred unless a case-specific evaluation is made and is reviewed by the NRC staff.

5.2.2 OVERPRESSURIZATION PROTECTION

This section provides evaluation of the systems that protect the RCPB from overpressurization.

The analysis for the initial cycle, documented in this section, was performed at a core power of 3,729 MWt. This analysis resulted in a peak pressure at the bottom of the vessel of 1,276 psig. An updated analysis, discussed in <Section 15.2.4>, was performed for the uprated power case. This analysis resulted in a peak pressure at the bottom of the vessel of 1,295 psig. In both cases, the peak pressure is below the 1,375 psig ASME limit.

The overpressurization protection analysis for the current cycle reload core is discussed in <Appendix 15B>, Reload Safety Analysis.

5.2.2.1 Design Bases

Overpressure protection is provided in conformance with General Design Criteria 15 <Section 3.1>. Preoperational and startup instructions are discussed in <Chapter 14>.

5.2.2.1.1 Safety Design Bases

The nuclear pressure relief system has been designed to:

a. Prevent overpressurization of the nuclear system that could lead to the failure of the reactor coolant pressure boundary.

b. Provide automatic depressurization if small breaks in the nuclear system should occur with subsequent failure/improper operation of the high pressure core spray (HPCS) system, requiring operation of the low pressure coolant injection (LPCI) mode of residual heat removal (RHR) and the low pressure core spray (LPCS) systems 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 safety/relief valves have been designed to meet the following operating bases:

a. Discharge to the containment suppression pool.

b. Correctly reclose following operation so that maximum operational continuity is maintained.

5.2.2.1.3 Discussion

The ASME Boiler and Pressure Vessel Code requires that each vessel designed to meet Section III be protected from overpressure due to upset conditions. The code allows a peak allowable pressure of 110 percent of vessel design pressure under upset conditions. The code specifications for safety/relief valves require that the lowest set pressure is at or below vessel design pressure and that the highest set pressure is such that total accumulated pressure does not exceed 110 percent of the design pressure for upset conditions. The safety/relief valves are designed to open by either of two modes of operation: automatically using a pneumatic power actuator or by self-actuation in the spring lift mode.

The safety/relief valve setpoints are listed in <Table 5.2-2>. These setpoints satisfy the ASME Code, Section III, specifications for safety/relief valves.

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

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

a. Must meet requirements of ASME Code, Section III.

b. Must qualify for 100 percent of nameplate capacity credit for the overpressure protection function.

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

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

5.2.2.1.4 Safety/Relief Valve Capacity

The safety/relief valve capacity of this plant is adequate to limit the primary system pressure, including transients, to the requirements of the ASME Boiler and Pressure Vessel Code, Section III, Nuclear Vessels up to and including the Winter Addenda, 1972. The essential ASME requirements are all met by this analysis.

It is recognized that the protection of vessels in a nuclear power plant is dependent upon many protective systems to relieve or terminate pressure transients. Installation of pressure relieving devices will not independently provide complete protection. The safety valve sizing evaluation assumes credit for operation of the scram protective system which may be tripped by either one of two sources (a direct or flux trip signal). The direct scram trip signal is derived from position switches mounted on the main steamline isolation valves or the turbine stop valves or from pressure switches mounted on the dump valve of the turbine control valve hydraulic actuation system. The position switches are actuated when the respective valves are closing and following 10 percent travel of full stroke. The pressure switches are actuated when a fast closure of the turbine control valves is initiated. Credit is taken for 50 percent of the total installed safety/relief valve capacity operating by the power operated mode as permitted by ASME Code, Section III.

Credit is also taken for the remaining safety/relief valve capacity which opens by the spring mode of operation direct from inlet pressure. The valve flow capacity and discharge coefficient were established through full scale and full flow tests.

The rated capacity of the safety/relief valves is sufficient to prevent a rise in pressure within the pressure vessel not exceeding 110 percent of the design pressure (1.10 x 1,250 psig = 1,375 psig) for the events defined in <Section 15.2> - Increase in Reactor Pressure.

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

<Table 5.2-3> lists the systems which could initiate during the design basis overpressure event.

5.2.2.2 Design Evaluation

5.2.2.2.1 Method of Analysis

To design the pressure protection for the nuclear boiler system, extensive analytical models representing all essential dynamic characteristics of the system are simulated on a large computing facility. These models include the hydrodynamics of the flow loop, the reactor kinetics, the thermal characteristics of the fuel and its transfer of heat to the coolant, and all the principal controller features, such as feedwater flow, recirculation flow, reactor water level, 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.

A detailed description of this model is documented in licensing topical report NEDO-24154, "Qualification of the One Dimensional Core Transient Model for BWR" (Reference 1). Safety/relief valves are simulated in a nonlinear representation, and the model thereby allows full investigation of the various valve response times, valve capacities and actuation setpoints that are available in applicable hardware systems.

Typical valve characteristics as modeled are shown in <Figure 5.2-1> and <Figure 5.2-2> for the pneumatically activated relief and spring action safety modes of the dual purpose safety/relief valves. The associated bypass, turbine control valve, main steam isolation valve and reactor recirculation pump trip due to high reactor pressure characteristics are also simulated in the model.

5.2.2.2.2 System Design

A parametric study was conducted to determine the required steam flow capacity of the safety/relief valves based on the assumptions that follow.

5.2.2.2.2.1 Operating Conditions

Operating conditions for the initial cycle performance were as follows:

a. Operating power is 3,729 MW_t (104.2 percent of nuclear boiler rated power).

b. Vessel dome pressure ≤1,045 psig.

c. Steam flow is 16.16 x 10^6 lb/hr (105 percent of nuclear boiler rated steam flow).

d. Nuclear characteristics: End-of-Cycle.

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

The operating conditions for reload cycle performance of the overpressurization analysis are specified in <Appendix 15B>, Reload Safety Analysis.

5.2.2.2.2.2 Transients

The overpressure protection system must accommodate the most severe pressurization transient. There are two major transients, the closure of all main steam line isolation valves and 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 indirect derived scrams; therefore, it is used as the overpressure protection basis event and the results for the initial cycle are shown in <Figure 5.2-3>. <Table 5.2-4> lists the sequence of events for the main steam line isolation valve closure event with flux scram (performed for the initial cycle) with the installed safety/relief valve capacity.

The transient response and sequence of events for the current reload cycle are provided within <Appendix 15B>, Reload Safety Analysis.

5.2.2.2.2.3 Scram

The scram reactivity curve and control rod drive scram motion are illustrated by <Figure 5.2-4> and <Figure 5.2-5>, respectively. The initial cycle analysis used the second safety grade scram signal with initial reactor pressure at 1,045 psig. The ATWS recirculation pump trip on high reactor pressure was also included.

5.2.2.2.2.4 Safety/Relief Valve Transient Analysis Specification

These assumptions are:

a. Simulated valve groups

Pneumatically actuated relief mode - 4 groups
Spring action safety mode - 5 groups

b. Opening pressure setpoint (maximum safety limit)

Power actuated relief mode -	Group 1:	1,145 psig
	Group 2:	1,155 psig
	Group 3:	1,165 psig
	Group 4:	1,175 psig

Spring action safety mode -	Group 1:	1,175 psig
	Group 2:	1,185 psig
	Group 3:	1,195 psig
	Group 4:	1,205 psig
	Group 5:	1,215 psig

The valve groups used in the reload analyses are based upon the three groups specified in the Technical Specifications for each mode.

The above analyses input setpoints are assumed at a conservatively higher level above the nominal setpoints. 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 2 to 4 percent above the nominal setpoints. Highly conservative safety/relief valve response characteristics are also assumed. Therefore, the analysis conservatively bounds all safety/relief operating conditions.

5.2.2.2.2.5 Safety/Relief Valve Capacity

Sizing of the safety/relief valve capacity is based on establishing an adequate margin from the peak vessel pressure to the vessel code limit (1,375 psig) in response to the reference transients.

Whenever system pressure increases to the relief pressure setpoint of a group of valves having the same setpoint, half of those valves are assumed to operate in the relief mode, opened by the pneumatic power actuation. When the system pressure increases to the valve spring set pressure of a group of valves, those valves not already considered open are assumed to begin opening and to reach full open at 103 percent of the valve spring set pressure. By this method, the total valve capacity can be determined.

5.2.2.2.3 Evaluation of Results

5.2.2.2.3.1 Safety/Relief Valve Capacity

The required safety/relief valve capacity is determined by analyzing the pressure rise from a MSIV closure with flux scram transient. The plant is assumed to be operating at the turbine generator design conditions at a maximum vessel dome pressure of 1,045 psig which is the maximum steady-state operating pressure allowed by the Technical Specification. The analysis hypothetically assumes the failure of the direct isolation valve position scram. The reactor is shut down by the backup, indirect, high neutron flux scram. For the initial cycle analysis, the power actuated relief setpoints of the safety/relief valve are assumed to be in the range of 1,145 to 1,175 psig and the spring action setpoints to be in the range of 1,175 to 1,215 psig. The resulting peak pressure at the bottom of the vessel for the initial cycle is 1,276 psig. Therefore, the analysis indicates that the design valve capacity is capable of maintaining adequate margin below the peak ASME code allowable pressure in the nuclear system (1,375 psig). <Figure 5.2-3>

shows curves produced by this analysis for the initial cycle. The sequence of events in <Table 5.2-4> assumed in this initial cycle analysis was investigated to meet code requirements and to evaluate the pressure relief system exclusively. The results of the overpressurization analysis for the current reload cycle are presented in <Appendix 15B>, Reload Safety Analysis. A curve showing vessel pressure versus valve capacity (number of valves) is shown in <Figure 5.2-7>. This curve is based on a sensitivity study for the BWR/6 design with a 231 inch vessel and shows the relationship between valves out-of-service and margin to the peak allowable ASME code pressure.

Under the General Requirements for Protection Against Overpressure as given in Section III of the ASME Boiler and Pressure Vessel Code, credit can be allowed for a scram from the reactor protection system. In addition, credit is also taken for the protective circuits which are indirectly derived when determining the required safety/relief valve capacity. The backup reactor high neutron flux scram is conservatively applied as a design basis in determining the required capacity of the pressure relieving dual purpose safety/relief valves. Application of the direct position scrams in the design basis could be used since they qualify as acceptable pressure protection devices when determining the required safety/relief valve capacity of nuclear vessels under the provisions of the ASME code. The safety/relief valves are operated in a relief mode (pneumatically) at setpoints lower than those specified for the safety function. This ensures sufficient margin between anticipated relief mode closing pressures and valve spring forces for proper seating of the valves.

The time response of the vessel pressure to the MSIV transient with flux scram for the initial cycle is illustrated in <Figure 5.2-8>. This shows that the pressure at the vessel bottom exceeds 1,250 psig for less than five seconds. This is not long enough to transfer any appreciable amount of heat into the vessel metal which was at a temperature well below 550°F at the start of the transient.

The peak pressure results in this overpressure analysis (and the overpressure analysis for the current reload cycle see <Appendix 15B>) bound all moderate frequency transients in <Chapter 15>.

5.2.2.2.3.2 Low-Low Set Relief Function

To assure that no more than one relief valve reopens following a reactor isolation event, two safety/relief valves are provided with lower opening and closing setpoints and four valves are provided with lower closing setpoints. These setpoints override the normal setpoints following the initial opening of the relief valves and act to hold open these valves longer, thus preventing more than a single valve from reopening subsequently. This system logic is referred to as the low-low set relief logic and functions to ensure that the containment design basis of one safety/relief valve operating on subsequent actuations is met.

The low-low set relief function is armed whenever any safety/relief valves are called upon to open in the relief mode by pressure instruments. Thus, the low-low set valves will not actuate during normal plant operation even though the reopening setpoints of one of the valves is in the normal operating pressure range. This arming method results in the low-low set safety/relief valves opening initially during an overpressure transient at the normal relief opening setpoint.

The lowest setpoint low-low set valve will cycle to remove decay heat. Since this valve will have a larger differential between its opening and closing set pressures than assumed for the normal relief function, the number of single safety/relief valve actuations during isolation events will be reduced. <Table 5.2-2> shows the opening and closing setpoints for the low-low set safety/relief valves.

The assumptions used in the calculation of the pressure transient after the initial opening of the relief valves are:

a. The transient event is a three-second closure of all MSIV's with position scram.

b. Nominal relief valve setpoints are used.

c. The maximum expected relief capacity is used.

d. Relief valve opening and closing response times shown in <Figure 5.2-6a> are used.

e. The closing setpoint of the relief valves is 100 psi below the opening setpoint.

f. ANS + 20 percent decay heat at infinite exposure is used.

The results using the above assumptions are shown in the reactor vessel pressure transient curve shown in <Figure 5.2-6b>. Despite the conservative input assumptions which tend to maximize the pressure peaks on subsequent actuations, there is a 65 psi margin for avoiding the second opening of more than one valve. The system is single failure proof since a failure of one of the low-low set valves still gives a 42 psi margin for avoiding multiple valve actuations.

The safety/relief valves are balanced type, spring loaded safety valves provided with an auxiliary pneumatically actuated device which allows opening of the valve even when pressure is less than the safety-set pressure of the valve. Previous undesirable performance on operating BWRs was associated principally with multiple stage pilot operated safety/relief valves. These newer, pneumatically operated safety valves employ significantly fewer moving parts wetted by the steam and are, therefore, considered an improvement of the previously used valves.

5.2.2.2.3.3 Pressure Drop in Inlet and Discharge

Pressure drop in the piping from the reactor vessel to the valves is taken into account in calculating the maximum vessel pressures. Pressure drop in the discharge piping to the suppression pool is limited by proper discharge line sizing to prevent backpressure on each safety/relief valve from exceeding 40 percent of the valve inlet pressure; this assures choked flow in the valve orifice and no reduction of valve capacity due to the discharge piping. Each safety/relief valve has its own separate discharge line.

5.2.2.3 Piping and Instrument Diagrams

<Figure 5.2-9> shows the schematic location of safety/relief valves for:

a. The reactor coolant system.

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 safety/relief valves.

The schematic arrangements of the safety/relief valves are shown in <Figure 5.2-10>.

5.2.2.4 Equipment and Component Description

5.2.2.4.1 Description

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

The safety/relief valves provides the following protection functions:

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

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

c. Depressurization operation. The ADS valves open automatically as part of the emergency core cooling system (ECCS) for events involving small breaks in the nuclear system process barrier. The location and number of the ADS valves can be determined from <Figure 5.2-10>.

<Chapter 15> discusses the events which are expected to activate the primary system safety/relief valves. The section also summarizes the number of valves expected to operate during the initial blowdown of the valves and the expected duration of this first blowdown. For several of the events it is expected that the lowest set safety/relief (pressure or power set) valve will reopen and reclose as generated heat drops into the decay heat characteristics. The pressure increase and relief cycle will continue with lower frequency and shorter relief discharges as the decay heat drops off, and until such time as the RHR system can dissipate this heat. Remote-manual actuation of the valves from the control room is recommended to minimize the total number of these discharges, with the intent of achieving extended valve seat life.

A schematic of the safety/relief valve is shown in <Figure 5.2-12>. It is opened by either of two modes of operation:

a. The spring mode which consists of direct action of the steam pressure against a spring loaded disk that will pop open when the valve inlet pressure force exceeds the spring force.

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

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

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

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

For overpressure relief valve operation (power actuated mode), each valve is provided with a pressure sensing device which operates at the setpoints designated in <Table 5.2-2>. When the set pressure is reached, it operates a solenoid air valve which in turn actuates the pneumatic piston/cylinder and linkage assembly to open the valve.

When the piston is actuated, the delay time, maximum elapsed time between receiving the overpressure signal at the valve actuator and the actual start of valve motion, will not exceed 0.1 seconds. The maximum elapsed time between signal to actuator and full open position of valve will not exceed 0.25 seconds.

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

Actuation of either solenoid A or solenoid B on the safety/relief valve will cause the safety/relief valve to open; hence, there is no single failure of a logic component or safety/relief valve solenoid valve which would result in failure of the safety/relief valve to open. The trip units for each safety/relief valve within each division are in series, and failure of one of the transmitters will not cause the safety/relief valves to open. Each safety/relief valve is provided with its own pneumatic accumulator and inlet check valve. The accumulator capacity is sufficient to provide one safety/relief valve actuation, all that is required for overpressure protection. Subsequent actuations for an overpressure event can be spring actuations to limit reactor pressure to acceptable levels.

The safety/relief valves are designed to operate to the extent required for overpressure protection in the following accident environments:

a. A temperature of 330°F for three hours at a drywell pressure ≤30 psig.

b. A temperature of 310°F for an additional three hour period at a drywell pressure ≤15 psig.

c. A temperature of 250°F for an additional 18 hour period at 15 psig.

d. A temperature drop of 250°F to 100°F at 15 psig from one day to 100 days. The valve must remain operable for the initial two days and be held either open or closed for the remaining 98 of the 100 days.

The automatic depressurization system (ADS) uses selected safety/relief valves for depressurization of the reactor as described in <Section 6.3>. Each of the safety/relief valves used for automatic depressurization is equipped with an air accumulator and check valve arrangement. These accumulators assure that the valves can be held open following failure of the air supply to the accumulators. They are sized to be capable of opening the valves and holding them open against the maximum drywell pressure of 30 psig. The accumulator capacity is sufficient for each ADS valve to provide two actuations against 70 percent of maximum drywell pressure. The ADS accumulators are recharged as described in <Section 6.8.1>.

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

As a part of the startup testing of the main steam lines, movement of the safety/relief valve discharge lines was monitored.

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

The nuclear pressure relief system automatically depressurizes the nuclear system sufficiently to permit the RHR and LPCS systems to operate as a backup for the high pressure core spray (HPCS) system. Further descriptions of the operation of the automatic depressurization feature are found in <Section 6.3> and <Section 7.3.1>.

5.2.2.4.2 Design Parameters

The specified operating transients for components within the RCPB are given in <Section 3.9>. Refer to <Section 3.7> for discussion of the input criteria for design of Seismic Category I structures, systems and components.

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

5.2.2.4.2.1 Safety/Relief Valve

The discharge area of the valve is 18.429 sq inches and the coefficient of discharge K_D is equal to 0.873 ($K = 0.9\ K_D$).

The design pressure and temperature of the valve inlet and outlet are 1,375 psig at 585°F and 625 psig at 500°F, respectively.

The valves have been designed to achieve the maximum practical number of actuations consistent with state-of-the-art technology.

5.2.2.5 Mounting of Pressure Relief Devices

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

a. Thermal expansion effects of the connecting piping.

b. Dynamic effects of the piping due to SSE.

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

d. 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 SRV discharge are contained in <Section 3.9.3>.

5.2.2.6 Applicable Codes and Classification

The vessel overpressure protection system is designed to satisfy the requirements of Section III of the ASME Boiler and Pressure Vessel Code. The general requirements of Section III of the code for protection against overpressure recognize that reactor vessel overpressure protection is one function of the reactor protective systems and allow the integration of pressure relief devices with the protective systems of the nuclear reactor. Hence, credit is taken for the scram protective system as a complementary pressure protection device. The NRC has also adopted the ASME codes as part of their requirements in <10 CFR 50.55a>.

5.2.2.7 Material Specification

Material specifications of pressure retaining components of safety/relief valves are reported in <Table 5.2-5>.

5.2.2.8 Process Instrumentation

Overpressure protection process instrumentation is listed in Table 1 of <Figure 5.1-3 (3)>.

5.2.2.9 System Reliability

The system is designed to satisfy the requirements of Section III of the ASME Boiler and Pressure Vessel Code. Therefore, it has high reliability. The consequences of failure are discussed in <Section 15.1.4> and <Section 15.6.1>.

5.2.2.10 Testing and Inspection

The safety/relief valves are tested at vendor's shop in accordance with quality control procedures to detect defects and to prove operability prior to installation. The following tests are conducted:

a. Hydrostatic test at specified test conditions.

b. Pneumatic seat leakage test at 90 percent of set pressure with maximum permitted leakage of 30 bubbles per minute emitting from a 0.250-in. diameter hole submerged 1/2 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 ±3 percent. As left, tolerance is ±1 percent of set pressure.

d. Response time test: each safety/relief valve 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. Specified manual actuation relief mode of each safety/relief valve is verified during the startup test program.

A minimum of 20 percent of the installed valves shall be removed for testing every refueling outage, with the maximum number of years for the testing of all valves not to exceed that specified in the PNPP ASME Code of record. Removed valves shall be inspected and tested as follows:

a. Set pressure test: Verify set pressure of the removed valves during refueling outages. Verify opening and closing times by using the pneumatic power actuator unless relief has been granted. Verify that valve mainseat leakage is within acceptable limits.

b. Inspection: Inspect all external surfaces and parts; disassemble and inspect internal surfaces and parts for wear/damage/erosion. Replace all damaged or worn parts and gaskets/seals as necessary due to inspections results. Lubricate valves and relap valve seats if inspection or testing necessitates. Retest all valves disassembled and make appropriate adjustments prior to use.

Valve operability is verified during the preoperational test program as discussed in <Chapter 14>. See <Figure 5.2-12> for a schematic cross section of the valve.

5.2.3 REACTOR COOLANT PRESSURE BOUNDARY MATERIALS

5.2.3.1 Material Specifications

<Table 5.2-5> lists the principal pressure retaining materials and the appropriate material specifications for the reactor coolant pressure boundary components.

5.2.3.2 Compatibility with Reactor Coolant

5.2.3.2.1 PWR Reactor Coolant Chemistry

This section is not applicable to PNPP.

5.2.3.2.2 BWR Reactor Coolant Chemistry

Materials in the primary system are primarily austenitic stainless steel and Zircaloy cladding. The reactor water chemistry limits are established to provide an environment favorable to these materials. Limits are placed on conductivity and chloride concentrations. Conductivity is limited because it can be continuously and reliably measured and gives an indication of abnormal conditions and the presence of undesired dissolved ionic species in the coolant. Chloride limits are specified to prevent stress corrosion cracking of stainless steel (Reference 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 such investigation is of the chloride-oxygen relationship by Williams (Reference 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.

The water quality requirements are further supported by General Electric stress corrosion test data summarized as follows:

a. Type 304 stainless steel specimens were exposed in a flowing loop operating at 537°F. The water contained 1.5 ppm chloride and 1.2 ppm oxygen at pH 7. Test specimens were bent beam strips stressed over their yield strength. After 2,100 hours exposure, no cracking or failures occurred.

b. Welded Type-304 stainless steel specimens were exposed in a refreshed autoclave operating at 550°F. The water contained 0.5 ppm chloride and 1.5 ppm oxygen at pH 7. Uniaxial tensile test specimens were stressed at 125 percent of their 550°F yield strength. No cracking or failures occurred at 15,000 hours exposure.

When conductivity is in its normal range, pH, chloride and other impurities affecting conductivity will also be within their normal range. When conductivity becomes abnormal, chloride measurements are made to determine whether or not they are also outside their normal operating values. If the chloride content is within limits, the conductivity could be high due to the presence of a neutral salt which 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 where near neutral pH is maintained, conductivity provides a good and prompt measure of the quality of the reactor water. Significant changes in conductivity provide the operator with a warning mechanism so he can investigate and remedy the condition before reactor water limits are reached. Methods available to the operator for correcting the off-standard condition include operation of the reactor water cleanup system, 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.

5.2.3.2.2.1 Summary of BWR Water Chemistry

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 on <Figure 5.2-13>. Reference to <Table 5.2-6> has been made as numbered on the diagram and correspondingly in the table.

   For normal operation starting with the condenser-hotwell, condensate water is processed through a condensate treatment system. This process consists of full flow filtration and full flow demineralization, resulting in effluent water quality represented in <Table 5.2-6>.

   The effluent from the condensate treatment system is pumped through the feedwater heater train, zinc injection occurs, and the feedwater enters the reactor vessel at an elevated temperature and with a chemical composition as shown in <Table 5.2-6>.

   During normal plant operation, boiling occurs in the reactor, decomposition of water takes place due to radiolysis, and oxygen and hydrogen gas is 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 5.2-6>. At the condenser, deaeration takes place and the gases are removed from the process by means of steam jet air ejectors (SJAEs). The deaeration is completed to a level of approximately 20 ppb (0.02 ppm) of oxygen in the condensate.

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 above described chemistry is modified with the operation of the hydrogen water chemistry system. Hydrogen is injected at the operating feedwater pump during normal plant operation. The hydrogen, in conjunction with Zinc Injection, Noble Metals Chemical Addition (NMCA), and the On-Line NobleChem System (OLNC), combines with oxygen and oxides in the reactor water, lowering the Electrochemical Corrosion Potential (ECP) to below the -230mV Standard Hydrogen Electrode threshold to mitigate the potential and growth of Intergranular Stress Corrosion Cracking (IGSCC) of the stainless steel piping and reactor internal components. Due to recombining of free hydrogen and oxygen, this injection also results in a substantial reduction of oxygen levels, thereby reducing a critical contributor to IGSCC.

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

A zinc injection system is provided which injects zinc into the feedwater for reduction of radiation fields on piping within the primary system. The presence of soluble zinc, in correctly controlled concentration ranges, reduces the amount of Co-60 incorporated into the corrosion film of the piping. Zinc is in the zinc oxide chemical form. The zinc or oxygen contribution does not appreciably alter either the BWR water chemistry or the radiological inventory of the reactor coolant.

Chemical parametric values for the reactor water are listed in <Table 5.2-6> for various plant conditions.

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

No other inputs of water or sources of oxygen are present during normal plant operation. During plant conditions other than normal operation additional inputs and mechanisms are present as outlined in the following section.

b. Plant Conditions Outside Normal Operation: During periods of plant conditions other than normal power production, transients take place, particularly with regard to oxygen levels in the primary coolant. Oxygen levels in the primary coolant will vary from the normal during plant startup, plant shutdown, hot standby, and when the reactor is vented and depressurized. The hotwell condensate will absorb oxygen from the air when vacuum is broken on the condenser. Prior to startup vacuum is established in the condenser and deaeration of the condensate takes place by means of mechanical vacuum pump and steam jet air ejector (SJAE) operation and condensate recirculation. During these plant conditions, continuous input of control rod drive (CRD) cooling water takes place as described previously.

1. Plant Depressurized and Reactor Vented

During certain periods such as refueling and maintenance outages, the reactor is vented to the condenser or atmosphere. Under these circumstances the reactor cools and the oxygen concentration increases to a maximum value of 8 ppm. Equilibrium between the atmosphere above the reactor water surface, the CRD cooling water input, any residual radiolytic effects, and the bulk reactor water will be established after some time. No other changes in water chemistry of significance take place during this plant condition because no appreciable inputs take place.

2. Plant Transient Conditions - Plant Startup/Shutdown

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

(a) Plant Startup

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

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

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

(b) Plant Shutdown

Upon plant shutdown following power operation, the radiolytic oxygen generation essentially ceases as the fission process is terminated. Oxygen is no longer generated but due to residual energy some steaming still takes place; the oxygen concentration in the coolant will decrease to a minimum value determined by the steaming rate temperature. If venting is performed, a gradual increase essentially to oxygen saturation at the coolant temperature will take place, reaching a maximum value of less than 8 ppm oxygen.

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

As can be concluded from the preceding descriptions, the maximum possible oxygen concentration in the reactor coolant and any other directly related or associated parts is that of air saturation at ambient temperature. At no time or location, in the water phase, will oxygen levels exceed the nominal value of 8 ppm. As temperature is increased and hence, oxygen solubility decreased accordingly, the oxygen concentration will be maintained at this maximum value, or reduced below it depending on available removal mechanisms (i.e., diffusion, steam stripping, flow transfer, or degassing).

Depending on the location, configuration, etc., such as dead legs or stagnant water, inventories may contain 8 ppm dissolved oxygen or some other value below this maximum limitation.

Conductivity of the reactor coolant is continuously monitored. Conductivity instruments are connected to redundant sources: the reactor water recirculation loop and the reactor water cleanup system inlet. The effluent from the reactor water cleanup system is also monitored for conductivity on a continuous basis. These measurements provide reasonable assurance for adequate surveillance of the reactor coolant.

Grab samples are provided, for the locations shown on <Table 5.2-7>, for special and non-continuous measurements such as pH, oxygen, chloride and radiochemical measurements.

The relationship of chloride concentration to specific conductance measured at 25°C for chloride compounds such as sodium chloride and hydrochloric acid can be calculated, as shown on <Figure 5.2-14>. Values for these compounds essentially bracket values of other common chloride salts or mixtures at the same chloride concentration. Surveillance requirements are based on these relationships.

In addition to this program, limits, monitoring and sampling requirements are imposed on the condensate, condensate treatment system and feedwater by warranty requirements and specifications. Thus, a total plant water quality surveillance program is established providing assurance that off specification conditions will quickly be detected and corrected.

The sampling frequency when reactor water has a low specific conductance is adequate for 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 Operational Requirements Manual 6.3.1.)

The primary coolant conductivity monitoring instrumentation, ranges, accuracy sensor and indicator locations are shown in <Table 5.2-7>. The sampling is coordinated in a reactor sample station especially designed with constant temperature control and sample conditioning and flow control equipment.

3. Water Purity During a Condenser Leak

The condensate cleanup system is designed to maintain the reactor water chloride concentration below 200 ppb during a condenser tube leak of 50 gallons per minute for one hour.

<Regulatory Guide 1.56> describes an acceptable method of implementing GDC 13, 14, 15, and 31 of <10 CFR 50, Appendix A> with regard to minimizing the probability of corrosion-induced failure of the RCPB in BWRs. This is done by maintaining acceptable purity levels in the reactor coolant, and acceptable instrumentation to determine the condition of the reactor coolant.

At PNPP, to protect against the effects of a major condenser tube leak, the six deep-bed demineralizers can be utilized to provide full flow polishing capability. Each vessel is equipped with two inline conductivity cells. One conductivity cell measures the water quality effluent and the other, located at 93 percent of bed depth, is set to alarm at or below 0.10 μmho/cm, indicating that the resin is near exhaustion and the demineralizer should be removed from service and the resin replaced. The in-bed conductivity cells are placed at a 93% bed depth to ensure adequate reserve ion exchange capacity is always maintained in the event of a condenser tube leak of system design magnitude. This ensures sufficient ion exchange capacity is available to conduct leak isolations or an orderly reactor shutdown, if necessary. The water quality limits for the condensate system meet or exceed the requirements of <Regulatory Guide 1.56>, Revision 1.

Total capacity of anion and cation demineralizer resins will be measured prior to addition to the vessel. Since the demineralizer beds are not equipped with sample taps, and to keep person-rem exposures ALARA, sampling and testing of inservice resins is not performed. Bed capacity assessment and replacement frequency are based on the inline conductivity measurements. Resins are replaced in accordance with the manufacturer’s recommendations.

Based on the above design features, particularly the inline conductivity measurements, the deep-bed demineralizer system provides adequate capacity for ionic impurity removal in the event of condenser leakage.

Corrective action for bed replacement is taken at or below 0.10 μmho/cm. This is 50 percent of the limit in <Regulatory Guide 1.56>, Revision 1.

As previously mentioned, the materials in the primary system are primarily austenitic stainless steel and Zircaloy cladding. The reactor water chemistry limits have been established to provide an environment favorable to these materials. Design Engineering and Operational Requirements Manual limits are placed on conductivity and chloride concentrations. Operationally, the 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 prevent stress corrosion cracking of stainless steel.

5.2.3.2.3 Compatibility of Construction Materials with Reactor Coolant

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

a. Solution annealed austenitic stainless steels (both wrought and cast) Types 304, 304L, 316, 316L, and 347 modified.

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 1,100°F).

e. Colmonoy, Stellite, and NOREM hardfacing material.

f. The compatible weld metals for joining items (a) through (d) above.

All of these materials of construction are resistant to stress corrosion in the BWR coolant except for Inconel 182 weld metal and welds in Type 304 material which were not solution annealed. Inconel 182 weld metal buttering is located at the RPV nozzle to safe-end connection of the reactor recirculation, feedwater, low pressure core spray, high pressure core spray, residual heat removal, and jet pump instrumentation systems. The weld overlay installed on the feedwater nozzle to safe-end weld 1B13-N4C-KB used Alloy 52. This material, if ever exposed to reactor coolant, is compatible and highly resistant to stress corrosion.

Non-solution annealed welds in Type 304 materials are located at the jet pump instrumentation nozzle safe-end to penetration seal connections. These welds have been stress improved with the Mechanical Stress

Improvement Process (MSIP) as discussed in <Section 5.3.3.1.4.5> to mitigate intergranular stress-corrosion cracking (IGSCC). 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 by the reactor water quality specifications. No detrimental effects will occur on any of the materials from allowable contaminant levels in the high purity reactor coolant. Expected radiolytic products in the BWR coolant have no adverse effects on the construction materials.

5.2.3.2.4 Compatibility of Construction Materials with External Insulation and Reactor Coolant

Metallic and nonmetallic insulation materials are discussed in <Section 6.1.1>, <Section 6.2.1>, and <Section 6.2.2>.

5.2.3.3 Fabrication and Processing of Ferritic Materials

5.2.3.3.1 Fracture Toughness

5.2.3.3.1.1 Compliance with Code Requirements

a. The ferritic materials used for piping, pumps and valves of the reactor coolant pressure boundary are 2-1/2 inches or less in thickness. Impact testing is performed in accordance with NB-2322 for thicknesses of 2-1/2 inches or less.

b. Materials for bolting with nominal diameters exceeding one inch are required to meet both the 25 mils lateral expansion specified in NB-2333 and the 45 ft-lb Charpy V value specified in <10 CFR 50, Appendix G>.

c. The reactor vessel complies with the requirements of NB-2331. The reference temperature, RT_{NDT}, will be established for all required pressure retaining materials used in the construction of Class I vessels. This includes plates, forgings, weld material, and heat affected zone. The RT_{NDT} differs from the nil-ductility temperature, NDT, in that in addition to passing the drop weight test, three Charpy-V-Notch specimens (traverse) must exhibit 50 ft-lbs absorbed energy and 35 mil lateral expansion at 60°F above the RT_{NDT}. The core beltline material must meet 75 ft-lbs absorbed upper shelf energy.

5.2.3.3.2 Control of Welding

5.2.3.3.2.1 Control of Preheat Temperature Employed for Welding of Low Alloy Steel

<Regulatory Guide 1.50> delineates preheat temperature control requirements and welding requirements procedure qualifications supplementing those in ASME Code Sections III and IX.

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

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

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

5.2.3.3.2.2 Control of Electroslag Weld Properties

No electroslag welding was performed on BWR components, therefore, <Regulatory Guide 1.34> is not applicable.

5.2.3.3.2.3 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 fabrication requirements of Section III and Section IX of the ASME Boiler and Pressure Vessel Code. It also recommends additional performance qualifications for welding in areas of limited access.

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

5.2.3.3.3 Nondestructive Examination of Ferritic Tubular Products

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

These RCPB components met the requirements of ASME codes existing at time of placement of order which predated <Regulatory Guide 1.66>. At the time of the placement of the orders, <10 CFR 50, Appendix B> requirements and the ASME code requirements ensured adequate control of quality for the products.

<Regulatory Guide 1.66> was withdrawn on September 28, 1977, by the NRC because the additional requirements imposed by the guide were satisfied by the ASME code.

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.

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

The purpose of <Regulatory Guide 1.44> is to address <10 CFR 50, Appendix A>, GDC 1 and 4, and <10 CFR 50, Appendix B> requirements to control "the application and processing of stainless steel to avoid severe sensitization that could lead to stress corrosion cracking." The guide proposes that this should be done by limiting sensitization due to welding as measured by ASTM A262 Practice A and E, or another method that can be demonstrated to show nonsensitization in austenitic stainless steel.

All austenitic stainless steel was purchased in the solution heat treated condition in accordance with applicable ASME and ASTM specifications.

Wrought stainless steel primary pressure boundary piping was fabricated using Type 304 material. To avoid sensitization which could result from welding of the Type 304 stainless steel wherever practical, the piping was solution heat treated after welding. All welded areas which could not be solution heat treated were protected by applying high ferrite (5 percent minimum ferrite) Type 308L weld overlay prior to the welding operation.

These methods of providing protection against stress corrosion cracking comply with the requirements of <NUREG-0313>, "Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping," July 1977. There is no nonconforming stainless steel piping on the primary pressure boundary.

All weld filler metal and castings were required by specification to have a minimum of 5 percent ferrite.

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 comply with the intent of <Regulatory Guide 1.44>.

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 was avoided by carefully controlling all cleaning and processing materials which contact the stainless steel during manufacture and construction.

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

The degree of surface cleanliness obtained by these procedures follows the recommendations of <Regulatory Guide 1.44>.

5.2.3.4.1.3 Cold Worked Austenitic Stainless Steels

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

5.2.3.4.2 Control of Welding

5.2.3.4.2.1 Avoidance of Hot Cracking

<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.

Written welding procedures which are approved by General Electric are required for all primary pressure boundary welds. These procedures comply with the requirements of Sections III and IX of the ASME Boiler and Pressure Vessel Code and applicable NRC regulatory guides.

All austenitic stainless steel weld filler materials were required by specification to have a minimum of 5 percent ferrite. Prediction of ferrite content was made by using the chemical composition in conjunction with the Schaeffler diagram. The use of the 5 percent 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 General Electric Company, with the concurrence of the regulatory staff, demonstrated that controlling weld filler metal ferrite at 5 percent 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 Branch Technical Position MTEB No. 5-1 "Interim Regulatory Position of <Regulatory Guide 1.31>, Control of Stainless Steel Welding."

5.2.3.4.2.2 Electroslag Welds

Electroslag welding was not employed for reactor coolant pressure boundary components, therefore <Regulatory Guide 1.34> is not applicable.

5.2.3.4.2.3 Welder Qualification for Areas of Limited Accessibility

Qualification for areas of limited accessibility, as recommended in <Regulatory Guide 1.71>, is discussed in <Section 5.2.3.3.2.3>.

5.2.3.4.3 Nondestructive Examination of Tubular Products

For discussion of compliance with <Regulatory Guide 1.66> see <Section 5.2.3.3.3>.

5.2.4 INSERVICE EXAMINATION AND PRESSURE TESTING OF REACTOR COOLANT PRESSURE BOUNDARY

All components in Quality Group A will be examined in accordance with Section XI of the ASME Code. The edition and addenda will be in accordance with <10 CFR 50.55a> as indicated in the Inservice Examination Program.

The Inservice Examination Program will cover Class 1 systems and components as described in ASME Code Section XI. Exceptions for those portions of systems that cannot be examined to fully meet the requirements of ASME Code Section XI, if any, will be fully identified and the reasons for the exceptions given in the program. The program will also define a schedule for examinations.

The design and arrangements of Class 1 system components will provide adequate clearances to conduct the required examinations at the code-required inspection interval.

The Inservice Examination Program will describe the scope of the examinations and include isometric drawings and component sketches. The drawings will show weld locations in the various piping systems and on components. Boundary diagrams and classification tables will be incorporated into the program to delineate systems boundaries. The program will specify the type of examinations to be performed and the total extent of the examination coverage for each system and component.

Detailed procedures for volumetric (ultrasonic), surface penetrant and visual examinations are used in support of the program. Accompanying drawings will include diagrams of calibration blocks and unique designations for each block to be used in the examination procedure.

An inspection schedule for Class 1 system components are developed in accordance with the guidance of ASME Code Section XI, Subarticle IWB-2400.

The inservice examination categories and requirements for Class 1 components are in agreement with ASME Code Section XI, Subarticle IWB-2500.

The evaluation of Class 1 component examination results complies with the requirements of Article IWB-3000 of ASME Code Section XI. The repair procedures for Class 1 components will comply with the requirements of Article IWA-4000 of ASME Code Section XI.

The program for Class 1 system pressure testing will comply with the criteria of ASME Code Section XI, Article IWB-5000.

5.2.4.1 System Boundary Subject to Examination

The reactor pressure vessel, system piping, pumps, valves, and components within the reactor coolant pressure boundary defined as quality Group A (ASME Code Section III Class 1) were designed and fabricated to permit full compliance with ASME Code Section XI. All components in Quality Group A were examined in accordance with the Summer 1978 Addenda of Section XI of the ASME Code for preservice examination. The preservice examination was performed in accordance with <10 CFR 50.55a(g)(3)>. Inservice examinations will be performed in accordance with <10 CFR 50.55a(g)(4)>. Access is provided for volumetric examination of pressure retaining welds from the external surface. The examination procedures have been considered in the design of components, weld joint configurations and system arrangements to assure inspectability. Periodic design reviews and onsite audits are made throughout the design and erection phases to insure that these objectives are being met.

The Inservice Examination Program covers Class 1 systems and components as described in ASME Code Section XI. Exceptions for those portions of systems that cannot be examined to fully meet the requirements of ASME Code Section XI, if any, are fully identified and the reasons for the exceptions given in the program. The program also defines a schedule for examinations.

The ASME Code Class 1 components (including supports and pressure retaining bolting) subject to inspection according to the method specified in Table IWB-2500-1 of ASME Code Section XI include the reactor pressure vessel and piping, pumps and valves within the following systems. Where the system penetrates primary containment the areas of examination on Class 1 components as defined in Table IWB-2500 will be extended up to and including the first isolation valve outside containment.

a. Reactor pressure vessel

b. Main steam

c. Reactor feedwater

d. Reactor recirculation

e. Control rod drive

f. Residual heat removal system

g. Core spray

h. Reactor core isolation cooling system

i. Standby liquid control/core ΔP

j. Reactor water cleanup

k. Reactor drain

The Inservice Examination Program describes the scope of the examinations and includes isometric drawings and component sketches. The drawings show weld locations in the various piping systems and on components. Boundary diagrams and classification tables are incorporated into the program to delineate systems boundaries. The program specifies the type of examinations to be performed and the total extent of the examination coverage for each system and component.

5.2.4.2 Provisions for Access to the Reactor Coolant Pressure Boundary

The design and arrangements of Class 1 system components provide adequate clearances to conduct the required examinations at the code-required inspection interval.

5.2.4.2.1 Reactor Pressure Vessel

Access for examination of the reactor pressure vessel has been provided through provisions incorporated into the design of the vessel, shield wall and vessel insulation as follows:

a. The shield wall and vessel insulation behind the shield wall are spaced away from the RPV outside surface. Access ports are located at each reactor pressure vessel nozzle and at the base of the shield wall. The shield wall openings at each nozzle in conjunction with removable insulation panels at each opening provide access for examination of nozzle-to-vessel and nozzle-to-piping welds using either manual or automated ultrasonic examination techniques. The annular space between the reactor vessel outside surface and insulation inside surface permits insertion of remotely operated ultrasonic devices for examination of vessel longitudinal and circumferential welds. Access for insertion of the automated devices is provided through the ports at the base of the shield wall or through removable insulation panels at the top of the shield wall.

b. Access to the reactor pressure vessel circumferential, longitudinal and nozzle-to-vessel welds above the shield wall is provided through use of removable insulation panels. Either manual or automated examination methods may be employed.

c. The vessel flange area and vessel closure head can be examined during normal refueling using manual ultrasonic methods. The examination of the flange-to-vessel weld can be performed manually from the flange seal surface.

d. The closure head is dry stored during refueling. Removable insulation permits manual examination of all welds on the vessel head from the outside surface. The nuts and washers are dry stored during refueling and may be examined at that time. All reactor pressure vessel studs are accessible for required examinations during refueling either in place or when removed.

e. Openings in the RPV support skirt provide access for manual or automated ultrasonic methods for examination of the accessible meridianal and circumferential welds within the support skirt.

f. With the closure head removed, access is provided to the upper interior portion of the vessel by removal of the steam dryer and steam separator assemblies. RPV internals can be visually examined by underwater TV systems. Items to be examined include, but are not limited to, core spray spargers, feed water nozzle internals and the top guide.

5.2.4.2.2 Pipe, Pumps and Valves

5.2.4.2.2.1 Arrangements

Physical arrangement of pipe, pumps and valves provide personnel access to each weld location. Working platforms are provided at areas to facilitate servicing of pumps and valves. Temporary platforms, scaffolding and ladders will be provided to gain access to piping welds including the pipe-to-reactor vessel nozzle welds. Removable thermal insulation is provided on welds and components which require frequent access for examination or are located in high radiation areas.

5.2.4.2.2.2 Accessibility for Ultrasonic Examination

Welds are located to permit ultrasonic examination from at least one side but where component geometries permit, access from both sides is provided. Consideration was given to weld joint configurations and surfaces during fabrication to permit thorough ultrasonic examinations.

5.2.4.3 Examination Techniques and Procedures

Detailed procedures for volumetric (ultrasonic), surface penetrant and visual examinations have been prepared. Accompanying drawings include diagrams of calibration blocks and unique designations for each block used in the examination procedure.

5.2.4.3.1 Equipment for Inservice Inspection

Manual ultrasonic examination is provided for the preoperational examination and the subsequent inservice examination of the welds in the reactor pressure vessel top and bottom heads including flange-to-vessel weld. Remote ultrasonic scanning, where practical, is used to examine the circumferential, longitudinal, nozzle-to-vessel, and nozzle to safe end welds on the balance of the vessel. As techniques and equipment are developed and improved, it may become beneficial (either due to reduced examination time or radiation exposure) to adopt remotely operated examination equipment to other areas of the reactor pressure vessel.

Remote ultrasonic scanning equipment, if employed for examination of nozzle-to-vessel welds, can be supported and guided from the nozzle or the pipe extending to the nozzle. The equipment provides radial and circumferential motion to the ultrasonic transducer while rotating about the nozzle. Attachment of the equipment can be accomplished through the access openings provided at each nozzle location or, if interfering conditions exist, from the top of the shield wall.

Surface examinations using manual methods are planned; however, should mechanized surface techniques become developed it may be beneficial to adopt such techniques.

Remote visual examination techniques where utilized provide resolution capabilities at least equivalent to that obtained by direct visual observations.

5.2.4.3.2 Coordination of Inspection Equipment with Access Provisions

Development of remotely controlled inspection equipment is followed closely to assure that inservice inspection access provisions are adequate to permit their use.

5.2.4.3.3 Recording and Comparing Data

Manual data recording is performed where manual examinations are performed. Electronic data recording and comparison analysis are employed with automated examination equipment. Each ultrasonic transducer is fed into an individual channel from which the key parameter of the reflectors is recorded. The data to be recorded for both manual and automated methods are:

a. Location

b. Maximum signal amplitude

c. Depth below the scanning surface

d. Length of reflector

The data is compared with data from subsequent examinations to determine the behavior of the reflector.

5.2.4.4 Inspection Intervals

The inspection intervals throughout the service lifetime are in accordance with Subarticle IWB-2400 of ASME Code Section XI. Inservice inspection may be performed during normal plant outages such as during normal refueling shutdown and/or maintenance shutdown occurring during the inspection interval. An inspection schedule for Class 1 system components is developed in accordance with the guidance of ASME Code Section XI, Subarticle IWB-2400.

5.2.4.5 Inservice Examination Program Categories and Requirements

The inservice examination categories and requirements for Class 1 components are in agreement with ASME Code Section XI, Subarticle IWB-2500.

Examination categories and requirements are defined in Technical Specifications, Operational Requirements Manual, and Table IWB-2500-1 of ASME Code Section XI.

5.2.4.6 Evaluation of Examination Results and Repair Procedures

The evaluation of Class 1 component examination results complies with the requirements of Article IWB-3000 of ASME Code Section XI. The repair procedures for Class 1 components comply with the requirements of Article IWA-4000 of ASME Code Section XI.

5.2.4.7 System Leakage and Hydrostatic Pressure Tests

The program for periodic Class 1 system pressure testing complies with the criteria of ASME Code Section XI, Article IWB-5000. The program for repair/replacement system leakage or system hydrostatic tests complies with the criteria of ASME Code Section XI, Article IWA-4540. Visual examinations for evidence of leakage are performed during these tests. Insulation and components need not be removed during the tests.

5.2.4.8 Preservice Inspection Commitment

Prior to the inservice inspection of the plant, the Preservice Program Plan was prepared and applied to all Class 1 systems and components in accordance with the Summer 1978 Addenda to Section XI of the ASME Code with the extent of examination for Examination Category B-J determined by Summer of 1975. The Preservice Inspection Program included the plan, described the scope of the examinations and included isometric drawings and component sketches.

5.2.4.9 Augmented Inservice Inspection for High Energy Piping Systems in Containment Penetration Break Exclusion Regions

An augmented Inservice Examination Program for high energy piping in containment penetration break exclusion regions <Section 3.6.2.1.7> is included in the Inservice Examination Program.

In this region, examination of 100% of the accessible circumferential and longitudinal pipe welds, or a number of these piping welds as determined using the Risk-Informed Program, as documented in the Inservice Examination Plan, will be performed during each inspection interval. Additionally, examination of the accessible welds attaching penetration head fittings to main steam and feedwater process piping will be performed during each inspection interval.

5.2.5 DETECTION OF LEAKAGE THROUGH REACTOR COOLANT PRESSURE BOUNDARY

5.2.5.1 Leakage Detection Methods

The nuclear boiler leak detection system consists of temperature, pressure, flow, airborne gaseous and particulate fission product sensors, and process radiation sensors with associated instrumentation used to indicate leakage from the reactor coolant pressure boundary and, in certain cases, to provide alarms or to initiate signals used for automatic closure of isolation valves to shut off leakage external to the primary containment. The system is designed to be in conformance with NRC <Regulatory Guide 1.45> and reference IEEE Standard 279.

Abnormal leakage from the following systems within the primary containment and within selected areas of the plant outside the primary containment is detected, indicated, and in certain cases alarmed or isolated:

a. Main steam lines.

b. Reactor Water Cleanup (RWCU) System.

c. Residual Heat Removal (RHR) System.

d. Reactor Core Isolation Cooling (RCIC) System.

e. Feedwater System.

f. High Pressure Core Spray (HPCS).

g. Coolant Systems within the primary containment.

h. Low Pressure Core Spray (LPCS).

i. Reactor pressure vessel.

j. Miscellaneous systems.

Leak detection methods used to obtain conformance with <Regulatory Guide 1.45> differ for plant areas inside the drywell as compared to these areas located outside the drywell. These areas are considered separately.

5.2.5.1.1 Detection of Leakage within the Drywell

The detection methods for small unidentified leaks within the drywell include monitoring of floor drain sump inleakage, upper cooler condensate flow rate, and airborne gaseous and particulate radioactivity. The sensitivity of the floor drain sump level and upper cooler condensate flow rate monitors for unidentified leakage within the drywell is 1 gpm within 1 hour. These variables are continuously indicated and/or recorded in the control room. If the unidentified leakage increases to a total of 5 gpm, floor drain sump level and upper cooler condensate flow rate instruments will trip and activate an alarm in the control room. No isolation trip will occur. Fixed-measurement-interval methods are also available, which can provide indication of floor drain sump inleakage. If airborne particulate or gaseous radioactivity levels increase to their monitor alarm setpoints, an alarm will be activated in the control room.

The additional detection methods, of drywell atmosphere pressure and temperature are used to detect gross unidentified leakage. High drywell pressure will alarm and trip the isolation logic which will result in closure of the containment isolation valves.

The detection of small identified leakage within the drywell is accomplished by monitoring of drywell equipment drain sump level inflow rate (gpm). The detection channel will activate an alarm in the control room when the total leak rate reaches 25 gpm. This measurement has a sensitivity for detection of leakage increases of 1 gpm over normal background leakage.

The determination of the source of identified leakage within the drywell is accomplished by monitoring the drain lines to the drywell equipment drain sumps from various potential leakage sources. These include upper containment pool seal drain flow, reactor recirculation pump seal drain flow, valve stem leakoff drain line temperatures, and reactor vessel head seal drain line pressure. Additionally, temperature is monitored in the safety/relief valve discharge lines to the suppression pool to detect leakage through each of the safety/relief valves. All of these monitors, except the reactor recirculation seal drain flow monitor, continuously indicate and/or record in the control room. All of these monitors will trip and activate an alarm in the control room on detection of leakage from monitored components.

Any possible leakage from the reactor vessel head flange is retained in the flange drain line to prevent the leaking steam from scoring the head surface. A pressure transmitter provides an alarm in the control room on high pressure in this line.

Each line that is used to route valve packing leakage to the drain sump is equipped with a temperature transmitter which provides an alarm in the control room on high temperature in the line. Leakage of such a magnitude that it was not being condensed would be indicated by this high temperature alarm. A manually operated solenoid valve provided in each line can then be closed by the operator to isolate the line.

In addition, the drains of the upper two coolers of the drywell air cooling system are equipped with a common flow transmitter which provides an alarm in the control room on high condensate drain flow. High drain flow is indicative of possible reactor coolant pressure boundary leakage.

To minimize the potential for drain system blockage, drywell floor and equipment drain sumps are monitored continuously for level or rate to indicate normal sump operation. Also, pressure switches located downstream of sump pumps, trip the pumps on high discharge pressure (line blockage). An inspection of the drywell and the drain sump areas will be performed prior to closing out the drywell after maintenance.

Excessive leakage inside the drywell (e.g., process line break or loss-of-coolant accident within primary containment) is detected by high drywell pressure, low reactor water level or steam line flow (for breaks downstream of the flow elements). The instrumentation channels for these variables will trip when the monitored variable exceeds a predetermined limit to activate an alarm and trip the isolation logic which will close appropriate isolation valves <Table 5.2-8>.

The alarms, indication and isolation trip functions initiated by the leak detection systems are summarized in <Table 5.2-8> and <Table 5.2-9>.

5.2.5.1.2 Detection of Leakage External to the Drywell (Within Reactor Building)

The detection of leakage within the reactor building but outside the drywell is accomplished by detection of increases in reactor building floor drain sump and reactor building equipment drain sump fillup time and pumpout time. The reactor building floor drain sump monitors detect unidentified leakage increases with a sensitivity of 50 percent of normal background and activate an alarm in the control room when total leakage reaches 5 gpm. The reactor building equipment drain sump monitors detect identified leakage increase with a sensitivity of 50 percent normal background leakage and activate an alarm in the control room when total leakage reaches 25 gpm.

The determination of the source of identified leakage to the reactor building equipment drain sump is accomplished by monitoring flow in the upper containment pool liner drain lines. High flow in a drain line activates an alarm in the control room.

5.2.5.1.3 Detection of Leakage External to Reactor Building

The areas outside the reactor building which are monitored for primary coolant leakage are: equipment areas in the auxiliary building, the main steam tunnel and the turbine building. The process piping for each system to be monitored for leakage is located in compartments or rooms separate from other systems where feasible so that leakage may be detected by area temperature indications. Each leakage detection system will detect leak rates that are less than the established leakage limits.

a. The main steam tunnel is monitored by dual element thermocouples for sensing high ambient temperature in the areas and high differential temperature between the inlet and outlet ventilation ducts which service the individual areas. The temperature elements are located or shielded so that they are sensitive to air temperatures only and not radiated heat from hot piping or equipment. Increases in ambient and/or differential temperature will indicate leakage of reactor coolant into the area. These monitors have sensitivities suitable for detection of reactor coolant leakage into the monitored areas. The temperature trip setpoints are a function of room size and the type of ventilation provided. These monitors provide alarm and indication and recording in the control room and will trip the isolation logic to close selected isolation valves.

b. Leakage detection in the turbine building is accomplished by the use of thermocouples for sensing high ambient temperature in the MSL areas. These monitors also alarm and indicate in the control room and trip the isolation logic to close the main steam line isolation and MSL drain isolation valves before leakage exceeds 280 gpm (32.9 lbm/sec).

c. Leakage detection in each ECCS system compartment is accomplished by monitoring increases in floor drain sump level. These monitors also alarm in the control room.

d. Excess leakage external to the containment (e.g., process line break outside containment) is detected by low reactor water level, high process line flow, high ambient and differential temperature in the piping or equipment areas, high differential flow and low main condenser vacuum. These monitors provide alarm and indication in the control room and trip the isolation logic to cause closure of appropriate system isolation valves on indication of excess leakage <Table 5.2-8>. Differential temperature provides alarm and indication only.

5.2.5.1.4 Intersystem Leakage Monitoring

Radiation monitors are used to detect reactor coolant leakage into cooling water systems supplying the RHR heat exchangers and the reactor water cleanup system (RWCS) heat exchangers. These monitoring channels are part of the process radiation monitoring system. Coolant leakage into the cooling water systems of the RHR systems is monitored using two channels: one for monitoring downstream of equipment in the emergency service water system Loop A and the other for Loop B. Coolant leakage into the cooling water systems supplying the RWCS heat exchangers is monitored by one channel in the nuclear closed cooling water system. Each channel will alarm on high radiation conditions indicating process leakage into the cooling water. No isolation trip functions are performed by this monitor.

Radioactive releases from the ADHR system to the service water system are monitored by the ADHR heat exchanger service water outlet radiation monitor. This channel will alarm on high radiation conditions indicating ADHR leakage into the service water. If a high radiation level is detected, the ADHR system can be manually isolated.

5.2.5.2 Leakage Detection Instrumentation and Monitoring

5.2.5.2.1 Leak Detection Instrumentation and Monitoring Inside Drywell

Leak detection instrumentation and monitoring inside drywell is as follows:

a. Floor Drain Sump Measurement

   The normal design leakage collected in the floor drain sump includes unidentified leakage from the control rod drives, valve flange leakage, component cooling water, air cooler drains, and any leakage not connected to the equipment drain sump. The floor drain sump instrumentation monitors and records sump level in terms of flow rate (gpm). Abnormal leakage rates are alarmed in the main control room. Collection in excess of background leakage would indicate an increase in reactor coolant leakage from an unidentified source.

Two fixed-measurement interval methods exist for determining unidentified drywell leakage rates. First, the leakage rate can be calculated using the change in the drywell floor drain sump level as indicated in the control room. By monitoring the level change over a period of time, the leakage rate can be calculated.

The second fixed-measurement method involves monitoring the drywell floor sump drain pump run time. By determining pump run time over a given period, the leakage rate can be determined if the pump rate is known or can be conservatively estimated.

b. Equipment Drain Sump

The equipment drain sump collects only identified leakage. This sump receives piped drainage from pump seal leakoff, reactor vessel head flange vent drain, and valve stem packing leakoff. Collection in excess of background leakage would indicate an increase in reactor coolant from an identified source. The equipment drain sump instrumentation is similar to that of the floor drain sump and, in addition, monitors sump drain pump fillup time and pumpout time.

c. Cooler Condensate Drain

Condensate from the upper two drywell coolers is routed to the floor drain sump and is monitored by use of a flow transmitter which measures flow in the condensate drain line and sends signals for indication and alarm instrumentation in the control room. An adjustable alarm is set to annunciate on the condensate high flow rate at a level exceeding normal flow rate conditions.

d. Temperature Measurement

The ambient temperature within the drywell is monitored by six single element RTD’S located equally spaced in the vertical direction within the drywell. An abnormal increase in drywell temperature could indicate a leak within the drywell. In addition, the drywell exit end of the containment penetration guard pipe for the main steam line is also monitored for abnormal temperature rise caused by leakage from the main steam line. Ambient temperatures within the drywell are recorded and high average temperatures are alarmed on the leakage detection and isolation system (LD&IS) control room panel.

e. Fission Product Monitoring

This drywell air sampling system is used along with the temperature, pressure, and flow variation method described above to detect leaks in the nuclear system process barrier. The system continuously monitors the drywell and drywell atmosphere for airborne radioactivity (iodine, noble gases and particulates). The sample is drawn directly from the drywell. A sudden increase of activity, which may be attributed to steam or reactor water leakage, is annunciated in the control room. The power supply for the atmospheric monitor is from a vital stub bus which receives power from a divisional bus through an isolation breaker located in Class 1E switchgear. This breaker is tripped upon receipt of a LOCA signal. The operator has the ability to restore power to the bus when required after the LOCA signal has reset <Section 12.3.4>.

f. Drywell Pressure Measurement

The drywell pressure varies slightly during reactor operation and is monitored by pressure sensors. The pressure fluctuates slightly as result of barometric pressure changes and outleakage. A pressure rise above the normally indicated values will indicate a possible leak within the drywell. Pressure exceeding the preset values will be annunciated in the main control room and safety action will be automatically initiated.

g. Reactor Vessel Head Seal

The reactor vessel head closure is provided with double seals with a leakoff connection between seals that is piped through a normally closed manual valve to the equipment drain sump. When leakage through the first seal is detected by an increase in pressure between the seals an alarm in the control room is actuated. The second seal then operates to contain the vessel pressure.

h. Reactor Water Recirculation Pump Seal

Reactor water recirculation pump seal leaks are detected by monitoring flow in the seal drain line. Leakage, indicated by high flow rate, alarms in the control room. The leakage is piped to the equipment drain sump.

i. Safety/Relief Valves

Temperature sensors connected to a multipoint recorder are provided to detect safety/relief valve leakage during reactor operation. Safety/relief valve temperature elements are mounted, using a thermowell, in the safety/relief valve discharge piping several feet downstream from the valve body. Temperature rise above the alarm setpoint is annunciated in the main control room. The nuclear boiler system piping and instrumentation diagram is shown on <Figure 5.1-3>.

j. Valve Stem Packing Leakage

Valve stem packing leaks of power-operated valves 2 inches or larger in the nuclear boiler system, reactor water cleanup system, high pressure core spray, low pressure core spray, reactor core isolation cooling system, residual heat removal system, and recirculation system are detected by monitoring packing leakoff. High temperature is recorded and annunciated by an alarm in the main control room.

k. High Flow in Main Steam Lines (for leaks downstream of flow elements)

High flow in each main steam line is monitored by differential pressure sensors that sense the pressure difference across a flow element in the line. Steam flow exceeding preset values for any of the four main steam lines results in annunciation and isolation closure of all the main steam and steam drain lines.

l. Reactor Water Low Level

The loss of water in the reactor vessel (in excess of makeup) as the result of a major leak from the reactor coolant pressure boundary is detected by using the same nuclear boiler system low reactor water level signals that alarm and isolate selected primary system isolation valves.

m. RCIC Steam Supply Line Flow (for leaks downstream of flow element)

The RCIC steam supply line provides motive power for the operation of the RCIC steam turbine. The line is monitored for abnormal flows. Steam flows exceeding preset values will initiate annunciation and isolation of the RCIC steam supply line.

n. High Differential Pressure Between ECCS Injection Lines (for leakage internal to reactor vessel only)

A break between ECCS injection nozzles and vessel shroud is detected by monitoring the differential pressure between RHR "A" and LPCS, RHR "B" and "C," and HPCS and reactor vessel plenum. Indicator and alarm are located in the main control room.

o. Upper Pool Leakage

Upper pool liner and bellows seal is monitored for leakage by means of a flow transmitter locally mounted on the upper pool drain line. Indicator and alarm are located in the control room.

As the primary method for detecting identified and unidentified leakage, the drywell floor drain sump level and the drywell equipment drain sump pump level will be used to monitor flow rate (gpm) into the sump. Other fixed-measurement-interval methods are also available utilizing sump level changes or pump out times.

Airborne particulate and gaseous radioactivity are monitored in the drywell as a qualitative method for determining high gross unidentified leakage. Correlating particulate and gaseous radioactivity readings with reactor coolant leakage rate is considered impractical in detecting increases in leakage rates of 1 gpm to 3 gpm and also for the maximum allowed sump leakage limit of 5 gpm.

Condensate flow rate from the upper two drywell coolers (Elevation 630’-1”) is also monitored as a method of leak detection. Readout units are in gallons per minute.

<Table 5.2-8> and <Table 5.2-9> summarize the actions taken by each leakage detection function. The tables show that those systems which detect gross leakage initiate immediate automatic isolation. The systems which are capable of detecting small leaks initiate an alarm in the control room or are monitored at appropriate intervals. The operator may manually isolate the leakage source or take other appropriate action. A record of background leakage shall be maintained in the control room. This record shall be kept by the control room operators and will be periodically reviewed to determine if any trends have developed.

Leakage monitoring for drywell equipment drain sump level and drywell floor drain sump level is contained in the ERIS computer. However, this is not the primary display method.

5.2.5.2.2 Leak Detection Instrumentation and Monitoring External to Drywell

The leak detection instrumentation and monitoring external to drywell is as follows:

a. Containment Sump Flow Measurement

   Instrumentation monitors and indicates the amount of unidentified leakage into the reactor building floor drainage system outside the drywell. Background leakage is identified during startup tests. Identified leakage within the reactor building outside the drywell includes the upper containment pool, transfer pool liner and separator liner leakage, which is piped to the containment equipment drain sump. The containment floor and equipment drain sump instrumentation monitors sump drain pump fillup time and pumpout time.

b. Visual and Audible Inspection

Accessible areas are inspected periodically and the temperature and flow indicators discussed above are monitored regularly. Any instrument indication of abnormal leakage will be investigated.

c. Differential Flow Measurement (reactor water cleanup system only)

Because of its arrangement, the reactor water cleanup system uses the differential flow measurement method to detect leakage. The flow into the cleanup system is compared with the flow from the system. An alarm in the control room and an isolation signal are initiated when high differential flow exists between flow into the system and flow back to the reactor vessel indicating that a leak equal to the established leak rate limit may exist.

d. Main Steam Line Area Temperature Monitors

High temperature in the main steam line tunnel areas is detected by dual element thermocouples. Some of the dual element thermocouples are used for measuring main steam tunnel ambient temperatures and are located in the area of the main steam and RCIC steam lines. The remaining dual elements are used in pairs to provide measurement of differential temperature across (inlet to outlet) the tunnel area vent system. All temperature elements are located or shielded so as to be sensitive to air temperatures and not to the radiated heat from hot equipment. One thermocouple of each differential temperature pair is located so as to be unaffected by tunnel temperature. High ambient or high differential temperature will alarm in the control room. High ambient will also provide a signal to close the main steam line and drain line isolation valves, RCIC steam line isolation valves, and the reactor water cleanup system isolation valves. A high temperature or differential temperature alarm may also indicate leakage in the reactor feedwater line which passes through the main steam tunnel. Leak detection in the main steam line area in the turbine building is accomplished by the use of thermocouples for sensing high ambient temperatures.

e. Temperature Monitors in Equipment Areas

Dual element thermocouples are installed in the equipment areas and in the inlet and outlet ventilation ducts to the RCIC, RHR and RWCU equipment rooms for sensing high ambient or high differential temperature. These elements are located or shielded so that they are sensitive to air temperature only and not radiated heat from hot equipment. High ambient and high differential temperature are alarmed in the control room.

f. Intersystem Leakage Monitoring

The intersystem leakage monitoring is included in the process radiation monitoring system to satisfy the requirements of that system.

g. Large Leaks External to the Primary Containment

The main steam line high flow, RCIC steam supply line high flow and reactor vessel low water level monitoring discussed in <Section 5.2.5.2.1>, Items k, l and m, can also indicate large leaks from the reactor coolant piping external to the primary containment.

5.2.5.2.3 Summary

<Table 5.2-8> and <Table 5.2-9> summarize the actions taken by each leakage detection function. The table shows that those systems which detect gross leakage initiate immediate automatic isolation. The systems which are capable of detecting small leaks initiate an alarm in the control room or are monitored at appropriate intervals. The operator can manually isolate the violated system or take other appropriate action. A time delay is provided before automatic isolation of the reactor core isolation cooling system on a high ambient temperature in the main steam tunnel so that the MSIV's and RWCU can be isolated first and thereby preserve the operation of the RCIC system for core cooling. A time delay is also provided for the RWCU differential flow to prevent normal system surges from isolating the system.

The leak detection system is a multi-dimensional system which is redundantly designed so that failure of any single element will not interfere with a required detection of leakage or isolation. In the four division portion of the LD&IS, applied where inadvertent isolation could impair plant performance (e.g., main steamline isolation valves), any single channel or divisional component malfunction will not cause a false indication of leakage or false isolation trip because it will only trip one of four channels. It thus combines a very high probability of operating when needed with a very low probability of operating falsely. The system is testable during plant operation.

5.2.5.3 <u>Indication in Control Room</u>

Leak detection methods are discussed in <Section 5.2.5.1>. Details of the leakage detection system indications are included in <Section 7.6.1>.

5.2.5.4 Limits for Reactor Coolant Leakage

5.2.5.4.1 Total Leakage Rate

The total leakage rate consists of all leakage, identified and unidentified, that flows to the floor drain and equipment drain sumps. The total leakage rate limit is well within the makeup capability of the RCIC system. The total leakage rate limit is established at 30 gpm averaged over the previous 24 hours. The unidentified leakage rate limit is established at 5 gpm.

The total leakage rate limit is low enough to prevent overflow of the sumps. The equipment sump and the floor drain sump, which collect all leakage, are each pumped out by two 50 gpm pumps.

5.2.5.4.2 Identified Leakage Inside Drywell

The pump packing glands, valve stems and other seals in systems that are part of the reactor coolant pressure boundary, and from which normal design identified source leakage is expected, are provided with leakoff drains. Large nuclear system valves inside the primary containment and recirculation pumps are equipped with double seals. Leakage from the primary recirculation pump seals is monitored for flow in the drain line and pipe to the equipment drain sump as described in <Section 5.4.1.3>. Leakage from the main steam safety/relief valves discharging to the suppression pool is monitored by temperature sensors that transmit to the control room. Any temperature increase above the drywell ambient temperature detected by these sensors indicates valve leakage.

Thus, the leakage rates from recirculation pumps, valve stem packings and the reactor vessel head seal, which all discharge to the equipment drain sump, are measured during plant operation.

5.2.5.5 Unidentified Leakage Inside Drywell

5.2.5.5.1 Unidentified Leakage Rate

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

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

The unidentified leakage rate limit is established at 5 gpm rate to allow time for corrective action before the process barrier could be significantly compromised. This 5 gpm unidentified leakage rate is a small fraction of the calculated flow from a critical crack in a primary system pipe <Figure 5.2-15>. Leakage limits are discussed in Technical Specifications.

5.2.5.5.2 Sensitivity and Response Times

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

5.2.5.5.3 Length of Through-Wall Flaw

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

a. Critical Crack Length

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

$$L_c = \frac{15{,}000D}{\sigma_h}$$ (see data correlation on <Figure 5.2-16>)

where:

L_c = critical crack length (in.)
D = mean pipe diameter (in.)
σ_h = nominal hoop stress (psi).

b. Crack Opening Displacement

The theory of elasticity predicts a crack opening displacement of

$$\omega = \frac{2\ L\sigma}{E}$$

where:

L = crack length
σ = applied nominal stress
E = Young's modulus

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

$$C = \sec\left(\frac{\pi}{2}\frac{\sigma_S}{\sigma_f}\right)$$

The crack opening area is given by

$$A = \frac{C\pi\omega L}{4} = \frac{\pi}{2}\left(\frac{L^2\sigma}{E}\right)\ \sec\left(\frac{\pi}{2}\frac{\sigma_s}{\sigma_f}\right)$$

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

c. Leakage Flow Rate

The maximum flow rate for blowdown of saturated water at 1,000 psi is 55 lb/sec-in.^{2}, and for saturated steam the rate is 14.6 lb/sec-in.^{2} (Reference 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 (1,050 psi).

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

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

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

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

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

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

<Figure 5.2-15> shows general relationships between crack length, leak rate, stress, and line size, using the mathematical model described previously. The asterisks denote conditions at which the crack opening displacement is 0.1 in., at which time instability is imminent as noted previously under Item c. This provides a realistic estimate of the leak rate to be expected from a crack of critical size. In every case, the leak rate from a crack of critical size is significantly greater than the 5 gpm criterion.

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

5.2.5.5.4 Margins of Safety

The margins of safety for a detectable flaw to reach critical size are presented in <Section 5.2.5.5.3>. <Figure 5.2-15> shows general relationships between crack length, leak rate, stress and line size using the mathematical model.

5.2.5.5.5 Criteria to Evaluate the Adequacy and Margin of the Leak Detection System.

For process lines that are normally open, there are at least two different methods of detecting abnormal leakage from each system within the nuclear system process barrier located in the primary containment, reactor building and auxiliary building as shown in <Table 5.2-8> and <Table 5.2-9>. The instrumentation is designed so it can be set to provide alarms at established leakage rate limits and isolate the affected system, if necessary, or it is monitored at appropriate intervals. The alarm points are determined analytically or based on measurements of appropriate parameters made during startup and preoperational tests.

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

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

5.2.5.6 Differentiation Between Identified and Unidentified Leaks

<Section 5.2.5.1> describes the systems that are monitored by the leak detection system. The ability of the leak detection system to differentiate between identified and unidentified leakage is discussed in <Section 5.2.5.4>, <Section 5.2.5.5>, and <Section 7.6>.

5.2.5.7 Sensitivity and Operability Tests

Sensitivity, including sensitivity testing and response time of the leak detection system, and the criteria for shutdown if leakage limits are exceeded, are covered in <Section 7.6>.

Testability of the leakage detection system is contained in <Section 7.6>.

5.2.5.8 Safety Interfaces

The Balance of Plant/GE Nuclear Steam Supply System safety interfaces for the leak detection system are the signals from the monitored balance of plant equipment and systems which are part of the nuclear system process barrier, and associated wiring and cable lying outside the nuclear steam supply system equipment.

5.2.5.9 Testing and Calibration

Provisions for testing and calibration of the leak detection system are covered in <Chapter 14>.

5.2.5.10 Regulatory Guide 1.45 Compliance

The detection of leakage through the reactor coolant pressure boundary, described in the preceding sections, meets the intent of the <Regulatory Guide 1.45>. Details of compliance are discussed in the following.

a. Leakage is separated into identified and unidentified categories and total flow rate for each is independently monitored, thus meeting Position C.1 of <Regulatory Guide 1.45>.

b. Small unidentified leaks (5 gpm and less) inside the drywell are detected by temperature changes, pressure changes, sump fill rate activities, fission product monitoring, and upper drywell cooler condensate flow monitoring. Large leaks are also detected by changes in reactor water level and changes in flow rates in process lines.

   The 5 gpm leakage rate is the plant Technical Specification limit on unidentified leakage inside the drywell. The leak detection system is fully capable of monitoring the flow rates of 1 gpm and is thus in compliance with Position C.2 of <Regulatory Guide 1.45>.

c. By monitoring floor drain sump level (flow rate), airborne particulate radioactivity, cooler condensate flow rate and airborne gaseous radioactivity, Position C.3 is satisfied.

   Isolation and/or alarm of affected systems and the detection methods used are summarized in <Table 5.2-8> and <Table 5.2-9>.

d. Monitoring of coolant for radiation in the RHR and RWCU heat exchangers satisfies Position C.4. For system details, see <Section 7.6>.

e. The floor drain sump monitoring, and the upper air cooler condensate monitoring systems are designed to detect leakage rates of 1 gpm within 1 hour, thus meeting Position C.5. The fission products monitoring subsystem is not designed to detect leakage rates of 1 gpm within 1 hour.

f. The fission products monitoring subsystem is qualified for SSE. The drywell floor drain sump level, equipment drain sump level and air cooler drain rate instrumentation are capable of performing their functions following seismic events that do not require plant shutdown, thus meeting Position C.6.

g. Leakage detection indicators and alarms for the drain sump, cooler condensate flow rate monitoring and radioactivity monitoring systems are provided in the main control room. Procedures for the fixed-measurement methods of determining drywell unidentified leakage rates will be available to the operators for converting the sump level changes and/or pump run times to a leakage rate. Procedures for converting the drywell floor drain sump rate monitoring instrumentation and cooler condensate flow rate monitoring instrumentation are not necessary since these indicators are expressed as gallons per minute. There is no attempt to correlate radioactivity monitoring indication to leakage flow rate due to the uncertainties involved. This satisfies the procedural requirements of Position C.7 of this guide.

h. The leakage detection system is equipped with provisions to permit testing for operability and calibration during plant operation using the following methods:

   1. Simulation of signals into trip units

   2. Comparing channel "A" to channel "B" of the same leak detection method (drywell temperature and pressure monitoring)

   3. Checking operability by comparing one method versus another (air cooler condensate flow versus floor drain sump level (flow rate)).

   4. Comparing one method versus another (sump fill up versus pump out and particulate monitoring, air cooler condensate flow versus sump fill up rate)

   5. Continuous monitoring of floor drain sump level is provided.

   These methods satisfy Position C.8.

i. Technical Specifications limit unidentified leakage to 5 gpm and total leakage (identified plus unidentified) to 30 gpm. This satisfies Position C.9.

5.2.6 REFERENCES FOR SECTION 5.2

1. Qualification of One-Dimensional Core Transient Model for BWR, NEDO-24154, dated October 1978.

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

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

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

5. "Investigation and Evaluation of Cracking in Austenitic Stainless Steel Piping of Boiling Water Reactor Plants," <NUREG-75/067>, NRC/PCSG, dated October 1975.

TABLE 5.2-1

REACTOR COOLANT PRESSURE BOUNDARY COMPONENTS/PIPING/INSTALLATION

APPLICABLE CODE CASES

Code Case	Title
1141 1141-1	Foreign Produced Steel (RPV)
1332 1332-6	Requirements for Steel Forgings, Section III and Section VIII, Division 2
1334	Requirements for Corrosion Resisting Steel Bars and Shapes, Section III
1335 1335-10	Requirements for Bolting Material, Section III
1337	Requirements for Special Type 403 Modified Forgings or Bars; Section III
1344	Requirements for Nickel-Chromium Age-Hardenable Alloys (Alloy x750), Section III
1361 1361-2	Socket Welds, Section III (CRD Housing)
1384	Requirements for Precipitation Hardening Alloy Bars and Forgings
1388	Requirements for Stainless Steel - Precipitation Hardening
1390	Requirements for Nickel-Chromium, Age - Hardenable Alloy for Bolting - Section III
1401	Welding Repairs to Cladding of Section III Component after Final Post-weld Heat Treatment
1433	Normalized and tempered 2 1/4 and 3 Cr low alloy forgings for Code Construction under Section I, III and VIII, Divisions 1 and 2
1434	Post-weld Heat Treatment of SA-487, Class 8N Steel Casting, Section III

TABLE 5.2-1 (Continued)

Code Case	Title
1456 (N-15)	Substitution of Ultrasonic Examination for Progressive Penetrant or Magnetic Particle Examination of Partial Penetration and Oblique Nozzle Attachment Welds, Section III
1492	Post-weld Heat Treatment Section I, III and VIII, Divisions 1 and 2
1508	Allowable Stresses, Design Intensity and/or Yield Strength Values, Section I, III and VIII, Divisions 1 and 2
1515	Ultrasonic Examination of Ring Forgings for Shell Section III Class I Vessels
1516	Welding of Seats or Minor Internal Permanent Attachments in Valve for Section III Applications
1519	Use of A105-71 in lieu of SA-105 (Mainsteam Piping)
1535 1535-2	Hydrostatic Testing of Section III, Class I Valves (MSIV)
1557 1557-1	Steel Products Refined by Secondary Remelting Section III and Section VIII, Divisions 1 and 2 (RPV)
1562	Qualification of Forming and Bending Processes for Classes 1, 2 and 3 Components
1567 (N38)	Testing Lots of Carbon and Low Alloy Steel Covered Electrodes, Section III
1571	Additional Material for SA 234 Carbon Steel Fittings, Section III (Mainsteam Piping)
1572	Fracture Toughness Class 1 Components Section III (RPV)
1578	SB-167, Ni-Cr-Fe (Alloy 600) Pipe or Tube, Section III
1588 (N46)	Electro-Etching of Section III Code Symbol
1620	Stress Category for Partial Penetration Welded Penetrations, Section III, Class 1 Construction (RPV)

TABLE 5.2-1 (Continued)

Code Case	Title
1622	PWHT of Repair Welds in Carbon Steel Castings Section III, Class 1, 2 and 3 (MSIV)
1637	Effective Date for Compliance with NA 3700 of Section III
1644-4 (N71) 1644-5 1644-6	Additional Materials for Component Supports and Alternate Design Requirements for Bolted Joints, Section III, Division 1, Subsection NF, Class 1, 2, 3, and MC Construction
1644-7 1644-8 1644-9	Additional Materials for Component Supports, Section III, Division 1, Subsection NF, Class 1, 2 and 3, and MC Component Supports
1644-11 1644-14	Additional Materials for Component Supports Fabricated by Welding, Section III, Division 1, Class 1, 2, 3, and MC
1651	Interim Requirements for Certification of Component Supports, Section III, Subsection NF
1682-1	Alternate Rules for Material Manufacturers and Suppliers, Section III, Subarticle NA-3700
1683	Bolt Holes for Section III, Class 1, 2, 3, and MC Component Supports
1690	Stock Materials for Section III Construction, Section III, Division 1
1706	Data Report Forms for Component Supports, Section III, Class 1, 2 and 3
1724	Deviation from the Specified Silicon Ranges in ASME Material Specifications, Section III, Division 1, and VIII, Division 1 and 2
1728	Steel Structural Shapes and Small Material Products for Component Supports, Section III, Division 1 Construction
1729	Minimum Edge Distance Bolting - Class 1, 2 and 3
1734	Weld Design of Component Supports - Class 1, 2, 3, and MC

TABLE 5.2-1 (Continued)

Code Case	Title
1768	Permanent Attachment to Containment Vessels - Class MC, Section III, Division 1 (Mechanical Penetrations)
1810 (N-172)	Testing Lots of Carbon Steel Solid Bare Welding Electrode or Wire, Section III, Division 1, 2, 3, MC, and CS
1819-1	Use of Type XM-19 for Construction, Section III, Division 1, Class 1, 2 and 3
1820 (N-177)	Alternate Ultrasonic Examination Technique, Section III, Division 1
2142	F-Number Grouping for Ni-Cr-Fe, Classification UNS N06052 Filler Metal, Section IX
2143	F-Number Grouping for Ni-Cr-Fe, Classification UNS W86152 Welding Electrode, Section IX
N-3-9 (1335-9) N-3-10 (1335-10)	Requirements for Bolting Materials, Section III
N-30 (1539) N-30-1 (1539-1)	Metal Bellows and Metal Diaphragm Stem Seal Valves Section III, Division 1, Class 1, 2 and 3
N-101 (1712)	Name Plates and Stamping for Section III, Division 1, Class 1, 2, 3, and MC; Construction as Referenced in NA-8300
N-154 (1791)	Projection Resistance Welding of Valve Seats Section III, Division 1, Class 1, 2 and 3 Valves
N-174 (1812)	Size of Fillet Welds for Socket Welding of Piping, Section III, Division 1
N-176	Use of Type XM-19 for Construction, Section III, Division 1, Class 1, 2 and 3 (Power Range Detector Dry Tube)
N-180	Examination of Springs for Class 1 Component Standard Supports, Section III, Division 1
N-207	Use of Modified SA-479 Type XM-19 for Section III, Division 1, Class 1, 2 and 3, or CS Construction (CRD)

TABLE 5.2-1 (Continued)

Code Case	Title
N-225	Certification and Identification of Material for Component Supports, Section III, Division 1
N-226	Temporary Attachment of Thermocouples, Section III, Division 1, Class 1, 2 and 3 Component Construction
N-233	Alternate Rules for PWHT of P-No. 6 Group 4 Material for Section III, Division 1, Class 1, 2 or 3 Construction
N-242	Materials Certification Section III, Division 1, Classes 1, 2, 3, MC, and CS Construction
N-243	Boundaries Within Castings Used for Core Support Structures, Section III, Division 1, Class CS
N-247	Certified Design Report Summary for component standard supports Section III, Division 1, Class 1, 2, 3 and MC.
N-249	Additional Materials for Component Supports Fabricated Without Welding, Section III, Division 1, Subsection NF, Class 1, 2, 3, and MC Component Supports
N-252	Low Energy Capacitive Discharge Welding Method for Temporary or Permanent Attachments to Components and Supports, Section III, Division 1
N-272	Compiling Data Report Records, Section III, Division 1
N-275	Repair of Welds Section III, Division 1
N-282	Name Plates for Valves, Section III, Division 1, Class 1, 2 and 3 Construction
N-328	Thermit Brazing or Welding of Nonstructural Attachments, Section III, Division 1
N-393	Repair Welding Structural Steel Rolled Shapes and Plates for Component Supports, Section III, Division 1
N-413	Minimum size of Fillet Welds for Subsection NF Linear Type Supports, Section III, Division 1

TABLE 5.2-1 (Continued)

Code Case	Title
N-411-1	Alternative Damping Values for Response Spectra Analysis of Classes 1, 2 and 3 Piping, Section III, Division 1.
N-416-1	Alternative Pressure Test Requirements for Welded Repairs or Installation of Replacement Items by Welding, Class 1, 2, and 3, Section XI, Division 1
N-504 (modified)	Alternative Rules for Repair of Class 1, 2, and 3 Austenitic Stainless Steel Piping, Section XI, Division 1

TABLE 5.2-1a

CODE CASE N-242 MATERIALS

PULLMAN POWER PRODUCTS (N-8405)

Fabrication F-Sheet No.	F-Sheet Item No.	Assembly Piece Mark No.
108	1	0-G41-G-FC-1-IB
109	1,2,3	0-G41-G-FC-2-IB
121	5	0-G41-G-FC-14-IB
122	5	0-G41-G-FC-15-IB
125	5	0-G41-G-FC-18-IB
127	1	0-G41-G-FC-20-IB
128	1,2	0-G41-G-FC-21-IB
134	1,2	0-G41-G-FC-27-IB
135	1	0-G41-G-FC-28-IB
136	1,2,3	0-G41-G-FC-29-IB
137	1,2,3,5	0-G41-G-FC-30-IB
138	1,2	0-G41-G-FC-31-IB
139	1,2	0-G41-G-FC-32-IB
140	1,2,4	0-G41-G-FC-33-IB
1343	3	0-P45-G-ESW-92-CC
1379	3	0-P45-G-ESW-154-CC
1381	2,3	1-P45-G-ESW-166-AB
1382	3	1-P45-G-ESW-167-AB
1384	2	1-P45-G-ESW-171-AB
1386	2	1-P45-G-ESW-173-AB
1396	2	1-P45-G-ESW-189-AB
1398	4	0-P45-G-ESW-193-IB
1404	5	1-P45-G-ESW-214-AB
1406	3	1-P45-G-ESW-216-AB
1407	2	1-P45-G-ESW-217-AB
1408	2	1-P45-G-ESW-218-AB
1410	3	1-P45-G-ESW-220-AB
1411	2	1-P45-G-ESW-221-AB
1434	3	0-P45-G-ESW-271-CC
1456	2	0-P45-G-ESW-344-IB
1459	2,3	0-P45-G-ESW-348-CC
1505	4	0-P42-G-ECC-20-CC
1508	5	0-P42-G-ECC-23-IB
1510	6	0-P42-G-ECC-25-IB
1513	4	0-P42-G-ECC-28-CC
1515	6	0-P42-G-ECC-30-CC
1519	6	0-P42-G-ECC-34-IB
1521	5	0-P42-G-ECC-36-IB
1529	1	0-P42-G-ECC-44-CC
1530	1	0-P42-G-ECC-45-CC
1531	1	0-P42-G-ECC-46-CC

TABLE 5.2-1a (Continued)

PULLMAN POWER PRODUCTS (N-8405)

Fabrication F-Sheet No.	F-Sheet Item No.	Assembly Piece Mark No.
1532	6	0-P42-G-ECC-47-IB
1533	3	0-P42-G-ECC-48-IB
1536	2	0-P42-G-ECC-51-IB
1537	2	0-P42-G-ECC-52-IB
1540	8	0-P42-G-ECC-55-IB
1546	4	0-P42-G-ECC-62-CC
1548	1	0-P42-G-ECC-64-CC
1549	1	0-P42-G-ECC-65-CC
1550	1	0-P42-G-ECC-66-CC
1562	3	0-P42-G-ECC-84-CC
1567	4	0-P42-G-ECC-89-CC
1570	6	0-P42-G-ECC-92-CC
1579	3	0-P42-G-ECC-102-CC
1581	4	0-P42-G-ECC-104-CC
1583	7	0-P42-G-ECC-106-CC
1584	1	0-P42-G-ECC-107-CC
1585	1	0-P42-G-ECC-108-CC
1586	1	0-P42-G-ECC-109-CC
1589	4	0-P42-G-ECC-112-CC
1596	4	0-P42-G-ECC-118-CC
1597	4	0-P42-G-ECC-119-CC
1599	4	0-P42-G-ECC-121-CC
1603	3	0-P42-G-ECC-125-CC
1604	4	0-P42-G-ECC-126-CC
1608	5	0-P42-G-ECC-130-CC
1616	5	0-P42-G-ECC-138-CC
1619	3	0-P42-G-ECC-141-CC
1621	3	0-P42-G-ECC-143-CC
1623	2	0-P42-G-ECC-145-IB
1627	2	0-P42-G-ECC-150-AB
1630	3	1-P42-G-ECC-156-AB
1631	3	1-P42-G-ECC-157-AB
1636	3	0-P42-G-ECC-162-CC
1638	3	0-P42-G-ECC-165-CC
1640	2	0-P42-G-ECC-167-IB
1644	3	1-P42-G-ECC-172-AB
1647	3	1-P42-G-ECC-178-AB
1648	3	1-P42-G-ECC-179-AB
1652	3	0-P42-G-ECC-184-CC
1654	3	0-P42-G-ECC-186-CC
1656	2	0-P42-G-ECC-188-IB
1661	2	1-P42-G-ECC-193-AB
1674	3	1-P42-G-ECC-206-AB

TABLE 5.2-1a (Continued)

PULLMAN POWER PRODUCTS (N-8405)

Fabrication F-Sheet No.	F-Sheet Item No.	Assembly Piece Mark No.
1679	2	0-P42-G-ECC-211-IB
1681	3	0-P42-G-ECC-213-CC
1682	3	0-P42-G-ECC-214-CC
1684	5	0-P42-G-ECC-217-CC
1686	3	0-P42-G-ECC-220-CC
1688	2	0-P42-G-ECC-222-CC
1689	3	0-P42-G-ECC-223-CC
1690	2	0-P42-G-ECC-224-CC
1691	3	0-P42-G-ECC-225-IB
1692	3	0-P42-G-ECC-226-IB
1693	3	0-P42-G-ECC-227-IB
1693	3	0-P42-G-ECC-227A-IB
1694	2	0-P42-G-ECC-228-IB
1695	3	0-P42-G-ECC-229-IB
1695	3	0-P42-G-ECC-229A-IB
1697	5	0-P42-G-ECC-231-IB
1698	2	0-P42-G-ECC-232-IB
1699	3	0-P42-G-ECC-233-IB
1703	5	0-P42-G-ECC-237A-IB
1704	3	0-P42-G-ECC-238-IB
1705	2	0-P42-G-ECC-239-IB
1706	5	0-P42-G-ECC-240-IB
1708	3	0-P42-G-ECC-242-IB
1708	3	0-P42-G-ECC-242A-IB
1709	2	0-P42-G-ECC-243-IB
1710	3	0-P42-G-ECC-244-IB
1710	3	0-P42-G-ECC-244A-IB
1711	3	0-P42-G-ECC-245-IB
1712	3	0-P42-G-ECC-246-IB
1713	2	0-P42-G-ECC-247-CC
1714	3	0-P42-G-ECC-248-CC
1715	2	0-P42-G-ECC-249-CC
1716	3	0-P42-G-ECC-250-CC
1717	3	0-P42-G-ECC-251-CC
1718	2	0-P42-G-ECC-252-CC
1719	3	0-P42-G-ECC-253-CC
1720	2	0-P42-G-ECC-254-CC
1721	3	0-P42-G-ECC-255-IB
1722	3	0-P42-G-ECC-256-IB
1723	3	0-P42-G-ECC-257-IB
1723	3	0-P42-G-ECC-257A-IB
1724	2	0-P42-G-ECC-258-IB
1725	2	0-P42-G-ECC-259-IB

TABLE 5.2-1a (Continued)

PULLMAN POWER PRODUCTS (N-8405)

Fabrication F-Sheet No.	F-Sheet Item No.	Assembly Piece Mark No.
1726	2	0-P42-G-ECC-260-IB
1727	2	0-P42-G-ECC-261-IB
1732	2	0-P42-G-ECC-266-IB
1733	2	0-P42-G-ECC-267-IB
1734	2	0-P42-G-ECC-268-IB
1735	2	0-P42-G-ECC-269-IB
1736	3	0-P42-G-ECC-270-IB
1736	3	0-P42-G-ECC-270A-IB
1737	3	0-P42-G-ECC-271-IB
1738	3	0-P42-G-ECC-272-IB
1739	2	0-P42-G-ECC-273-CC
1740	3	0-P42-G-ECC-274-CC
1741	3	0-P42-G-ECC-275-CC
1742	2	0-P42-G-ECC-276-IB
1745	2	0-P42-G-ECC-281-IB
1747	2	0-P42-G-ECC-286-IB
1750	2	0-P42-G-ECC-291-IB
1752	3	0-P42-G-ECC-296-CC
1753	3	0-P42-G-ECC-297-CC
1754	3	0-P42-G-ECC-298-CC
1755	3	0-P42-G-ECC-299-CC
1758	4	0-P42-G-ECC-302-IB
1762	3,5	0-P42-G-ECC-306-IB
1765	2	1-E51-G-RCIC-2-AB
1766	2	1-E51-G-RCIC-3-AB
1767	2	1-E51-G-RCIC-3A-AB
1770	1	1-E51-G-RCIC-6-AB
1775	2	1-E51-G-RCIC-11-AB
1776	2	1-E51-G-RCIC-12-AB
1779	3	1-E51-G-RCIC-15-AB
1780	2	1-E51-G-RCIC-16-AB
1781	2	1-E51-G-RCIC-17-AB
1783	4	1-E51-G-RCIC-19-AB
1806	3	1-E51-G-RCIC-42-RB
1820	2	1-G42-G-SPCU-4-AB
1821	2,3	1-G42-G-SPCU-5-AB
1822	2	1-G42-G-SPCU-6-AB
1832	2	1-G42-G-SPCU-16-AB
1833	2	1-G42-G-SPCU-17-AB
1836	2	1-G42-G-SPCU-20-IB
1904	3	0-P45-G-ESW-33-CC
1908	4	0-P45-G-ESW-38-CC
1911	2	0-P45-G-ESW-41-DGB

TABLE 5.2-1a (Continued)

PULLMAN POWER PRODUCTS (N-8405)

Fabrication F-Sheet No.	F-Sheet Item No.	Assembly Piece Mark No.
1912	2	0-P45-G-ESW-42-DGB
1913	2	0-P45-G-ESW-43-DGB
1936	2	0-P45-G-ESW-162-DGB
1937	3	0-P45-G-ESW-164-DGB
1948	2	0-P45-G-ESW-209-DGB
1954	4	0-P45-G-ESW-256-CC
1955	3	0-P45-G-ESW-257-CC
1957	3	0-P45-G-ESW-259-DGB
1959	3	0-P45-G-ESW-262-DGB
1977	3	0-P45-G-ESW-307-DGB
1979	4	0-P45-G-ESW-310-DGB
1986	4	0-P45-G-ESW-325-CC
1990	3	0-P45-G-ESW-330-DGB
1993	2	0-P45-G-ESW-333-DGB
2000	3	0-P45-G-ESW-357-DGB
2001	2	0-P45-G-ESW-358-DGB
2005	3	0-P45-G-ESW-369-CC
2010	4	0-P45-G-ESW-376-CC
2013	2	0-P45-G-ESW-379-DGB
2014	3	0-P45-G-ESW-380-DGB
2017	3	0-P45-G-ESW-384-DGB
2019	1	1-E61-G-ILR-2-AB
2020	2	1-E61-G-ILR-3-AB
2021	3	1-E61-G-ILR-4-AB
2028	2	1-E61-G-ILR-11-AB
2029	3	1-E61-G-ILR-12-AB
2046	1,2	0-G60-G-FDS-1-IB
2055	4,5	1-G61-G-LRS-8-RB
2056	2	1-G61-G-LRS-9-RB
2057	2	1-G61-G-LRS-10-RB
2102	4,5	1-G61-G-LRS-57-RB
2104	3	1-G61-G-LRS-58-RB
2105	3,4	1-G61-G-LRS-59-RB
2436	2	1-E21-G-LPC-8-AB
2442	4	1-E21-G-LPC-14-AB
2451	10	1-E21-G-LPC-23-AB
2454	3	1-E21-G-LPC-25-AB
2455	3	1-E21-G-LPC-26-AB
2456	2	1-E21-G-LPC-26A-AB
2458	3	1-E21-G-LPC-28-AB
2459	2	1-E21-G-LPC-29-AB
2460	2	1-E21-G-LPC-30-AB
2566	4	1-G41-G-FC-126-AB

TABLE 5.2-1a (Continued)

PULLMAN POWER PRODUCTS (N-8405)

Fabrication F-Sheet No.	F-Sheet Item No.	Assembly Piece Mark No.
2568	4	1-G41-G-FC-128-AB
2578	3	0-G41-G-FC-139-IB
2579	2	0-G41-G-FC-140-IB
2588	3	0-G41-G-FC-149-IB
2589	5	0-G41-G-FC-150-IB
2591	4	0-G41-G-FC-152-IB
2609	4,5	1-G41-G-FC-170-AB
2610	4	1-G41-G-FC-171-AB
2616	5	1-G41-G-FC-177-AB
2618	2	1-G41-G-FC-179-AB
2619	5,6	1-G41-G-FC-180-AB
2622	7	0-G41-G-FC-187-IB
2624	3	0-G41-G-FC-189-IB
2627	5	0-G41-G-FC-192-IB
2635	3	0-G41-G-FC-200-IB
2636	3	0-G41-G-FC-201-IB
2646	1	0-G41-G-FC-211A-IB
2649	2	0-G41-G-FC-214-IB
2663	1	0-G41-G-FC-227A-IB
2672	4	0-G41-G-FC-236-IB
3140	3	1-P43-G-NCC-14-RB
3141	1	1-P43-G-NCC-15-RB
3182	3	1-P43-G-NCC-54-RB
3183	1	1-P43-G-NCC-55-RB
3315	2	0-R45-G-DG-1-YD
3317	2	0-R45-G-DG-4-YD
3321	2	0-R45-G-DG-10-YD
3334	3	0-R45-G-DG-26-YD
3335	2	0-R45-G-DG-27-YD
3337	3	0-R45-G-DG-30-YD
3338	3	0-R45-G-DG-31-YD
3340	3	0-R45-G-DG-33-YD
3341	3	0-R45-G-DG-34-YD
3348	2	0-R45-G-DG-44-YD
3350	3	0-R45-G-DG-47-YD
3354	2	0-R45-G-DG-52-YD
3357	2	0-R45-G-DG-56-YD
3374	3	0-R45-G-DG-77-YD
3375	2	0-R45-G-DG-78-YD
3377	3	0-R45-G-DG-80-YD
3378	2	0-R45-G-DG-81-YD
3380	3	0-R45-G-DG-83-YD
3389	4	0-R45-G-DG-94-YD

TABLE 5.2-1a (Continued)

PULLMAN POWER PRODUCTS (N-8405)

Fabrication F-Sheet No.	F-Sheet Item No.	Assembly Piece Mark No.
3390	2	0-R45-G-DG-95-YD
3392	4	0-R45-G-DG-98-YD
3395	3	0-R45-G-DG-102-YD
3406	2	0-R45-G-DG-115-YD
3408	3	0-R45-G-DG-117-YD
3409	2	0-R45-G-DG-118-YD
3411	3	0-R45-G-DG-120-YD
3412	2	0-R45-G-DG-121-YD
3414	3	0-R45-G-DG-123-YD
3769	3	1-G50-G-LRW-44-RB
3770	4	1-G50-G-LRW-45-RB
3841	2	0-P45-G-ESW-439-YD
3843	2	0-P45-G-ESW-441-YD
3845	2	0-P45-G-ESW-443-YD
3847	3	0-P45-G-ESW-445-YD
3849	3,4	0-P45-G-ESW-447-YD
3857	3	0-P45-G-ESW-455-YD
3859	2	0-P45-G-ESW-457-YD
3861	2	0-P45-G-ESW-459-YD
3881	3	0-P45-G-ESW-479-YD
3920	3	0-P45-G-ESW-518-YD
3921	2	0-P45-G-ESW-519-YD
3923	2	0-P45-G-ESW-521-YD
3924	3	0-P45-G-ESW-522-YD
3947	4	0-P45-G-ESW-545-YD
3958	3	0-P45-G-ESW-556-YD
3959	2	0-P45-G-ESW-557-YD
4139	1,5	1-G33-G-RWCU-39-RB
4142	1,2,6	1-G33-G-RWCU-42-RB
4152	5	1-G33-G-RWCU-52-RB
4360	3	1-E12-G-RH-228-RB
4369	4	1-E12-G-RH-236-RB
4445	4	1-E12-G-RH-39-AB
4447	6	1-E12-G-RH-41-AB
4448	5	1-E12-G-RH-42-AB
4449	4	1-E12-G-RH-43-AB
4450	4	1-E12-G-RH-44-AB
4453	2	1-E12-G-RH-47-AB
4454	3	1-E12-G-RH-48-AB
4456	4	1-E12-G-RH-50-AB
4458	4	1-E12-G-RH-52-AB
4459	2	1-E12-G-RH-53-AB
4460	8	1-E12-G-RH-53A-AB

TABLE 5.2-1a (Continued)

PULLMAN POWER PRODUCTS (N-8405)

Fabrication F-Sheet No.	F-Sheet Item No.	Assembly Piece Mark No.
4461	3	1-E12-G-RH-54-AB
4462	4	1-E12-G-RH-55-AB
4467	4	1-E12-G-RH-60-AB
4468	4	1-E12-G-RH-61-AB
4469	4,5	1-E12-G-RH-62-AB
4470	6	1-E12-G-RH-63-AB
4473	4	1-E12-G-RH-66-AB
4485	7	1-E12-G-RH-77-AB
4492	7	1-E12-G-RH-84-AB
4500	4	1-E12-G-RH-92-AB
4503	5	1-E12-G-RH-95-AB
4504	2	1-E12-G-RH-96-AB
4505	2	1-E12-G-RH-97-AB
4506	3	1-E12-G-RH-98-AB
4507	4	1-E12-G-RH-99-AB
4510	5	1-E12-G-RH-102-AB
4511	4	1-E12-G-RH-103-AB
4512	3	1-E12-G-RH-104-AB
4513	2	1-E12-G-RH-105-AB
4514	5	1-E12-G-RH-106-AB
4515	2	1-E12-G-RH-107-AB
4525	3	1-E12-G-RH-116-AB
4530	4	1-E12-G-RH-121-AB
4566	4	1-E12-G-RH-157-AB
4567	2	1-E12-G-RH-158-AB
4574	3	1-E12-G-RH-165-AB
4576	4,5	1-E12-G-RH-167-AB
4585	5	1-E12-G-RH-176-AB
4586	5	1-E12-G-RH-177-AB
4595	4	1-E12-G-RH-186-AB
4596	2	1-E12-G-RH-187-AB
4607	6	1-E12-G-RH-198-AB
4608	3	1-E12-G-RH-199-AB
4609	2	1-E12-G-RH-200-AB
4611	3	1-E12-G-RH-202-AB
4614	4	1-E12-G-RH-205-AB
4622	4	1-E12-G-RH-213-AB
4623	2	1-E12-G-RH-214-AB
4624	4	1-E12-G-RH-215-AB
4625	4	1-E12-G-RH-216-AB
4626	2	1-E12-G-RH-217-AB
4627	4	1-E12-G-RH-218-AB
4755	2	0-P45-G-ESW-328-DGB

TABLE 5.2-1a (Continued)

PULLMAN POWER PRODUCTS (N-8405)

Fabrication F-Sheet No.	F-Sheet Item No.	Assembly Piece Mark No.
4814	5	0-P45-G-ESW-310B-DGB
4816	4	0-P45-G-ESW-358A-DGB
4819	5	0-P45-G-ESW-379B-DGB
4822	4	0-P45-G-ESW-380B-DGB
4824	3	0-P45-G-ESW-384A-DGB
4825	5	0-P45-G-ESW-384B-DGB
4828	2	0-P45-G-ESW-558-CC
4832	4	0-P45-G-ESW-563-DGB
4833	2	0-P45-G-ESW-564-DGB
4834	2	0-P45-G-ESW-565-DGB
4835	5	0-P45-G-ESW-44-DGB
4846	2	1-E15-G-CS-2-RB
4849	1	1-E15-G-CS-5-RB
4852	1	1-E15-G-CS-8-RB
4853	1	1-E15-G-CS-9-RB
4855	1	1-E15-G-CS-11-RB
4860	1	1-E15-G-CS-16-RB
4861	1	1-E15-G-CS-17-RB
4862	1	1-E15-G-CS-18-RB
4864	1	1-E15-G-CS-20-RB
4865	1	1-E15-G-CS-20A-RB
4868	1	1-E15-G-CS-23-RB
4869	1	1-E15-G-CS-24-RB
4873	2	1-E15-G-CS-28-RB
4874	1	1-E15-G-CS-29-RB
4876	1	1-E15-G-CS-31-RB
4882	1	1-E15-G-CS-37-RB
4886	1	1-E15-G-CS-40A-RB
4888	1	1-E15-G-CS-42-RB
4889	1	1-E15-G-CS-43-RB
4890	1	1-E15-G-CS-44-RB
4892	1	1-E15-G-CS-46-RB
4893	1	1-E15-G-CS-47-RB
4895	1	1-E15-G-CS-49-RB
4896	2	0-P47-G-CCW-1-CC
4898	3	0-P47-G-CCW-3-CC
4900	5	0-P47-G-CCW-6-CC
4904	2	0-P47-G-CCW-10-CC
4906	5	0-P47-G-CCW-12-CC
4907	2	0-P47-G-CCW-13-CC
4910	2	0-P47-G-CCW-20-CC
4911	2	0-P47-G-CCW-21-CC
4914	2	0-P47-G-CCW-24-CC

TABLE 5.2-1a (Continued)

PULLMAN POWER PRODUCTS (N-8405)

Fabrication F-Sheet No.	F-Sheet Item No.	Assembly Piece Mark No.
4916	2	0-P47-G-CCW-26-CC
4932	5	0-P47-G-CCW-40-CC
4943	2	0-P47-G-CCW-50-CC
4944	2	0-P47-G-CCW-51-CC
4946	4,5	0-P47-G-CCW-57-CC
4947	2	0-P47-G-CCW-58-CC
4948	3	0-P47-G-CCW-59-CC
4949	3,4	0-P47-G-CCW-60-CC
4950	3	0-P47-G-CCW-61-CC
4951	2	0-P47-G-CCW-62-CC
4952	2	0-P47-G-CCW-63-CC
4953	3	0-P47-G-CCW-64-CC
4956	3	0-P47-G-CCW-67-CC
4957	2,3	0-P47-G-CCW-68-CC
4961	3	0-P47-G-CCW-72-CC
4962	2	0-P47-G-CCW-73-CC
4963	2	0-P47-G-CCW-74-CC
4964	3	0-P47-G-CCW-75-CC
4965	2	0-P47-G-CCW-76-CC
4966	3,4	0-P47-G-CCW-77-CC
4967	2	0-P47-G-CCW-78-CC
4968	3	0-P47-G-CCW-79-CC
4969	2	0-P47-G-CCW-80-CC
4970	2	0-P47-G-CCW-81-CC
4972	3	0-P47-G-CCW-87-CC
4973	2	0-P47-G-CCW-88-CC
4975	2	0-P47-G-CCW-91-CC
4976	3	0-P47-G-CCW-92-CC
4978	3	0-P47-G-CCW-94-CC
4979	5	0-P47-G-CCW-95-CC
4991	3	0-P47-G-CCW-107-CC
4994	2	0-P47-G-CCW-110-CC
4995	3	0-P47-G-CCW-111-CC
4997	2	0-P47-G-CCW-117-CC
4998	2	0-P47-G-CCW-118-CC
5010	4	0-P47-G-CCW-130-CC
5011	3	0-P47-G-CCW-131-CC
5013	2	0-P47-G-CCW-136-CC
5014	2	0-P47-G-CCW-137-CC
5015	3	0-P47-G-CCW-138-CC
5017	3	0-P47-G-CCW-141-CC
5019	2	0-P47-G-CCW-143-CC
5020	3	0-P47-G-CCW-144-CC

TABLE 5.2-1a (Continued)

PULLMAN POWER PRODUCTS (N-8405)

Fabrication F-Sheet No.	F-Sheet Item No.	Assembly Piece Mark No.
5022	2	0-P47-G-CCW-146-CC
5024	3	0-P47-G-CCW-149-CC
5026	2	0-P47-G-CCW-154-CC
5027	3	0-P47-G-CCW-155-CC
5028	3	0-P47-G-CCW-156-CC
5029	2	0-P47-G-CCW-157-CC
5030	2	0-P47-G-CCW-158-CC
5031	2	0-P47-G-CCW-159-CC
5035	3	0-P47-G-CCW-163-CC
5036	3	0-P47-G-CCW-164-CC
5038	3	0-P47-G-CCW-166-CC
5040	2	0-P47-G-CCW-168-CC
5042	2	0-P47-G-CCW-170-CC
5047	3	0-P47-G-CCW-175-CC
5048	2	0-P47-G-CCW-176-CC
5050	2	0-P47-G-CCW-179-CC
5052	2	0-P47-G-CCW-184-CC
5053	2	0-P47-G-CCW-185-CC
5062	2	0-P47-G-CCW-195-CC
5075	3	0-P47-G-CCW-208-CC
5619	3	0-P47-G-CCW-180-CC
5620	3	0-P47-G-CCW-183-CC
5646	4	2-P45-G-ESW-38-AB
5668	2	2-P45-G-ESW-66-AB
5671	3	2-P45-G-ESW-74-AB
5673	3	2-P45-G-ESW-76-AB
5694	3,4	2-P45-G-ESW-104-AB
5696	2	2-P45-G-ESW-112-AB
5698	3	2-P45-G-ESW-114-AB
5749	6	0-P45-G-ESW-600-ESWPE
5753	2	0-P45-G-ESW-604-ESWPE
5755	2	0-P45-G-ESW-606-ESWPE
5757	2	0-P45-G-ESW-608-ESWPE
5759	2	0-P45-G-ESW-610-ESWPE
5762	3	0-P45-G-ESW-613-ESWPE
5764	3	0-P45-G-ESW-615-ESWPE
5766	3	0-P45-G-ESW-617-ESWPE
5768	3	0-P45-G-ESW-619-ESWPE
5772	4	0-P45-G-ESW-623-ESWPE
6198	3	0-R45-G-DG-169-DGB
6199	3	0-R45-G-DG-170-DGB
6200	2	0-R45-G-DG-171-DGB
6201	3	0-R45-G-DG-172-DGB
6202	2	0-R45-G-DG-173-DGB

TABLE 5.2-1a (Continued)

PULLMAN POWER PRODUCTS (N-8405)

Fabrication F-Sheet No.	F-Sheet Item No.	Assembly Piece Mark No.
6204	2	0-R45-G-DG-175-DGB
6205	3	0-R45-G-DG-176-DGB
6206	3	0-R45-G-DG-177-DGB
6207	3	0-R45-G-DG-178-DGB
6208	2	0-R45-G-DG-179-DGB
6209	3	0-R45-G-DG-180-DGB
6210	2	0-R45-G-DG-181-DGB
6213	3	0-R45-G-DG-184-DGB
6215	3	0-R45-G-DG-186-DGB
6217	3	0-R45-G-DG-188-DGB
6218	3	0-R45-G-DG-189-DGB
6220	3	0-R45-G-DG-191-DGB
6223	4	0-R45-G-DG-194-DGB
6224	3	0-R45-G-DG-195-DGB
6225	4	0-R45-G-DG-196-DGB
6226	3	0-R45-G-DG-197-DGB
6227	4	0-R45-G-DG-198-DGB
6228	3	0-R45-G-DG-199-DGB
6229	3	0-R45-G-DG-200-DGB
6230	3	0-R45-G-DG-201-DGB
6231	3	0-R45-G-DG-202-DGB
6234	3	0-P45-G-ESW-625-DGB
6236	3	0-P45-G-ESW-626-DGB
6935	4	0-P45-G-ESW-628-YD
7109	2	1-P45-G-ESW-121-AB
10157	3	1-P45-G-ESW-213B-AB
10158	3	1-P45-G-ESW-214B-AB
10511	2	1-P54-G-FP-131-AB
10512	3	1-P54-G-FP-132-AB
13455	2	2-M51-G-CGC-3-RB
13456	2	2-M51-G-CGC-4-RB
13457	2	2-M51-G-CGC-5-RB
13460	2	2-M51-G-CGC-8-RB
13463	2	2-M51-G-CGC-10A-RB
13464	3	2-M51-G-CGC-11-RB
13465	2	2-M51-G-CGC-12-RB
13466	2	2-M51-G-CGC-13-RB
13471	2	2-M51-G-CGC-18-RB
13472	2	2-M51-G-CGC-19-RB
13474	2	2-M51-G-CGC-22-RB
13477	4	2-M51-G-CGC-24-RB
13479	2	2-M51-G-CGC-30-RB
13480	2	2-M51-G-CGC-31-RB
13481	2	2-M51-G-CGC-32-RB

TABLE 5.2-1a (Continued)

PULLMAN POWER PRODUCTS (N-8405)

Fabrication F-Sheet No.	F-Sheet Item No.	Assembly Piece Mark No.
13496	4	1-M14-G-CVDP-3-RB
13509	4	2-M14-G-CVDP-4-RB
13570	1	1-E15-G-CS-1A-RB
13571	1	1-E15-G-CS-26A-RB
13736	3	2-E12-G-RH-123-AB
13737	3	2-E12-G-RH-124-AB
13744	3	2-E12-G-RH-133-AB
13745	3	2-E12-G-RH-134-AB
13780	3	2-E12-G-RH-55-AB
13788	3	2-E12-G-RH-73-AB
13789	2	2-E12-G-RH-74-AB
13793	3	2-E12-G-RH-78-AB
13794	3	2-E12-G-RH-79-AB
13797	2	2-E12-G-RH-82-AB
13802	5	2-E12-G-RH-87-AB
13804	4,5	2-E12-G-RH-89-AB
13805	4	2-E12-G-RH-90-AB
13806	5	2-E12-G-RH-91-AB
13810	5	2-E12-G-RH-97-AB
13834	7	2-E12-G-RH-142-AB
13844	4	2-E12-G-RH-164-AB
13845	2	2-E12-G-RH-170-AB
13858	6	2-E12-G-RH-68-AB
13861	3,7	2-E12-G-RH-71-AB
13871	7	2-E12-G-RH-166-AB
13874	3	2-E12-G-RH-169-AB
13876	9	2-E12-G-RH-176-AB
13879	5,6	2-E12-G-RH-192-AB
13880	3	2-E12-G-RH-193-AB
13931	2	2-E12-G-RH-203-AB
13932	2	2-E12-G-RH-204-AB
14196	2	2-E21-G-LPC-11-AB
14197	2	2-E21-G-LPC-12-AB
14321	4	2-P43-G-NCC-146-RB
14333	3	2-P43-G-NCC-161-RB
14766	2	2-G42-G-SPCU-17A-AB
14769	3	2-G42-G-SPCU-20-AB
14772	3	2-G42-G-SPCU-23-AB
14809	3	2-G41-G-FC-76-AB
14810	3	2-G41-G-FC-77-AB
14811	3,4	2-G41-G-FC-78-AB
14813	2	2-G41-G-FC-81-AB
14815	2	2-G41-G-FC-83-AB

TABLE 5.2-1a (Continued)

PULLMAN POWER PRODUCTS (N-8405)

Fabrication F-Sheet No.	F-Sheet Item No.	Assembly Piece Mark No.
14817	3	2-G41-G-FC-85-AB
14818	3	2-G41-G-FC-86-AB
14819	4	2-G41-G-FC-87-AB
14821	3	2-G41-G-FC-91-AB
14822	4,5	2-G41-G-FC-92-AB
14823	3,4	2-G41-G-FC-93-AB
14824	5,6	2-G41-G-FC-94-AB
15000	2	2-E22-G-HPC-34-AB
15197	2	2-G36-G-RFD-80-RB

N-8405 TOTAL = 568 ASSEMBLIES

TABLE 5.2-1a (Continued)

PULLMAN POWER PRODUCTS (N-7691)

Fabrication F-Sheet No.	F-Sheet Item No.	Assembly Piece Mark No.
5	2	0-P47-G-CCW-63R-00
6	2	0-P47-G-CCW-73R-00
7	2	0-P47-G-CCW-158R-CC
23	2	0-P47-G-CCW-78R-CC
24	4	0-P47-G-ESW-97R-CC
33	3	1-E12-G-RH-105R-AB
198	4,5	1-E12-G-RH-I478-1-AB
199	5,6	1-E12-G-RH-X1248-1-AB

N-7691 TOTAL = 8 ASSEMBLIES

TABLE 5.2-1a (Continued)

MATERIALS PROCURED BY GENERAL ELECTRIC

ITEM	MPL NUMBER
SNUBBER	1B21G006
SNUBBER	2B21G006
SNUBBER	1B33G006
SNUBBER	2B33G006
BELLOWS	1F42G001
BELLOWS	2F42G001

TOTAL = 6 ITEMS

MATERIALS PROCURED BY WESTINGHOUSE ELECTRIC

	MPL Number
ELECTRICAL PENETRATION BULKHEAD MATERIAL	1R72S001
ELECTRICAL PENETRATION BULKHEAD MATERIAL	1R72S002
ELECTRICAL PENETRATION BULKHEAD MATERIAL	1R72S003
ELECTRICAL PENETRATION BULKHEAD MATERIAL	1R72S004
ELECTRICAL PENETRATION BULKHEAD MATERIAL	1R72S005
ELECTRICAL PENETRATION BULKHEAD MATERIAL	1R72S006
ELECTRICAL PENETRATION BULKHEAD MATERIAL	1R72S007
ELECTRICAL PENETRATION BULKHEAD MATERIAL	1R72S008
ELECTRICAL PENETRATION BULKHEAD MATERIAL	1R72S009
ELECTRICAL PENETRATION BULKHEAD MATERIAL	1R72S010
ELECTRICAL PENETRATION BULKHEAD MATERIAL	1R72S011
ELECTRICAL PENETRATION BULKHEAD MATERIAL	1R72S012
ELECTRICAL PENETRATION BULKHEAD MATERIAL	1R72S013
ELECTRICAL PENETRATION BULKHEAD MATERIAL	1R72S014
ELECTRICAL PENETRATION BULKHEAD MATERIAL	1R72S015
ELECTRICAL PENETRATION BULKHEAD MATERIAL	1R72S016
ELECTRICAL PENETRATION BULKHEAD MATERIAL	1R72S017
ELECTRICAL PENETRATION BULKHEAD MATERIAL	1R72S018
ELECTRICAL PENETRATION BULKHEAD MATERIAL	1R72S019
ELECTRICAL PENETRATION BULKHEAD MATERIAL	1R72S020
ELECTRICAL PENETRATION BULKHEAD MATERIAL	1R72S021
ELECTRICAL PENETRATION BULKHEAD MATERIAL	1R72S022
ELECTRICAL PENETRATION BULKHEAD MATERIAL	1R72S023
ELECTRICAL PENETRATION BULKHEAD MATERIAL	1R72S024
ELECTRICAL PENETRATION BULKHEAD MATERIAL	1R72S025
ELECTRICAL PENETRATION BULKHEAD MATERIAL	1R72S026
ELECTRICAL PENETRATION BULKHEAD MATERIAL	1R72S027
ELECTRICAL PENETRATION BULKHEAD MATERIAL	1R72S028
ELECTRICAL PENETRATION BULKHEAD MATERIAL	1R72S029
ELECTRICAL PENETRATION BULKHEAD MATERIAL	1R72S030
ELECTRICAL PENETRATION BULKHEAD MATERIAL	1R72S031
ELECTRICAL PENETRATION BULKHEAD MATERIAL	1R72S032

TABLE 5.2-1a (Continued)

MATERIALS PROCURED BY WESTINGHOUSE ELECTRIC

	MPL Number
ELECTRICAL PENETRATION BULKHEAD MATERIAL	1R72S033
ELECTRICAL PENETRATION BULKHEAD MATERIAL	1R72S035
ELECTRICAL PENETRATION BULKHEAD MATERIAL	1R72S036
ELECTRICAL PENETRATION BULKHEAD MATERIAL	1R72S038

TOTAL = 36 ITEMS

TABLE 5.2-2

NUCLEAR SYSTEM SAFETY/RELIEF SETPOINTS USED IN THE INITIAL CYCLE ANALYSIS

(Nominal Value)

No. of Valves	Spring Set Pressure (psig)	ASME Rated Capacity @ 103% Spring Set Pressure (lb/hr each)	Relief Pressure Set Pressure (psig)	Low-Low Set Relief No. of Valves	Low-Low Set Relief Setpoint Close/Re-Open
8	1,165	895,000			
6	1,180	906,000			
5	1,190	913,000			
1			1,103	1	926/1,033
9			1,113[1]	1	936/1,073
9			1,123[1]	4	946/1,113

NOTE:

[1] Closing setpoint is 100 psi below opening setpoint for non-low-low set valves.

TABLE 5.2-3

SYSTEMS WHICH MAY INITIATE DURING OVERPRESSURE EVENT

System	Initiating/Trip Signal(s)[1]
Reactor Protection System	Reactor trips "OFF" on High Flux
RCIC	"ON" when Reactor Water Level ≥L2
	"OFF" when Reactor Water Level ≤L8
HPCS	"ON" when Reactor Water Level ≥L2
	"ON" when Drywell Pressure ≥2 psig
	"OFF" when Reactor Water Level ≤L8
Recirculation System	"OFF" when Reactor Water Level <L2
	"OFF" when Reactor Pressure >1,125 psig
RWCU	"OFF" when Reactor Water Level <L2

NOTE:

[1] Vessel levels are shown on <Figure 5.3-7>. Trip settings are analytical limits. Refer to the Operational Requirements Manual for actual setpoints.

TABLE 5.2-4

SEQUENCE OF EVENTS FOR
INITIAL CYCLE MSIV CLOSURE EVENT WITH FLUX[1]
SCRAM <FIGURE 5.2-3> OVERPRESSURIZATION
PROTECTION ANALYSIS

Time-Sec	Events
0	Closure of all main steam isolation valves (MSIV) was initiated.
0.3	MSIVs reached 90% open. Failure of direct position scram was assumed.
1.6	Neutron flux reached the high APRM flux scram setpoint and initiated reactor scram.
2.1	Reactor dome pressure reached the setpoint of recirculation pump trip.
2.1	Reactor dome pressure reached the Group 1 safety/relief valves pressure setpoint (power-actuated mode). Only one half of valves in this group was assumed functioning.
2.3	Steamline pressure reached the Group 1 safety/relief valves pressure setpoint (spring-action mode). Valves which were not opened in this power-actuated mode were opened.
2.4	Recirculation pump initiated to coastdown.
2.8	All safety/relief valves opened in either power-actuated mode or spring action mode due to high pressure.
2.9	Vessel bottom pressure reached its peak value.
3.0	MSIVs completely closed.
>10 (est)	Safety/relief valves opened in their spring-action mode closed.
>20 (est)	Wide-range sensed water level reached L2 setpoint. HPCS and RCIC flow entered reactor vessel. Safety valves closed and reopen cyclicly.

NOTE:

[1] For the current reload safety analysis sequence of events for this transient refer to <Appendix 15B>, Reload Safety Analysis.

TABLE 5.2-5

REACTOR COOLANT PRESSURE BOUNDARY MATERIALS

Component	Form	Material	Specification (ASTM/ASME)
Reactor Vessel			
Shell	Plate	Low Alloy Steel	SA-533 Grade B, Class I
Bottom Head	Plate	Low Alloy Steel	SA-533 Grade B, Class I
Top Head	Plate	Low Alloy Steel	SA-533 Grade B, Class I
Main Closure Flanges	Forging	Low Alloy Steel	SA-508 Class II
Nozzles over 2" Nominal Size	Forging	Low Alloy Steel	SA-508 Class II
Nozzles 2" and under except nozzle core ΔP nozzle	Forging	Stainless Steel	SA-336 Class F8
Drain Nozzle	Forging	Carbon Steel	SA-508 Class I
Core ΔP Nozzle	Forging	Inconel	SB-166
Nozzle Safe Ends	Forging	Inconel	SB-166
Nozzle Safe Ends	Forging	Carbon Steel	SA-508 Class I
Nozzle Safe Ends	Forging	Stainless Steel	SA-366 Class F8
Bolting, Studs Nuts and Washers	Forging	Alloy Steel	SA-540 Grade B23 or Grade 24 Class 3
Main Steam Piping - ASME Code, Class I			
Elbow	Fitting	Carbon Steel	SA-234, WPBW
Pipe	Welded	Carbon Steel	SA-155, Gr. KCF70CL1
	Plate	Carbon Steel	SA-516, Gr. 70
Pipe	Seamless	Carbon Steel	SA-106, Gr. B
Flange	Forging	Carbon Steel	SA-105
Nozzles	Forging	Carbon Steel	SA-105
Lugs	Plate	Carbon Steel	SA-516, Gr. 70
Studs	Bolting	Low Alloy	SA-193, Gr. B7
Nuts	Bolting	Low Alloy	SA-194, Gr. 7
Hydraulic Nuts with Spherical Washers[1]	Bolting	Alloy Steel	SA-540, Grade B23 Class 3

TABLE 5.2-5 (Continued)

Component	Form	Material	Specification (ASTM/ASME)
Safety/Relief Valve Piping - ASME Code, Class III			
Pipe	Seamless	Carbon Steel	SA-106, Gr. B
Ball Joint	Fitting	Carbon Steel	SA-234, Gr. WPB
Elbow	Fitting	Carbon Steel	SA-234, Gr. WPB
Flange	Forging	Carbon Steel	SA-105
Nozzle	Forging	Carbon Steel	SA-105, Code Case 1519
	Forging	Carbon Steel	SA-181, Gr. II
Studs	Bolting	Low Alloy	SA-193, Gr. B7
Nut	Bolting	Low Alloy	SA-194, Gr. 2H
Hydraulic Nuts with Spherical Washers[1]	Bolting	Alloy Steel	SA-540, Grade B23 Class 3
Attachment	Plate	Carbon Steel	SA-516, Gr. 70

TABLE 5.2-5 (Continued)

Component	Form	Material	Specification (ASTM/ASME)
Recirculation Piping - ASME Code, Class I			
Pipe	Welded	Stainless	SA-358, Gr. 304, CL I
Elbow	Fitting	Stainless	SA-403, Gr. WP304 or WPW304
	Plate	Stainless	SA-240, Gr. 304
Nozzle	Fitting	Stainless	SA-403, Gr. WP 304
Flange	Forging	Stainless	SA-182, Gr. F316
Lug	Plate	Stainless	SA-240, Gr. 304
Bolt	Bolting	Low Alloy	SA-193, Gr. B7
Nut	Bolting	Low Alloy	SA-194, Gr. 7
Control Rod Drive			
Flanges, Plugs	Forging	Stainless Steel	SA-182
Nut, Base	Bar	XM-19	SA-479
Indicator tube	Pipe	Stainless Steel	SA-312
Housing	Tube	Stainless Steel	SA-312
	Tube	Inconel 600	SB-167
	Flange (Forging)	Stainless Steel	SA-182
	Welds	Stainless Steel	SFA 5.9, SFA 5.4 Type 308 or 316
	Welds	Inconel	SFA 5.11 ER NiCrFe-3 SFA 5.14 ER NiCr-3
Incore Housing	Tube	Inconel 600	SB-167
	Flange (Forging)	Stainless Steel	SA-182
	Welds	Inconel	SFA 5.11 ER NiCrFe-3 SFA 5.14 ER NiCr-3
MSIV			
Body	Cast	Carbon Steel	SA-216 Gr. WCB
Disc	Forging	Steel	SA-350 Gr. LF-2
Stem	Rod	Stainless Steel	SA-564 Gr. 630
Studs	Bolting	Alloy Steel	SA-540 B23 CL5
Nuts	Bolting	Alloy Steel	SA-540 B23 CL5
Cover	Forging	Forged Steel	SA-105
Main Steam Flow Element			
Instrument Nozzle	Forged	Carbon Steel	SA-105

TABLE 5.2-5 (Continued)

Component	Form	Material	Specification (ASTM/ASME)
Safety/Relief Valve			
Body	Casting	Carbon Steel	SA-352 LCB
Seat	Forging	Carbon Steel	SA-350 LF2
Disk	Casting	Carbon Steel	SA-351 CF3A
Recirculation Gate Valves			
Body	Cast	Stainless Steel	SA-351, GR-CF8M
Bonnet	Cast	Stainless Steel	SA-351, GR-CF8M
Disc	Cast	Stainless Steel	SA-351 - CF3A
Bolts	Bar	Carbon Steel	SA-194 - GR7
Nuts	Bar	Carbon Steel	SA-193 - B7
Stem	Bar	Stainless Steel	SA-564 Type 630 Cond 1150
Recirculation Flow Control Valve			
Body	Casting	Stainless Steel	SA-351, GR-CF8M
Housing	Casting	Stainless Steel	SA-351, GR-CF8M
Covers	Casting	Stainless Steel	SA-351, GR-CF8M
Bonnet	Casting	Stainless Steel	SA-351, GR-CF8M
Recirculation Pump			
Pump Case	Cast	Stainless Steel	SA-351 GR-CF8
Case Stud	Bar	Alloy Steel	SA-540 GR-B23,CL4
Case Nut	Bar	Stainless Steel Alloy Steel	SA-194 GR7 or SA-540 GR B23CL4
Cover	Forging	Carbon Steel	SA-105 Clad with 308SS
Pump Heater-Cooler Assembly			
Assembly, Cooler			
Housing, Cooler	Forging	Stainless Steel	SA182, GR. F316
Cylinder, Outer	Forging	Stainless Steel	SA182, GR. F316
Cylinder, Enclosing	Forging	Stainless Steel	SA182, GR. F316
Union, .75, 3000 lb	Forging	Stainless Steel	SA182, GR. F316
Pipe, .75, Sched.80	Pipe	Stainless Steel	SA182, GR. F316
Thermowells	Bar	Stainless Steel	SA182, GR. F316

TABLE 5.2-5 (Continued)

Component	Form	Material	Specification (ASTM/ASME)
Pump Heater-Cooler Assembly (continued)			
Housing, Thermowell	Bar	Stainless Steel	SA182, GR. F316
Pipe, 1 25, Sched 80	Pipe	Stainless Steel	SA182, GR. F316
Housing, Heater	Forging	Stainless Steel	SA182, GR. F316
Stud, 1.5-8 x 12.0	Bar	Alloy Steel	SA540, GR. B23, CL. 4
Nut, Hex, 1.5-8	Bar	Stainless Steel	SA194, GR. 7
Cap Screw, Hx Hd, 1.25-8 x 4.5	Bar	Stainless Steel	SA540, GR. B23, CL. 4
Seal Assembly			
Pressure Breakdown Device (No. 1 Static Seal Housing)	Forging	Stainless Steel	SA-182, Gr. F304
Seal Flange	Forging	Stainless Steel	SA-182, Gr. F316
Seal Flange Weld Plug	Bar, or Forging	Stainless Steel	SA-479, Type 316, or SA-182, Gr. F316
Thermowell	Bar, or Forging	Stainless Steel	SA-479, Type 316, or SA-182, Gr. F316

NOTE:

(1) Hydraulic Nuts (or HydraNuts) may be installed as an equivalent alternate or replacement for the SRV inlet and outlet flange nuts.

TABLE 5.2-6

OPERATIONAL BWR WATER CHEMISTRY[1,2,3,7] MEASUREMENTS

	Concentrations - Parts per Billion (ppb) Iron	Copper	Chloride	Oxygen	Conductivity µmho/cm 25°C	pH 25°C	Zinc
Condensate (1)	-	-	-	See Note[4]	<0.5	~7	-
Condensate Treatment Effluent (2)	-	-	-		<0.1	~7	-
Feedwater (3)	See Note[4]	See Note[4]	-	See Note[4]	<0.1	~7	≤1.0
Reactor Water (4)							
a. Normal Operation	-	-	<200	100-300 <5[8]	<1.0	~7	<7[5] or <10[6]
b. Shutdown	-	-	<100		<2.0	~7	-
c. Hot Standby	-	-	<100		<2.0	~7	-
d. Depressurized	-	-	<500	8000	<10.0	~7	-
Steam (5)	-	-		10,000-30,000	-		-
Control Rod Drive Cooling Water (6)	See Note[4]	-	<20	See Note[4]	≤0.1	~7	-

NOTES:

(1) Numerals in parentheses refer to locations delineated on <Figure 5.2-13>.

(2) Values in table for Iron, Copper, Oxygen, Ph, Zinc, Feedwater Conductivity and Control Rod Drive Cooling Water Conductivity and Chloride are typical measurements during normal plant operation.

(3) Values in table for Reactor Water Conductivity and Chloride, Condensate Conductivity and Condensate Treatment Effluent Conductivity are maximum values during normal plant operation or during operational condition specified.

(4) These parameters are controlled by chemistry procedures.

(5) Natural Zinc

(6) Depleted Zinc

(7) Normal Water Chemistry (NWC) unless otherwise indicated.

(8) Hydrogen Water Chemistry (HWC).

TABLE 5.2-7

WATER SAMPLE LOCATIONS

Sample Origin	Sensor Location	Indicator Location	Recorder Location	Conductivity (μmho/cm) Range	Alarm Setpoint Low	Alarm Setpoint High	Minimum[1] Accuracy
Reactor Water Recirculation Loop	Sample Line	Sample Station	Control Room	0-10	0.03	1.0	±15% or ±0.10 whichever is least restrictive
Reactor Water Cleanup System Inlet	Sample Line	Sample Station	Control Room	0-1	0.03	1.0	±15% or ±0.05 whichever is least restrictive
Reactor Water Cleanup System Outlets	Sample Line	Sample Station	Control Room	0-1	0.03	0.1	±15% or ±0.02 whichever is least restrictive
Control Rod Drive System	Sample Line	Sample Station	Control Room	0-1	0.03	0.2	±15% or ±0.02 whichever is least restrictive

NOTE:

(1) The accuracy is expressed as percent of reading. A ± band is established to account for accuracy of instrument when water chemistry parameters are low in the operating range. The instruments are sensitive to within or less than the accuracy, and are periodically calibrated against laboratory calibration instruments.

TABLE 5.2-8

SUMMARY OF ISOLATION/ALARM OF SYSTEM MONITORED
AND THE LEAK DETECTION METHODS USED

(Summary of Isolation Signals and Alarms[3] System Isolation vs Variable Monitored)

Variable Monitored	System Isolated				
	Main Steam	RHR	RCIC	RWCU	Balance of Plant
Reactor Vessel Water Level[4][2]	1	3		2	2
Reactor Pressure[2]		I[6]			
Turbine Building Leak Detection	I				
MS Tunnel Ambient Temp, High	I		I	I	
MS Tunnel Differential Temp, High[A]					
MS Line Flow Rate, High	I				
Drywell Pressure, High[2]		I[7]	I[5]		I
RHR Equipment Area Ambient Temp, High		I	I		
RHR Equipment Area Differential Temp, High[A][8]					
RCIC Equipment Area Ambient Temp, High			I		
RCIC Equipment Area Differential Temp, High[A][8]					
RCIC Exhaust Diaphragm Pressure, High[2]			I		
RCIC Steam Supply Differential Pressure (High Flow)			I		
RCIC Steam Supply Differential Pressure (Instr Line Break)			I		
RWCU Process Piping Differential Flow, High				I	
RWCU Equipment Area Ambient Temp, High				I	
RWCU Equipment Area Differential Temp, High[A]					

REFERENCES:

[A] Alarm only.
[I] Isolate, alarm, and indicate (or record).

TABLE 5.2-8 (Continued)

NOTES:

(1) Systems or selected valves within the system that isolate.
(2) These leak detection signals are provided by other systems.
(3) An alarm is associated with each isolation signal.
(4) Numerals in this row correspond to reactor water levels as shown on condensate and feedwater Specification MPL-C34 and are levels at which isolation valves of the related system are closed.
(5) RCIC turbine exhaust vacuum breaker line valves only.
(6) Shutdown cooling mode.
(7) Except shutdown cooling mode.
(8) Effective only when room cooler is running.

TABLE 5.2-9

SUMMARY OF ISOLATION/ALARM OF SYSTEM MONITORED
AND THE LEAK DETECTION METHODS USED

(Summary of Variable Trip Alarms Leakage Source vs Generated Variables)

	Source of Leakage (Inside Drywell/Outside Drywell)													
Affected Variable Monitor	Main Steam Line	RCIC Steam Line	RCIC Steam Line	RWCU Water	HPCS Water	LPCS Water	Recirc Pump Seal	Feed-Water	RHR Water	Reactor Vessel Head Seal	Upper Cont. Pool	Misc. Leaks	Valve Stem Packing	RCIC Water
Leakage, Inside Drywell/Outside Drywell	X/X	X/X	NA/X	X/X	X/X	X/X	X/X	X/X	X/X	X/X	X/X	X/X	X/X	X/X
Drywell Pressure, High	A/NA	A/NA		A/NA			A/NA	A/NA	A/NA					
Reactor Water Level, Low	A/A	A/A	NA/A	A/A				A/A	A/NA					
Floor Drain Sump Flow Rate, High	A/A	A/NA		A/A	A/NA	A/NA		A/A	A/A					
Floor Drain Sump Level	A/NA	A/NA		A/NA	A/NA	A/NA		A/NA	A/NA					
Equipment Drain Sump Flow Rate, High							A/NA				A/NA	A/A	A/NA	A/NA
Fission Product Radiation, High	A/NA	A/NA		A/NA			A/NA	A/NA						
Drywell Temperature, High	A/NA	A/NA		A/NA			A/NA	A/NA	A/NA					
Safety/Relief Valve Discharge Pipe Temp, High	A/NA													
MSL Guard Pipe Temp, High	A/NA													
Valve Stem Leakoff Temp, High													A/NA	
Recirc Pump Seal Flow, High							A/NA							
Vessel Head Seal Pressure, High										A/NA				
Air Cooler Condensate Flow, High	A/NA	A/NA		A/NA				A/NA	A/NA					
Flow Rate, High	A/A	A/A	NA/A											
Sump or Drain Flow, High (Equip. Area)	NA/A	NA/A	NA/A	NA/A	NA/A	NA/A		NA/A	NA/A		NA/A	NA/A	NA/A	
MSL Tunnel Ambient and Differential Temp, High	NA/A	NA/A	NA/A	NA/A				NA/A						
Equipment Area Ambient and Differential Temp, High		NA/A[1]	NA/NA	NA/A					NA/A					
RWCU Differential Flow, High				NA/A										
Pool Seal Drain Flow, High											A/A			
Intersystem Leakage (Radiation), High														
ECCS Injection on Line Leakage (Internal to Reactor Vessel) Differential Pressure				A/NA	A/NA	A/NA			A/A					

REFERENCES:

A. Alarm and indicate (or record) only.
B. Indicate (or record) only.
X. Location of leakage source.
NA Not applicable.

NOTE:

[1] Differential temperature measurement effective only when room cooler is running.

5.3 REACTOR VESSEL

5.3.1 REACTOR VESSEL MATERIALS

5.3.1.1 Materials Specifications

The materials used in the reactor pressure vessel and appurtenances are shown in <Table 5.2-5> together with the applicable specifications.

5.3.1.2 Special Processes Used for Manufacturing and Fabrication

The reactor pressure vessel is primarily constructed from low alloy, high strength steel plate and forgings. Plates are ordered to ASME SA-533 Grade B, Class 1, and forgings to ASME SA-508, Class 2. These materials are melted to fine grain practice and are supplied in the quenched and tempered condition. Further restrictions include a requirement for vacuum degassing to lower the hydrogen level and improve the cleanliness of the low alloy steels. Materials used in the core beltline region also specify limits of 0.12 percent maximum copper and 0.015 percent maximum phosphorus content in the base materials, and a 0.10 percent maximum copper and 0.025 percent maximum phosphorus content in weld materials.

Studs, nuts and washers for the main closure flange are ordered to ASME SA-540, Grade B23 or Grade B24. Welding electrodes are low hydrogen type ordered to ASME SFA 5.5.

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

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

Submerged arc and manual stick electrode welding processes are employed. Electroslag welding is not permitted. Preheat and interpass temperatures employed for welding of low alloy steel meet or exceed the requirements of ASME Code Section III, Subsection NB-4600. Post weld heat treatment at 1,100°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 Code Section III, Subsection NB-5320. In addition, all welds are given a supplemental ultrasonic examination.

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

During RF07, a weld overlay was applied to the feedwater nozzle to safe-end weld 1B13-N4C-KB. The overlay is designed as a full structural overlay in accordance with the recommendations of <NUREG-0313>, Revision 2 (forwarded by <Generic Letter 88-01>), ASME Code Case N-504, and Section XI of the ASME Boiler and Pressure Vessel Code, 1989 Edition (Paragraph IWB-3640). Examination of the final weld overlay was performed in accordance with ASME Code Case N-504 (modified for welding of P-1 and P-43 materials) and <NUREG-0313>, Revision 2 (modified as necessary for examination of Ni-Cr-Fe overlays). Pressure testing of the weld overlay repair was performed in accordance with ASME Section XI, 1989 Edition, no Addenda per ASME Code Case N-416-1 (Reference PNPP letter PY-CEI/NRR-1851L and NRC Safety Evaluation Response dated 2/10/95).

5.3.1.3 Special Methods for Nondestructive Examination

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

5.3.1.4 Special Controls for Ferritic and Austenitic Stainless Steels

5.3.1.4.1 Compliance with Regulatory Guides

Compliance with regulatory guides is as follows:

a. <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>.

b. <Regulatory Guide 1.34>, Control of Electroslag Weld Properties

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

c. <Regulatory Guide 1.43>, Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components

Reactor pressure vessel specifications require that all low alloy steel be produced to fine grain practice. The requirements of this regulatory guide are not applicable to BWR vessels.

d. <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>.

e. <Regulatory Guide 1.50>, Control of Preheat Temperature for Welding of Low-Alloy Steel

Preheat controls are discussed in <Section 5.2.3.3.2>.

f. <Regulatory Guide 1.71>, Welder Qualification for Areas of Limited Accessibility

Qualification for areas of limited accessibility is discussed in <Section 5.2.3.3.2>.

g. Unit 1: <Regulatory Guide 1.99> Revision 2, Radiation Embrittlement of Reactor Vessel Materials

Predictions for changes in transition temperature and upper shelf energy are discussed in <Section 5.3.1.6.3> and <Section 5.3.2.1.5>.

5.3.1.5 Fracture Toughness

5.3.1.5.1 Compliance with 10 CFR 50, Appendix G

<10 CFR 50, Appendix G> is interpreted for Class I RCPB components of the BWR 6 reactor design and complied with as discussed in <Section 5.3.2> and below with the following exceptions:

a. The specific temperature limits for operation when the core is critical are based on <10 CFR 50, Appendix G>, and are given in GE Licensing Topical Report NEDO-21778-A (Reference 1).

b. A minimum boltup and pressurization temperature of 70°F is called for, which is at least 60°F above the flange region RT_{NDT}. This exceeds the minimum RT_{NDT} temperature required by ASME Code Section III, Appendix G, Paragraph 2222(c), Summer 1976 and later editions. A flange region flaw size less than 10 percent of the wall thickness can be detected at the outside surface of the flange to shell and head junctions where stresses due to boltup are most limiting.

The following Items a through g are the interpretations and methods used to comply with <10 CFR 50, Appendix G>. Item h reports the fracture toughness test results and the background information used as the basis to show compliance with <10 CFR 50, Appendix G>.

a. Records and Procedures for Impact Testing

   Personnel conducting fracture toughness testing were qualified by experience and training that demonstrated competency to perform tests in accordance with required procedures. No record of qualification of individuals performing these tests were required at that time as the order of the Perry components predates the requirements of <10 CFR 50, Appendix G>.

b. Specimen Orientation for Original Qualification Versus Surveillance

The special beltline longitudinally oriented Charpy specimens required by the general reference NB-2300 and, specifically, NB-2322.2(a)(6) are not included in the surveillance program base metal. Instead, the orientation of the Charpy specimens is in accordance with Figure 1 of ASTM E-185-73, as described below.

c. Charpy-V Curves for the RPV Beltline

It is understood that the orientation of impact test specimens shall comply with the requirements of NB-2322(a)4 (transverse specimen) for plate material as opposed to NB-2322(a)(6) (longitudinal specimen). This understanding of the general reference to NB-2322 in G-III C results in meaningful and conservative beltline curves of unirradiated materials for comparison with the results of surveillance program testing of irradiated transverse base metal specimens and also allow this curve to comply with ASTM E-185-73.

The procedures of ASTM E-185-73 were used for selection of surveillance specimen base material to provide a conservative adjusted reference temperature for the beltline base material. The test plate weld materials are equivalent to beltline construction weld materials. The weld test plate for the surveillance program specimens had the principal working direction normal to the weld seam to assure that heat affected zone specimens are oriented such that they parallel actual production weld conditions.

d. Upper Shelf Energy for Beltline

All beltline material meets the Charpy-V-Notch test minimum upper shelf energy of 75 ft-lbs for Perry reactor pressure vessels.

e. Bolting Materials

See <Section 5.3.1.7>.

f. Alternative Procedures for the Calculation of Stress Intensity Factor

Stress Intensity Factors were calculated by the methods of Appendix G to Section III of the ASME Code. Discontinuity regions were evaluated, as well as shell and head areas, as part of the detailed thermal and stress analysis in the vessel stress report. Equivalent margins of safety to those required for shells and heads were demonstrated using a 1/4 t defect at all locations, with the exception of the main closure flange to head and shell discontinuity locations. Here it was found that additional restriction on operating limits would be required for outside surface flaw size greater than 0.24 inches at the outside surface of the flange to shell joint (based on additional analyses made for BWR 6 reactor vessels). It has been demonstrated using a test mockup of these areas that smaller defects can be detected by the ultrasonic inservice examinations procedures required at the adjacent weld joint. Since the stress intensity factor is greatest at the outside surface of the flange to shell and head joints a flaw can also be detected by outside surface examination techniques.

g. Fracture Toughness Margins in the Control of Reactivity

Appendix G of the ASME Code, Section III (1971 Edition with Addenda to and including Winter 1972 or later), "Protection Against Non-ductile Failure," was used in determining pressure/temperature limitations for all phases of plant operation. Additionally, when the core is critical a 40°F temperature allowance is included in the reactor vessel operating pressure vs. temperature limits to account for operational occurrences in the control of reactivity as described in GE BWR Licensing Topical Report NEDO-21778-A and the Nuclear Regulatory Commission's acceptance basis which is included therein.

h. Results of fracture toughness tests are reported in <Table 5.3-1> and <Table 5.3-2>.

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.

Reactor vessel materials surveillance specimens are provided in accordance with requirements of ASTM E-185-73 and <10 CFR 50, Appendix H>. Materials for the program are selected to represent materials used in the reactor beltline region. Specimens are manufactured from a plate actually used in the beltline region and a weld typical of those in the beltline region and thus represent base metal, weld material, and the weld heat affected zone material. The plate and weld are heat treated in a manner which simulates the actual heat treatment performed on the core region shell plates of the completed vessel.

Each in-reactor surveillance capsule contains Charpy-V-Notch specimens with base metal, weld metal and heat affected zone material as shown in the following tables. A set of out-of-reactor baseline Charpy-V-Notch specimens and archive material are provided with the surveillance test specimens.

	Number of Charpy V-Notch Specimens		
Capsule Location	Base	Weld	HAZ
3°[1]	12	12	12
177°[3]	12	12	12
183°	12	12	12
3° Reconstituted[2]	12	12	N/A
177° Unit 2 Spare[4]	See Note[4]	See Note[4]	See Note[4]

<u>NOTES</u>:

[1] Removed in RF05 at 5.5 EFPY (effective full power years)

[2] Installed in RF06

[3] Removed in RFO14 in accordance with BWRVIP-86-A

[4] Unit 2 spare capsule installed in RFO14 for Noble Metal Deposition Sampling purposes only. Capsule shall not be used for ISP purposes.

Three capsules are provided in accordance with Case A requirements of <10 CFR 50, Appendix H> since the predicted (at time of design) increase in reference temperature of the reactor vessel steel was less than 100°F at end of life.

The program for implementation of the scheduling, withdrawal, and testing of the material surveillance specimens is governed and controlled by the Boiling Water Reactor Vessel and Internals Project (BWRVIP) BWRVIP-86 (Reference 9). The BWRVIP Integrated Surveillance Program (ISP) complies with the requirements of 10 CFR 50, Appendix H.

The specimens will be pulled in accordance with the test matrix included in BWRVIP-86 as modified by the NRC's safety evaluation.

Capsule	ISP Capsule ID		Date
First Capsule	PY1	3°	Withdrawn 1/96 after 5.5 Effective Full-Power Years
Second Capsule	PY1	177°	Withdrawn 1R14 (2013) after 20.0 Effective Full-Power Years
Third Capsule	PY1	183°	As specified in Reference 9
Reconstituted Capsule	PY1	3°	Standby

5.3.1.6.2 Neutron Flux and Fluence Calculations

A neutron fluence calculation methodology which has been approved by the NRC staff and conforms with U.S. Nuclear Regulatory Commission Regulatory Guide 1.190, “Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence”, will be used for the determination of neutron fluence values for the PNPP.

5.3.1.6.3 Predicted Irradiation Effects on Vessel Beltline Materials

Estimated maximum changes in nil ductility temperature (RT_{NDT}) and upper shelf fracture energy at end of plant life are listed for selected reactor vessel materials in <Table 5.3-3>. Reference nil ductility temperatures were established in accordance with <10 CFR 50, Appendix G> and NB-2330 of the ASME Code.

5.3.1.6.4 Positioning of Surveillance Capsules and Methods of Attachment (Refer to 10 CFR 50, Appendix H)

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 5.3-1>. The capsule holder brackets allow the removal and reinsertion of capsule holders. These brackets are designed, fabricated and analyzed to the requirements of ASME Code Section III.

In areas where brackets, such as the surveillance specimen holder brackets, are located, additional non-destructive examinations are performed on the vessel base metal and stainless steel weld deposited cladding or weld buildup pads during vessel manufacture. The base metal is ultrasonically examined by straight beam techniques to a depth at least equal to the thickness of the bracket being joined. The area examined is the area of the subsequent attachment weld plus a band around this area of width equal to at least half the thickness of the part joined. The required stainless steel weld deposited cladding is similarly examined. The full penetration welds are liquid penetrant examined to ASME Code Section III Standards. Cladding thickness is required to be at least 1/8 inch.

The above requirements have been successfully applied to a variety of bracket designs which are attached to weld deposited stainless steel cladding or weld buildups in many operating BWR reactor pressure vessels.

Inservice inspection examinations of core beltline pressure retaining welds are performed from the outside surface of the reactor pressure vessel. If a bracket for mechanically retaining surveillance specimen capsule holders were located at or adjacent to a vessel shell weld, it would not interfere with the straight beam or half node angle beam inservice inspection ultrasonic examinations performed from the outside surface of the vessel.

5.3.1.6.5 Time and Number of Dosimetry Measurements

GE provided a separate neutron dosimeter so that fluence measurements may be made at the vessel ID after the first fuel cycle to verify the predicted fluence at an early date in plant operation. In addition, each surveillance capsule contains iron and copper flux wires. When the first capsule is removed, these wires can be used to determine the relationship between reactor power and neutron fluence. The Unit 1 vessel measurement was made in 1989 with a measured flux somewhat lower than the calculated design value. The Unit 1 measurements were repeated when the 3° azimuth capsule was withdrawn in 1996 after 5.5 effective full power years. The results obtained were slightly higher than those obtained in 1989 after 1.09 effective full power years, but still within calculated design values.

5.3.1.7 Reactor Vessel Fasteners

The reactor vessel closure head (flange) is fastened to the reactor vessel shell flange by multiple sets of threaded studs and nuts. The lower end of each stud is installed in a threaded hole in the 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. Hardness tests are performed on all main closure bolting to demonstrate that heat treatment has been properly performed.

<Regulatory Guide 1.65> defines acceptable materials and testing procedures with regard to reactor vessel closure stud bolting for light-water-cooled reactors. The vessel order date preceded implementation of <Regulatory Guide 1.65>. The design and analysis of reactor vessel bolting materials is in full compliance with ASME Code Section III, Class I requirements. The reactor pressure vessel closure studs are SA-540, Grade B 23 or 24 (AISI4340). The maximum reported ultimate tensile strength is 174,000 psi. Also, the Charpy impact test requirements of <10 CFR 50, Appendix G>, were satisfied, since the lowest reported Charpy-V-Notch energy was 44 ft-lbs at $+10^{\circ}F$, compared to the requirement of 45 ft-lbs at the lowest service temperature, and the lowest reported Charpy-V-Notch expansion was 25 mils at $+10^{\circ}F$ compared to the 25 mils required.

There are no metal platings applied to closure studs, nuts or washers. A phosphate coating is applied to threaded areas of studs and nuts and bearing areas of nuts and washers to act as a rust inhibitor and to assist in retaining lubricant (either graphite/alcohol or nickel powder base lubricant) on these surfaces.

In relationship to <Regulatory Guide 1.65>, Position C.2.b., the bolting materials were ultrasonically examined in accordance with ASME Code Section III, NB-2585 after final heat treatment and prior to threading. The specified requirement for examination according to SA-388 was complied with. The procedures approved for use in practice were judged to insure comparable material quality and, moreover, were considered adequate on the basis of compliance with the applicable requirements of ASME Code Paragraph NB-2583. Straight beam examination was performed on 100 percent of cylindrical surfaces, and from both ends of each stud using a 3/4-inch maximum diameter transducer. In addition to the code required notch, the reference standard for the radial scan contains a 1/2 inch diameter flat bottom hole with a depth of 10 percent of thickness, and the end scan standard contains a 1/4-inch diameter flat bottom hole 1/2-inch deep. Also, angle beam examination was performed on the outer cylindrical surface in both axial and circumferential directions. Any indication greater than the indication from the applicable calibration feature is unacceptable. A distance-amplitude correction curve per NB-2585 is used for the longitudinal wave examination. Surface examinations were performed on the studs and nuts after final heat treatment and threading, as specified in the Guide, in accordance with NB-2583 of the applicable ASME code.

In relationship to <Regulatory Guide 1.65>, Position C.3, GE practice allows exposure to stud bolting surfaces to high purity fill water; nuts and washers are dry stored during refueling.

5.3.2 PRESSURE-TEMPERATURE LIMITS

5.3.2.1 Limit Curves

The limit curves (Reference 14) are based on the requirements of <10 CFR 50, Appendix G>; <ASME Code Appendix G>; and (Reference 4), (Reference 5), (Reference 7), and (Reference 15).

All the vessel shell and head areas remote from discontinuities plus the feedwater nozzles were evaluated, and the operating limit curves are based on the limiting location. The boltup limits for the flange and adjacent shell region are based on a minimum metal temperature of RT_{NDT} + 60°F. The maximum through-wall temperature gradient from continuous heating or cooling at 100°F per hour was considered. The Unit 1 limit curves are provided for up to 32 EFPY (Reference 14). The curves include the stress concentration of the water level instrument nozzle (Reference 15).

5.3.2.1.1 Temperature Limits for Boltup

A minimum temperature of 70°F is required for the closure studs. A sufficient number of studs may be tensioned at 70°F to seal the closure flange O-rings for the purpose of raising the 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 temperature of 70°F before they are stressed by the full intended bolt preload. The fully preloaded boltup limits are shown on the limit curves (Reference 14).

5.3.2.1.2 Temperature Limits for Preoperational System Hydrostatic Tests and ISI Hydrostatic or Leak Pressure Tests

Based on <10 CFR 50, Appendix G>, with no fuel in the reactor during system preoperational hydrostatic pressure tests, the minimum permissible test temperature is 100°F at 1,563 psig.

The fracture toughness analysis for ISI system pressure tests resulted in pressure test curves (Reference 14). The curve labeled "upper vessel and beltline limits” is based on an initial RT_{NDT} of -30°F for the beltline weld material and -20°F for the upper vessel material.

The predicted shift in the RT_{NDT} is based on vessel ID neutron fluence attenuated to the 1/4 T depth according to <Regulatory Guide 1.99>, Revision 2 and have been added to the beltline curve to account for the effect of fast neutrons.

5.3.2.1.3 Operating Limits During Heatup, Cooldown and Core Operation

The fracture toughness analysis was done for the normal heatup or cooldown rate of 100°F/hour. The temperature gradients and thermal stress effects corresponding to this rate were included. The results of the analyses are a set of operating limits for non-nuclear heatup or cooldown shown as Core Critical Operation and Non-Nuclear Heatup and Cooldown curves (Reference 14).

5.3.2.1.4 Reactor Vessel Annealing

In place annealing of the reactor vessel because of radiation embrittlement is unnecessary because the predicted value in transition of adjusted reference temperature does not exceed 200°F.

5.3.2.1.5 Predicted Shift in RT_{NDT}

The adjusted reference temperatures for the most limiting beltline materials are based on <Regulatory Guide 1.99>, Revision 2.

5.3.2.2 Operating Procedures

By comparison of the pressure vs. temperature limit in <Section 5.3.2.1> with intended normal operating procedures for the most severe upset transient, it is shown that these limits will not be exceeded during any foreseeable upset condition. Reactor operating procedures have been established such that actual transients will not be more severe than those for which the vessel design adequacy has been demonstrated. Of the design transients, the upset condition producing the most adverse temperature and pressure condition anywhere in the vessel head and/or shell areas occurs in the bottom head, yielding a minimum fluid temperature of 250°F and a maximum pressure peak of 1,180 psig. Scram automatically occurs as a result of this event, prior to the reduction in bottom head fluid temperature, so the applicable operating limits are given by the non-nuclear heating limit Curve (Reference 14). For a temperature of 250°F, the maximum allowable pressure exceeds 1,400 psig for the intended margin against non-ductile failure. The maximum transient pressure of 1,180 psig is therefore within the specified allowable limits.

5.3.3 REACTOR VESSEL INTEGRITY

The reactor vessel was fabricated for General Electric by CBI Nuclear Co., and was subject to the requirements of General Electric's quality assurance program.

The CBI Nuclear Co. has had extensive experience with GE reactor vessels and has been the primary supplier of GE domestic reactor vessels and some foreign vessels since the company was formed in 1972 from a merger. Prior experience by the Chicago Bridge and Iron Company with an agreement between Chicago Bridge and Iron Co. and General Electric GE reactor vessels dates back to 1966.

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

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

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

An analysis of the structural integrity of boiling water reactor pressure vessels during a design basis accident (DBA) has been performed.

The analysis included:

a. Description of the LOCA event.

b. Thermal analysis of the vessel wall to determine the temperature distribution at different times during the LOCA.

c. Determination of the stresses in the vessel wall including thermal, pressure and residual stresses.

d. Consideration of radiation effect on material toughness (NDTT shift and changes in toughness).

e. Fracture mechanics evaluation of vessel wall for different postulated flaw sizes.

This analysis incorporated conservative assumptions in all areas (particularly in the areas of heat transfer, stress analysis, effects of radiation on material toughness, and crack tip stress intensity factor evaluation). The analysis concluded that even in the presence of large flaws, the vessel will have considerable margin against brittle fracture following a loss-of-coolant accident.

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 5.3-6> is a vertical, cylindrical pressure vessel of welded construction. The vessel is designed, fabricated, tested, inspected, and stamped in accordance with the ASME Code Section III, Class I requirements including the addenda in effect at the date of order placement, Winter 1972. Design of the reactor vessel and its support system satisfies Seismic Category I equipment requirements. The materials used in the reactor pressure vessel are shown in <Table 5.2-5>.

The cylindrical shell and top and bottom heads of the reactor vessel are fabricated of low alloy steel, the interior of which is clad with stainless steel weld overlay, except for the top head and top head nozzle and nozzle weld zones.

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 and the predicted value of adjusted reference temperature does not exceed 200°F. Radiation embrittlement is not a problem outside of the vessel beltline region because of the low fluence in those areas.

Quality control methods used during the fabrication and assembly of the reactor vessel and appurtenances assure 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 and to a cylinder supported by vertical stilt legs from the bottom head. This support is designed to carry the weight of 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 boiling water reactor does not use borated water for reactivity control during normal operation.

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 appurtenances 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 have been 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 assure that 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 1,250 psig and the design temperature is 575°F. The maximum installed test pressure is 1,563 psig.

5.3.3.1.4.1 Vessel Support

5.3.3.1.4.2 Control Rod Drive Housings

The control rod drive housings are inserted through the control rod drive penetrations in the reactor vessel bottom head and are welded to the reactor vessel. Each housing transmits loads to the bottom head of the reactor. These loads include the weights of a control rod, a control rod drive, a control rod guide tube, a four-lobed fuel support piece, and the four fuel assemblies that rest on the fuel support piece. The material used to fabricate the housings is described in <Section 4.5.2.1>, item f.

5.3.3.1.4.3 In-Core Neutron Flux Monitor Housings

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

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

5.3.3.1.4.4 Reactor Vessel Insulation

The reactor vessel insulation is of the reflective type and is constructed completely of metal. The outer surface temperature of the insulation is expected to be at 150°F and the heat transfer rate through the insulation is approximately 65 Btu/hr-ft^2 under normal operating conditions. The insulation consists of several self-contained assemblies latched together, each of which can be easily removed and replaced. The insulation assemblies are designed to remain in place and resist permanent damage during a safe shutdown earthquake.

The reactor top head insulation is supported from a structure on the bulkhead. During refueling, the support structure along with the top head insulation is removed. The insulation for the reactor vessel cylindrical surface is supported by brackets welded on the shield wall liner plate.

5.3.3.1.4.5 Reactor Vessel Nozzles

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

The vessel top head nozzles are provided with flanges having small groove facing. The drain nozzle is of the full penetration weld design. The recirculation inlet nozzles (located as shown in <Figure 5.3-6>), feedwater inlet nozzles, core spray inlet nozzles and LPCI nozzles, all have thermal sleeves. Nozzles connecting to stainless steel piping have safe ends, or extensions made of stainless steel. These safe ends or extensions were welded to the nozzles after the pressure vessel was heat treated to avoid furnace sensitization of the stainless steel. The material used is compatible with the material of the mating pipe.

The solution of the feedwater nozzle cracking problems involve several elements including nozzle clad removal and thermal sleeve redesign. A description of these changes and appropriate analysis is available in (Reference 2).

In order to mitigate IGSCC in reactor vessel nozzle to safe-end welds that contain an IGSCC susceptible material (Inconel 182 weld metal buttering or welds in Type 304 material which were not solution annealed), a stress improvement process has been performed on these welds. The Mechanical Stress Improvement Process (MSIP) discussed in <NUREG-0313>, Revision 2 and <Generic Letter 88-01> was the process utilized. MSIP was performed on the RPV nozzle to safe-end connections for the following systems: reactor recirculation, feedwater, low pressure core spray, high pressure core spray, residual heat removal, jet pump instrumentation. It was also performed on the jet pump instrumentation nozzle safe-end to penetration seal connections.

During RF07, a weld overlay was applied to the feedwater nozzle to safe-end weld 1B13-N4C-KB. The overlay is designed as a full structural overlay in accordance with the recommendations of <NUREG-0313>, Revision 2 (forwarded by <Generic Letter 88-01>), ASME Code Case N-504, and Section XI of the ASME Boiler and Pressure Vessel Code, 1989 Edition (Paragraph IWB-3640).

5.3.3.1.4.6 Materials and Inspections

The reactor vessel was designed and fabricated in accordance with the appropriate ASME Boiler and Pressure Vessel Code as defined in <Section 5.2.1>. <Table 5.2-5> 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 5.3-6>. Trip system water levels are indicated as shown in <Figure 5.3-7>.

5.3.3.2 Materials of Construction

All materials used in the construction of the reactor pressure vessel conform to the requirements of ASME Code Section II materials. The vessel heads, shells, flanges, and nozzles are fabricated from low alloy steel plate and forgings purchased in accordance with ASME Specifications SA533 Grade B Class 1 or 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 of construction were selected because they provide adequate strength, fracture toughness, fabricability, and compatibility with the BWR environment. Their suitability has been demonstrated by long term successful operating experience in reactor service.

5.3.3.3 Fabrication Methods

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

Submerged arc and manual stick electrode welding processes were employed. Electroslag welding was not permitted. Preheat and interpass temperatures employed for welding of low alloy steel satisfied or exceeded the requirements of ASME Code Section III, Subsection NA. Post weld heat treatment of 1,100°F minimum was applied to all low alloy steel welds.

All previous BWR pressure vessels have employed similar fabrication methods. These vessels have operated for an extensive number of years and their service history is excellent.

The vessel fabricator, CBI Nuclear Co., has had extensive experience with General Electric Co. reactor vessels and has been the primary supplier for General Electric domestic reactor vessels and some foreign vessels since the company was formed in 1972 from a merger agreement between Chicago Bridge and Iron Co. and General Electric Co. Prior experience by the Chicago Bridge and Iron Co. with General Electric Co. reactor vessels dates back to 1966.

5.3.3.4 Inspection Requirements

All plate, forgings and bolting were 100 percent ultrasonically tested and surface examined by magnetic particle methods or liquid penetrant methods in accordance with ASME Code Section III requirements. Welds on the reactor pressure vessel were examined in accordance with methods prescribed and satisfy the acceptance requirements specified by ASME

Code Section III. In addition, the pressure retaining welds were ultrasonically examined using acceptance standards which are required by ASME Code Section XI.

5.3.3.5 Shipment and Installation

The completed reactor vessel was given a thorough cleaning and examination prior to shipment. The vessel was tightly sealed for shipment to prevent entry of dirt or moisture. Preparations for shipment are in accordance with detailed written procedures. On arrival at the reactor site the reactor vessel was carefully examined for evidence of any contamination as a result of damage to shipping covers. Suitable measures were taken during installation to assure that vessel integrity was maintained; for example, access controls were applied to personnel entering the vessel, weather protection is provided and periodic cleanings are performed.

5.3.3.6 Operating Conditions

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

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

b. If the coolant temperature difference between the dome (inferred from P(sat)) and the bottom head drain exceeds 100°F, neither reactor power level 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 the saturated water temperature corresponding to the steam dome pressure.

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

The above operational limits, when maintained, ensure that the stress limits within the reactor vessel and its components are within the thermal limits to which the vessel was designed for normal operating conditions. To maintain the integrity of the vessel in the event that these operational limits are exceeded the reactor vessel has also been designed to withstand a limited number of transients caused by operator error. Also, for abnormal operating conditions where safety systems or controls provide an automatic temperature and pressure response in the reactor vessel, the reactor vessel integrity is maintained since the severest anticipated transients have been included in the design conditions. Therefore, it is concluded that the vessel integrity will be maintained during the most severe postulated transients, since all such transients are evaluated in the design of the reactor vessel. The postulated transient for which the vessel has been designed is shown on <Figure 5.2-3> and discussed in <Section 5.2.2>.

5.3.4 REFERENCES FOR SECTION 5.3

1. Cooke, F., "Transient Pressure Rises Affecting Fracture Toughness Requirements for BWR," NEDO-21778-A, dated December 1978.

2. Watanabe, H., "Boiling Water Reactor Feedwater Nozzle/Sparger, Final Report," (NEDE-21821-2 and NEDO-21821-2), dated August 1979.

3. Caine, T. A., "Implementation of <Regulatory Guide 1.99>, Revision 2 for Perry Nuclear Power Plant Unit 1," November 1989 (SASR 89-76/DRF 137-0100).

4. "Radiation Effects in Boiling Water Reactor Pressure Vessel Steels," (NEDO-21708), dated October 1977.

5. Tilly, L. J., "Perry Unit 1 RPV Surveillance Materials Testing and Analysis," November 1996 (GE-NE-B1301793-01, Revision 0)

6. American Society for Testing and Materials (ASTM) E 185-73, Standard Recommended Practice for Surveillance Tests for Nuclear Rector Vessels, 1973.

7. O'Connor, M. C., "Pressure-Temperature Curves for FirstEnergy Corporation, Using the KI_{c} Methodology, Perry Unit 1," April 2002 (GE-NE-0000-0000-8763, Revision 0).

8. Superseded by BWRVIP-86-A, see (Reference 9).

9. Boiling Water Reactor Vessel and Internals Project (BWRVIP) BWRVIP-86, "BWR Vessel and Internals Project, Updated BWR Integrated Surveillance Program (ISP) Implementation Plan", Current Effective Version published by EPRI.

10. Superseded by BWRVIP-86-A, see (Reference 9).

11. Superseded by BWRVIP-86-A, see (Reference 9).

12. Superseded by BWRVIP-86-A, see (Reference 9).

13. Boiling Water Reactor Vessel and Internals Project (BWRVIP) BWRVIP-135, "Integrated Surveillance Program (ISP) Data Source Book and Plant Evaluations", Current Effective Revision published by EPRI.

14. Perry Technical Specification, Section 3.4.11, Figure 3.4.11-1(a) Pressure Test Curves, Figure 3.4.11-1(b) Non-Nuclear Heatup/Cooldown Curves, Figure 3.4.11-1(c) Core Critical Operation Curves.

15. Topical Report (TR) BWROG-TP-11-023A, Revision 0, "Linear Elastic Fracture Mechanics Evaluation of General Electric Boiling Water Reactor Water Level Instrument Nozzles for Pressure - Temperature Curve Evaluations May 2013.

TABLE 5.3-1

CHARPY TEST RESULTS AND CHEMICAL COMPOSITION

UNIT 1

I. VESSEL BELTLINE MATERIAL IDENTIFICATION

A. Number 2 Shell Ring

Plates - Pc. 22-1-1, Heat C2557, Slab 1
Pc. 22-1-2, Heat B6270, Slab 1
Pc. 22-1-3, Heat A1155, Slab 1

B. Welds in No. 2 Shell Ring Vertical Seams

Seam BD - Type E8018NM, Heat 627260, Lot B322A27AE
Type E8018NM, Heat 626677, Lot C301A27AF
Type RACO-1NMM, Heat 5P6214B, Lot 0331

Seam BE - Type E8018NM, Heat 624063, Lot D228A27A
Type E8018NM, Heat 626677, Lot C301A27AF
Type E8018NM, Heat 627069, Lot C312A27A
Type RACO-1NMM, Heat 5P6214B, Lot 0331

Seam BF - Type E8018NM, Heat 627260, Lot B322A27AE
Type E8018NM, Heat 626677, Lot C301A27AF
Type RACO-1NMM, Heat 5P6214B, Lot 0331

II. CHEMICAL ANALYSES FOR BELTLINE MATERIAL

A. Plates	C	Mn	P	S	Cu	Si	Ni	Mo	V	A1
Pc. 22-1-1, Ht C2557	.23	1.32	.010	.025	.06	.27	.61	.54	.001	.039
Pc. 22-1-2, Ht B6270	.20	1.28	.012	.015	.06	.23	.63	.53	0	.039
Pc. 22-1-3, Ht A1155	.20	1.33	.010	.013	.06	.28	.63	.54	.002	.031

B. Welds	C	Mn	Ni	Si	Mo	Cu	P	S	V
Ht. 627260 Lot B322A27AE	.04	1.25	1.08	.56	.64	.06	.020	.022	.02
Ht. 626677 Lot C301A27AF	.048	1.10	.85	.45	.45	.010	.015	.022	.009
Ht. 5P6214B Lot 0331	.051	1.39	.82	.53	.52	.02	.013	.017	.004
Ht. 624063 Lot D228A27A	.041	1.12	1.00	.41	.54	.03	.009	.018	.01
Ht. 627069 Lot C312A27A	.037	1.07	.94	.60	.52	.010	.013	.019	.012

TABLE 5.3-2

UNIRRADIATED FRACTURE TOUGHNESS PROPERTIES

UNIT 1

Plates Ht.No.	Drop Wt. NDT(°F)	Transverse CVN ft-lbs	MLE	Temp (°F)	Reference Temp.(°F)	Upper Shelf (ft-lb)
C2557-1						
Top	-20	52,50,52	42,46,42	+70	+10	84
Bottom	-20	54,64,76	63,53,46	+60		
B6270-1						
Top	-40	53,78,56	43,58,44	+20		
Bottom	-30	63,63,64	51,51,52	+30	-30	94
A1155-1						
Top	-20	65,63,67	54,60,52	+50	-10	114
Bottom	-20	54,66,85	68,55,44	+40		

Plates Metal	Drop Wt. NDT(°F)	ft-lbs	CVN MLE	Temp (°F)	Reference Temp.(°F)	Upper Shelf (ft-lb)
Ht. 627260 Lot B322A27AE	-40	52,56,51	36,37,35	+30	-30	104
Ht. 626677 Lot C301A27AF	-40	53,51,54	36,37,35	+40	-20	90
Ht. 5P6214B	-50	56,50,54	45,41,46	+10	-50	88
Lot 0331	-40	50,61,64	46,50,52	+10	-40	96
Ht. 624063 Lot D228A27A	-60	57,59,68	37,38,46	+10	-50	105
Ht. 627069 Lot C312A27A	-60	72,64,78	52,48,56	0	-60	112

TABLE 5.3-3

EOL BELTLINE PLATE RT_{NDT} AND WELDS
1/4 T 32 EFPY FLUENCE = 2.9 x 10^{18} n/cm^2
based upon measurements taken after 5.5 EFPY from the 3° azimuth capsule and fluence calculated in accordance with <Regulatory Guide 1.190> (2002)

Shell #2

Thickness in inches = 6.00 Ratio Peak/Location = 1.00
32 EFPY Peak I.D. fluence = 4.1E+18 n/cm^2
32 EFPY Peak 1/4 T fluence = 2.9E+18 n/cm^2
32 EFPY Peak 1/4 T fluence = 2.9E+18 n/cm^2

Shell #2 Vertical Welds

Thickness in inches = 6.00 Ratio Peak/Location = 1.00
32 EFPY Peak I.D. fluence = 4.1E+18 n/cm^2
32 EFPY Peak 1/4 T fluence = 2.9E+18 n/cm^2
32 EFPY Peak 1/4 T fluence = 2.9E+18 n/cm^2

COMPONENT	HEAT OR HEAT/LOT	% Cu	% Ni	CF	Initial RT_{NDT} °F	1/4 T Fluence n/cm^2	32 EFPY Δ RT_{NDT} °F	σ_1	σ_{Δ}	Margin °F	32 EFPY Shift °F	32 EFPY ART °F
PLATES:												
Shelf #2:												
Mk 22-1-1	C2557-1	0.060	0.61	37	10	2.9E+18	24	0	12	24	49	59
Mk 22-1-2	B6270-1	0.060	0.63	37	-30	2.9E+18	24	0	12	24	49	19
Mk 22-1-3	A1155-1	0.060	0.63	37	-10	2.9E+18	24	0	12	24	49	39
	C2557-1[a]	0.054	0.62	33	10	2.9E+18	22	0	11	22	43	53
WELDS:												
Vertical Welds:												
Seam BD,BE,BF	5P6214B	0.020	0.82	27	-40	2.9E+18	18	0	9	18	36	-4
Seam BD,BF	627260	0.060	1.08	82	-30	2.9E+18	54	0	27	54	108	78
Seam BD,BE,BF	626677	0.010	0.85	20	-20	2.9E+18	13	0	7	13	26	6
Seam BE	624063	0.030	1.00	41	-50	2.9E+18	27	0	13	27	54	4
Seam BE	627069	0.010	0.94	20	-60	2.9E+18	13	0	7	13	26	-34
	5P6214B[b]	0.025	0.91	34	-40	2.9E+18	22	0	11	22	45	5
	5P6214B[c]	(d)	(d)	39[d]	-40	2.9E+18	26	0	13	26	52	12

NOTES:

[a] Surveillance Plate (Best Estimate Chemistry)
[b] Surveillance Weld (Best Estimate Chemistry)
[c] Integrated Surveillance Program (Best Estimate Chemistry)
[d] Chemical composition is based on multiple test specimens, and results in the adjusted Chemistry Factor (CF)

5.4 COMPONENT AND SUBSYSTEM DESIGN

5.4.1 REACTOR RECIRCULATION PUMPS

5.4.1.1 Safety Design Bases

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

a. An adequate fuel barrier thermal margin is assured 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. 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 parallel 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 <Figure 5.4-2>. Each external loop contains one high capacity, motor driven recirculation pump, a flow control valve, and two motor operated gate valves (for pump maintenance). Each pump suction line contains a flow measuring system. The recirculation loops are part of the reactor coolant pressure boundary and are located inside the drywell structure. The jet pumps are reactor vessel internals. Their location and mechanical design are discussed in <Section 3.9>. However, certain operational characteristics of the jet pumps are also discussed in this section. A tabulation of the important design and performance characteristics of the reactor recirculation system is shown in <Table 5.4-1>. Typical head, NPSH, flow, and efficiency curves are shown in <Figure 5.4-3>, while the typical flow control valve characteristic is shown in <Figure 5.4-4>. Instrumentation and control description is provided in <Section 7.1>.

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 a ring header from which individual recirculation inlet lines are routed to the jet pump risers within the reactor vessel. The remaining portion of the coolant mixture in the annulus becomes the driven flow for the jet pumps. This flow enters the jet pump at suction inlets and is accelerated by the driving flow. The flows, both driving and driven, are mixed in the jet pump throat section and result in partial pressure recovery. The balance of recovery is obtained in the jet pump diffuser <Figure 5.4-5>. 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 that for the reactor vessel. If one loop is shut down, the idle loop can be kept hot by leaving the loop valves open; this permits the active jet pump head to cause reverse flow in the idle loop. Justification for single loop operation is provided in <Appendix 15F>.

Because the removal of the reactor recirculation gate valve internals may require unloading the core due to the resulting draining of 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 backseats that permit renewal of stem packing while the system is full of water.

When the pump is operating at 25 percent speed, the head provided by the elevation of the reactor water level above the recirculation pump is sufficient to provide the required NPSH for the recirculation pumps, flow control valve, and jet pumps. When the pump is operating at 100 percent speed, most of the NPSH is supplied by the subcooling provided by the feedwater flow. Temperature detectors are provided in the recirculation lines and the steam dome. The difference between these two readings is a direct measurement of the subcooling. If the subcooling falls below approximately 8°F, the 100 percent speed power supply is tripped to the 25 percent speed power source to prevent cavitation of recirculation pump, jet pumps, and/or the flow control valve. The capability exists to bypass the cavitation interlock above the 70% Rod Line.

When preparing for hydrostatic tests, the nuclear system temperature must be raised above the vessel nil ductility transition temperature limit. The vessel is heated by core decay heat and/or by operating the recirculation pumps at 100 percent speed.

Each recirculation pump is driven by a constant speed motor and is equipped with mechanical shaft seal assemblies. The two seals built into a cartridge can be readily replaced without removing the motor from the pump. Each individual seal in the cartridge is designed for pump operating pressure so that any one seal can adequately limit leakage in the event that the other seal should fail. The flow passages along the pump shaft are designed to limit leakage in the event of a gross failure of both shaft seals. The cavity temperature and pressure drop across each individual seal is monitored.

Each recirculation pump motor is a constant speed, vertical, solid shaft, totally enclosed, air-water cooled induction motor. The combined rotating inertias of the recirculation pump and motor provide a slow coastdown of flow following loss of power to the drive motors so that the core is adequately cooled during the transient. This inertia requirement is met without a flywheel.

The pump discharge flow control valve can throttle the discharge flow of the pump proportionally to an instrument signal. The flow control valve is provided with an equal percentage characteristic. The recirculation loop flow rate can be rapidly changed, within the expected flow range, in response to rapid changes in system demand.

The design objective for the recirculation system equipment is to provide units that will not require removal from the system for rework or overhaul. Pump casing and valve bodies are designed for a 40 year life and are welded to the pipe.

The pump drive motor, impeller, wear rings and flow control valve internals are designed for as long a life as is practical. Pump mechanical seal parts and the valve packing are expected to have a life expectance which affords convenient replacement during the refueling outages.

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

The reactor recirculation system pressure boundary equipment is designed as Seismic Category I equipment. As such, it is designed to resist sufficiently the response motion for the safe shutdown earthquake at the installed location within the supporting structure. The pump is assumed to be filled with water for the analysis. Snubbers located at the top of the motor and at the bottom of the pump casing are designed to resist the 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 drywell integrity. This restraint system provides adequate clearance for normal thermal expansion movement of the loop. The criteria for the protection against the dynamic effects associated with a postulated pipe rupture are contained in <Section 3.6>.

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

The insulation is of the fiberglass blanket type as described in <Section 6.1.2>. It is prefabricated into components for field installation. Removable insulation is provided at various locations to permit periodic inspection of the equipment.

5.4.1.4 Safety Evaluation

Reactor recirculation system malfunctions that pose threats of damage to the fuel barrier are described and evaluated in <Chapter 15>. It is shown in <Chapter 15> that none of the malfunctions result in significant fuel damage. The recirculation system has sufficient flow coastdown characteristics to maintain fuel thermal margins during abnormal operational transients. <Appendix 15F> provides justification that PNPP can safely operate with a single recirculation loop up to 2500 Megawatts-Thermal power.

The core flooding capability of a jet pump design plant is discussed in detail in the emergency core cooling systems document filed with the NRC as a General Electric topical report (Reference 1). The ability to reflood the BWR core to the top of the jet pumps is shown schematically in <Figure 5.4-6> and is discussed in (Reference 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 parts are chosen in accordance with applicable codes. Use of the listed code design criteria assures that a system designed, built and operated within design limits has an extremely low probability of failure caused by any known failure mechanism.

General Electric Purchase Specifications require that the first critical speed of the recirculation pump not be less than 130 percent of operating speed. Calculation submittal was verified by General Electric Design Engineering.

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

Pump overspeed occurs during the course of a LOCA due to blowdown through the broken loop pump. Design studies determined that the overspeed was not sufficient to cause destruction of the motor; 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 are used during fabrication and assembly of the reactor recirculation system to assure that design specifications are met. Inspection and testing is carried out as described in <Chapter 3>. The reactor coolant system is thoroughly cleaned and flushed before fuel is loaded initially.

During the preoperational test program, the reactor recirculation system is hydrostatically tested at 125 percent reactor vessel design pressure. Preoperational tests of the reactor recirculation system also include checking 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 are observed; supports are adjusted, as necessary, to assure that components are free to move as designed. Nuclear system responses to recirculation pump trips at rated temperatures and pressures are evaluated during the startup tests, and plant power response to recirculation flow control is determined.

5.4.2 STEAM GENERATORS (PWR)

This section is not applicable to PNPP.

5.4.3 REACTOR COOLANT PIPING

The reactor coolant piping is discussed in <Section 3.9.3> and <Section 5.4.1>. The recirculation loops are shown in <Figure 5.4-1> and <Figure 5.4-2>. The design characteristics are presented in <Table 5.4-1>. Avoidance of stress corrosion cracking is discussed in <Section 5.2.3.4.1>.

5.4.4 MAIN STEAM LINE FLOW RESTRICTORS

5.4.4.1 Safety Design Bases

The main steam line flow restrictors were designed as follows:

a. To 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 main steam line isolation valves (MSIV).

b. To withstand the maximum pressure difference expected across the restrictor, following complete severance of a main steam line.

c. To limit the amount of radiological release outside of the drywell prior to MSIV closure.

d. To provide trip signals for MSIV closure.

5.4.4.2 Description

A main steam line flow restrictor <Figure 5.4-7> is provided for each of the four main steam lines. The restrictor is located in the drywell and is a complete assembly welded into the main steam line.

The restrictor limits the coolant blowdown rate from the reactor vessel in the event a main steam line break occurs outside the containment to the maximum (choke) flow of 6.14×10^6 lb/hr at 1,025 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," 6th edition, 1971.

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 1,375 psi, i.e., the reactor vessel ASME Code limit pressure.

The ratio of venturi throat diameter to steam line inside diameter of approximately 0.515 results in a maximum pressure differential (unrecovered pressure) of about 16 psi at 150 percent of rated flow. This design limits the steam flow in a severed line to less than 150 percent rated flow, yet it results in negligible increase in steam moisture content during normal operation. The restrictor is also used to measure steam flow to initiate closure of the main steam line isolation valves when the steam flow exceeds preselected operational limits.

5.4.4.3 Safety Evaluation

If a main steam line should break outside the containment, the critical flow phenomenon would restrict the steam flow rate in the venturi throat to 170 percent of the rated value. Prior to isolation valve closure, the total coolant losses from the vessel are not sufficient to cause core uncovering and the core is thus adequately cooled at all times.

Analysis of the steam line rupture accident <Chapter 15> shows that the core remains covered with water and that the amount of radioactive materials released to the environs through the main steam line break does not exceed the guideline values of published regulations.

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

The steam flow restrictor is exposed to steam of about 0.2 percent moisture flowing at velocities of 160 ft/sec (steam piping ID) to 630 ft/sec (steam restrictor throat). ASTM A351 (Type 304) cast stainless steel was selected 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; this film is not removed by the steam.

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

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

5.4.4.4 Inspection and Testing

Because the flow restrictor forms a permanent part of the main steam line piping and has no moving components, no testing program is planned. Only very slow erosion will occur with time, and such a slight enlargement will have no safety significance. Stainless steel resistance to erosion has been substantiated by turbine inspections at the Dresden Unit 1 facility, which have revealed no noticeable effects from erosion on the stainless steel nozzle partitions. The Dresden inlet velocities are about 300 ft/sec and the exit velocities are 600 to 900 ft/sec. However, calculations show that, even if the erosion rates are as high as 0.004 in./yr after 40 years of operation the increase in restrictor chocked flow rate would be no more than 5 percent. A 5 percent 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 main steam line isolation valves, individually or collectively, will:

a. Close the main steam lines within the time established by design basis accident analysis to limit the release of reactor coolant.

b. Close the main steam lines slowly enough that simultaneous closure of all steam lines will not induce transients that exceed the nuclear system design limits.

c. Close the main steam line when required despite single failure in either 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. Be able to close the steam lines, either during or after seismic loadings, to assure 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 Description

Two isolation valves are welded in a horizontal run of each of the four main steam pipes; one valve is as close as possible to the inside of the drywell and the other is just outside the containment.

<Figure 5.4-8> shows a main steam line isolation valve. Each is a 26 inch Y-pattern, globe valve. Rated steam flow rate through each valve is 4.07×10^6 lb/hr. The main disc or poppet is attached to the lower end of the stem. Normal steam flow tends to close the valve, and higher inlet pressure tends to hold the valve closed. The bottom end of the valve stem closes a small pressure balancing hole in the poppet. When the hole is open, it acts as a pilot valve to relieve differential

pressure forces on the poppet. Valve stem travel is sufficient to give flow areas past the wide open poppet greater than the seat port area. The poppet travels approximately 90 percent of the valve stem travel to close the main seat port area; approximately the last 10 percent of valve stem travel closes the pilot valve. The air cylinder actuator can open the poppet with a maximum differential pressure of 200 psi across the isolation valve in a direction that tends to hold the valve closed.

A 45 degree angle permits the inlet and outlet passages to be streamlined. This minimizes pressure drop during normal steam flow and helps prevent debris blockage. The pressure drop at 105 percent of rated flow is 7.8 psi maximum. The valve stem penetrates the valve bonnet through a stuffing box that has two sets of replaceable packing. A lantern ring and leakoff drain are located between the two sets of packing. For the outboard main steam valves, the leakoff drain is capped at the valves. The poppet backseats when the valve is fully open to help prevent poppet rotation and stem bending. The stem does not backseat; however, Live Load packing reduces the potential for packing leakage. A stem anti-rotation arrangement precludes stem separation. The outer stem packing of the MSIV is relied upon to prevent leakage.

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

The air cylinder is supported on the valve bonnet by actuator support and spring guide shafts. Helical springs around the spring guide shafts close the valve if air pressure is not available. The motion of the spring seat member actuates switches in the near open, near closed valve positions.

The valve is operated by pneumatic pressure and by the action of compressed springs. The control unit is attached to the air cylinder.

This unit contains air pilot valves and solenoid operated valves. The solenoid valves control opening and closing of the air valves and provide exercising capability at slow speed. Remote-manual switches in the control room enable the operator to operate the valves.

Normal operating air is supplied to the valves from the nonsafety-related plant instrument air system. To assure leak tightness, the outboard MSIVs also utilize the “B” train safety-related instrument air system as a postaccident makeup air supply. The safety-related air supply is manually initiated postaccident and it is assumed that its initiation occurs within one hour after the Design Basis LOCA.

Each valve is designed to accommodate saturated steam at plant operating conditions, with a moisture content of approximately 0.25 percent, 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 should rupture downstream of the valve, steam flow would quickly increase to approximately 150 percent of rated flow. Further increase is prevented by the venturi flow restrictor inside the containment.

During approximately the first 75 percent 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 percent 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 (including exercise cycles) are estimated to be 50 to 400 cycles per year (full open to full close and return).

In addition to minimum wall thickness required by applicable codes, a corrosion allowance of 0.120 inch minimum is added to provide for 40 years service.

Design specification ambient conditions for normal plant operation are 135°F normal temperature, 150°F maximum temperature and 90 percent maximum humidity. Design normal gamma plus neutron radiation dose over a 5 year maintenance period is 7.4 mRad. The inside valves are not continuously exposed to maximum conditions, particularly during reactor shutdown. Valves outside the primary containment and shield building are in ambient conditions that are considerably less severe.

The main steam line isolation valves are designed to close under accident environmental conditions of 330°F for one hour at drywell pressures of 30 psig maximum and 14 psig minimum. In addition, they are designed to remain closed under the following postaccident environment conditions:

a. 330°F for an additional 2 hours at drywell pressure of 15 psig maximum.

b. 310°F for an additional 3 hours at 15 psig maximum.

c. 250°F for an additional 18 hours at 15 psig maximum.

d. 250°F to 100°F ramp during the next 99 days at 15 psig maximum.

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

5.4.5.3 Safety Evaluation

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 main steam isolation valve closure is within specified limits, including instrumentation delay to initiate valve closure after the break. The calculated radiological effects of the radioactive material assumed to be released with the steam are shown to be well within the guideline values for such an accident.

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

The ability of this 45 degree, 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 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 (response time for full closure is set prior to plant operation at 2.5 second minimum, 5.0 second maximum), each valve is tested at 1,000 psig line pressure and no flow. The valve is stroked several times, and the closing time is recorded. The valve is closed by spring only and by the combination of air cylinder and springs. The closing time is slightly greater when closure is by springs only.

b. Leakage is measured with the valve seated and backseated. The specified maximum seat leakage, using cold water at design pressure, is 2 cm^3/hr/in. of nominal valve size. In addition, an air seat leakage test is conducted using 50 psig 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.

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. Tests include radiographic, liquid penetrant or magnetic particle examinations of castings, forgings, welds, hardfacings, and bolts.

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

After the valves are installed in the nuclear system, each valve is tested as discussed in <Chapter 14>.

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

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

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

5.4.5.4 Inspection and Testing

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

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

Leakage from the valve stem packing will become suspect during reactor operation from measurements of leakage into the drywell, or from observations in the steam tunnel. During shutdown while the nuclear system is pressurized, the leak rate through the inner packing of the inboard isolation valves can be measured by collecting and timing the leakage. Leakage through the inner packing would be collected from the packing drain line. For the outboard MSIV, the packing drain line is capped.

The leak rate through the pipeline valve seats (pilot and poppet seats) can be measured accurately using the periodic surveillance tests developed from the requirements in <Section 6.2.6>.

5.4.6 REACTOR CORE ISOLATION COOLING SYSTEM

5.4.6.1 Design Bases

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

a. Should the vessel be isolated and maintained in the hot standby condition.

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

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

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

In the event the reactor vessel is isolated, and the feedwater supply unavailable, relief valves are provided to automatically (or remote manually) maintain vessel pressure within desirable limits. The water level in the reactor vessel will drop due to continued steam generation by decay heat. Upon reaching a predetermined low level, the RCIC system shall be initiated automatically. The turbine driven pump will supply demineralized makeup water from the condensate storage tank to the reactor vessel; an alternate source of water is available from the suppression pool. Suppression pool water is not usually demineralized and hence should only be used in the event that all sources of demineralized water have been exhausted. When the Control Room is notified of the issuance of a tornado warning for the vicinity of the plant, or if a tornado is sighted in the immediate vicinity of the plant, administrative controls require the RCIC suction to be aligned to the tornado missile protected suppression pool. The RCIC turbine will be driven with a portion of the decay heat steam from the reactor vessel, and will exhaust into the suppression pool. The RHR system, in the pool cooling mode, can be used to control pool temperature.

5.4.6.1.1 Residual Heat and Isolation Function

The RCIC system shall initiate and discharge, within 30 seconds, a specified constant flow into the reactor vessel over a specified pressure range. The RCIC water discharged into the reactor vessel varies between a temperature of 40°F up to and including a temperature of 140°F. The mixture of the cool RCIC water and the hot steam does the following:

a. Quenches the steam.

b. Removes reactor residual heat.

c. Replenishes reactor vessel inventory.

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

The RCIC system design includes interfaces with redundant leak detection devices, namely:

a. A high pressure drop across a flow device in the RCIC steam supply line equivalent to 300 percent of the steady-state steam flow at 1,192 psia.

b. A high RCIC equipment area temperature, or high differential temperature (effective when room cooler is running), utilizing temperature switches as described in the leak detection system. High area and high differential temperature (effective when room cooler is running) shall be alarmed in the control room.

c. RCIC steam line low reactor pressure of 50 psig minimum.

d. A high pressure between the turbine exhaust rupture diaphragms.

e. Main steamline tunnel high area temperature or high differential temperature.

f. A high differential pressure in RCIC steam supply line.

These devices, with the exception of the RCIC equipment area high differential temperature activated by the redundant power supplies, automatically isolate the steam supply to the RCIC turbine.

Other isolation cases are defined in the following paragraphs. HPCS provides redundancy for RCIC should RCIC become isolated, hence, providing single failure protection.

Isolation valve arrangements include the following:

a. Two RCIC lines penetrate the reactor coolant pressure boundary. The first is the RCIC steam line which branches off one of the main steam lines between the reactor vessel and the main steam isolation valve. This line has two automatic motor operated isolation valves. One is located inside and the other outside primary containment. An automatic motor operated inboard RCIC isolation bypass valve is used. The isolation signals noted earlier close these valves. These two automatic isolation valves satisfy the requirements of General Design Criterion 55.

b. The RCIC pump discharge line is the other line that penetrates the reactor vessel. This line has two testable check valves (one inside primary containment and the other outside primary containment). Additionally, an automatic motor operated valve is located outside primary containment. These two automatic isolation valves satisfy the requirements of General Design Criterion 55.

c. The RCIC turbine exhaust line vacuum breaker system line has two automatic motor operated valves and two check valves. This line runs between the suppression pool air space and the turbine exhaust line downstream of the exhaust line check valve. Positive isolation shall be automatic via a combination of low reactor pressure and high drywell pressure. This set of valves satisfied the requirements of General Design Criterion 56.

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

d. The RCIC pump suction line and minimum flow pump discharge line, and turbine exhaust line all penetrate the primary containment and are submerged in the suppression pool. The isolation valves for the RCIC pump suction and min flow discharge line require remote-manual operation. The turbine exhaust isolation valve is an automatic motor operated valve. These isolation valves are all outside primary containment and satisfy the requirements of General Design Criterion 56 and 57 respectively.

5.4.6.1.2 Reliability, Operability, and Manual Operation (Also see <Section 5.4.6.2.4> and <Section 5.4.6.2.5>

The RCIC system as noted in <Table 3.2-1> is designed commensurate with the safety importance of the system and its equipment. Each component is individually tested to confirm compliance with system requirements. The system as a whole 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 performed at predetermined intervals throughout the life of the reactor plant.

A design flow functional test of the RCIC system may be performed during normal plant operation by drawing suction from the condensate storage tank and discharging through a full flow test return line to the condensate storage tank. The discharge valve to the head cooling spray nozzle remains closed during the test, and reactor operation remains undisturbed. All components of the RCIC system are capable of individual functional testing during normal plant operation. System control provides automatic return from test to operating mode if system initiation is required. There are three exceptions:

a. Auto/manual initiation on the flow controller. This feature is required for operator flexibility during system operation.

b. Steam inboard and outboard isolation valves. Closure of these valves requires operator action to properly sequence their opening. An alarm sounds when either valve leaves the fully open position.

c. Other bypassed or otherwise deliberately rendered inoperable parts of the system shall be automatically indicated in the control room at the system level.

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

5.4.6.1.3 Loss of Offsite Power

The RCIC system power is to be derived from a highly reliable source that is maintained by either onsite or offsite power <Section 5.4.6.1.1>.

5.4.6.1.4 Physical Damage

The system is designed to the requirements of <Table 3.2-1> commensurate with the safety importance of the system and its equipment. The RCIC is physically located in a different quadrant of the auxiliary building and utilizes different divisional power (and separate electrical routings) than its redundant system as discussed in <Section 5.4.6.1.1> and <Section 5.4.6.2.4>.

5.4.6.1.5 Environment

The system operates for the time intervals and the environmental conditions specified in <Section 3.11>. The RCIC system is for the most part located indoors and is not subject to cold weather conditions. However, the primary RCIC water source consists of underground piping to the condensate storage tank (CST) and the CST itself. There is a nonsafety, CST heating system. Although the CST is the primary source of RCIC water, it is not the safety-related source. The suppression pool provides the safety-related source of RCIC water.

5.4.6.2 System Design

5.4.6.2.1 General

A summary description of the reactor core isolation cooling 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 operation is presented in <Section 5.4.6.2>.

The following diagrams are included for the RCIC systems:

a. A schematic “Piping and Instrumentation Diagram” <Figure 5.4-9> showing 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” <Figure 5.4-10> showing temperature, pressures, and flows for RCIC operation and system process data hydraulic requirements.

The following defines the various electrical interlocks:

a. (Deleted)

b. F031’s limit switch activates when fully open and closes F010, F022 and F059.

c. F068's limit switch activates when full open and clears F045 permissive so F045 could open.

d. F045's limit switch activates when F045 is not fully closed and initiates the startup ramp function and also, after a 15 second time delay, activates turbine pump and gland seal air compressor annunciators. The startup ramp function resets each time F045 is closed.

e. F045's limit switch activates when fully closed and permits F004, F005, F025, and F026 to open and closes F013 and F019.

f. The turbine trip throttle valve (part of C002) limit switch deactivates when fully closed and closes F013 and F019.

g. The combined activation of pressure switches at reactor low pressure and high drywell pressure closes F077, F078 and F068.

h. High turbine exhaust pressure, low pump suction pressure, or an isolation signal actuates and trips the turbine trip throttle valve. When signal is cleared, the trip throttle valve must be reset from the control room.

i. Overspeed of 120 percent trips the mechanical trip at the turbine which closes the trip throttle valve.

j. An isolation signal closes F063, F064, F076, F031, and other valves as noted in item f. and item h. of this section.

k. An initiation signal opens F010 if closed, F013 and F045; starts gland seal system; and closes F022 and F059 if open.

l. High and low inlet RCIC steam supply drain pot levels, respectively, open and close F054.

m. The combined signal of low flow plus pump discharge pressure open and with increased flow closes F019. Also see item e. and item f. of this section.

n. High reactor water level closes F045.

5.4.6.2.2 Equipment and Component Description

Operating parameters for the components of the RCIC system, defined below, are shown on <Figure 5.4-10>. The RCIC components are:

a. One 100 percent capacity turbine and accessories.

b. One 100 percent capacity pump assembly and accessories.

c. Piping, valves and instrumentation for:

    1. RCIC steam supply line.

    2. (Deleted)

    3. Turbine exhaust to the suppression pool. The RCIC turbine exhaust line sparger is described in <Figure 5.4-20>.

    4. Makeup supply from the condensate storage tank to the pump suction.

    5. Makeup supply from the suppression pool to the pump suction.

    6. (Deleted)

7. Pump discharge to the head cooling spray nozzle, including a test line to the condensate storage tank, a minimum flow bypass line to the suppression pool and a coolant water supply to accessory equipment.

The basis for the design conditions is the American Society of Mechanical Engineers (ASME) Section III, Nuclear Power Plant Components.

Design parameters for the RCIC system components are listed in <Table 5.4-2>. See <Figure 5.4-9> for cross-reference of component numbers listed in <Table 5.4-2>.

5.4.6.2.3 Applicable Codes and Classifications

The RCIC system components within the drywell up to and including the outer isolation valve are designed in accordance with ASME Code, Section III, Class 1, Nuclear Power Plant Components. The RCIC system is also designed as Seismic Category I.

The reactor core isolation cooling system component classifications and those for the condensate storage system are given in <Table 3.2-1>.

5.4.6.2.4 System Reliability Considerations

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

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

a. Physical Independence - The two systems are located in separate areas of the auxiliary building. 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 a steam turbine to drive the RCIC and an electric motor-driven pump for the HPCS system. The HPCS motor is supplied from either normal ac power or a separate diesel generator.

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

d. Environmental Independence - Both systems are designed to meet Safety Class 2 requirements. The environment in the equipment rooms is maintained by separate auxiliary systems.

e. Periodic Testing - A design flow functional test of the RCIC can be performed during plant operation by taking suction from the condensate storage tank and discharging through the full flow test return line back to the condensate storage tank. The discharge valve to the head-spray line remains closed during the test, and reactor operation is undisturbed. All components of the RCIC system are capable of individual functional testing during normal plant operation. Control system design provides automatic return from test to operating mode if system initiation is required. The three exceptions are as follows:

1. The auto/manual station on the flow controller. This feature is required for operator flexibility during system operation.

2. Steam inboard and outboard isolation valves. Closure of these valves requires operator action to properly sequence their opening. An alarm sounds when either of these valves leaves the fully open position.

3. Bypassed or other deliberately rendered inoperable parts of the system shall be automatically indicated in the control room.

Additionally, all components of the RCIC system shall be capable of individual functional testing during normal plant operation.

f. General - Inspections and maintenance of the turbine pump unit are conducted periodically. Valve position indication and instrumentation alarms are displayed in the control room.

g. If the steam isolation valves were temporarily closed for maintenance, administrative control, specific operating procedures, and system design preclude the possibility of thermal shock or waterhammer to the steamline, valve seats and discs, and condensate water collection in the turbine.

Operating procedures involve opening the outboard isolation valve, warming the steamline by throttling the warmup valve located on a pipeline bypassing the inboard isolation valve, opening the inboard isolation valve and then closing the bypass valve. This returns the steamline to the standby condition. An interlock between the bypass valve and the inboard valve will prevent the inboard isolation valve from being opened, unless the bypass valve is fully open. The system design has also provided for a drain pot, which drains to the main condenser through a restricting orifice, installed in the low point of the steamline to minimize the amount of condensate that could reach the turbine.

To prevent any liquid buildup at the turbine's exhaust, which could create a condition favorable to a thermal shock or liquid collection in the turbine, a vacuum break system is installed close to the RCIC turbine exhaust line suppression pool penetration to avoid siphoning of water from the suppression pool into the exhaust line as steam in the line condenses during and after turbine operation. The vacuum breaker line runs from the suppression pool air volume to the RCIC exhaust line through two normally open motor operated gate valves and two check valves arranged to allow air flow into the exhaust line, precluding steam flow to the suppression pool air volume.

During turbine operation, condensate buildup in the turbine exhaust line is minimized by the installation of a drain pot in a low point of the line near the turbine exhaust connection. The condensate collected in the drain pot is drained to the CRW when the system is not operating.

5.4.6.2.5 Manual Actions

Manual actions required for the various modes of RCIC are defined below. Caution: System and hardware operating and maintenance instruction documents should be read and understood before any operation is started. Operating precautions shall be observed during operation.

5.4.6.2.5.1 Automatic Operation

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

a. Verify the flow controller has the correct flow set point and is in the automatic mode.

b. Verify that the turbine trip throttle valve, part of E51-C002, is in the full open position. If not fully open, the valve may need to be reset.

   There are two trips for the turbine, i.e., a solenoid operated trip and a mechanical overspeed trip. The overspeed trip must be reset at the turbine itself. Once the mechanical overspeed trip is reset or if only a solenoid trip occurred, the trip throttle valve shall be reset. See <Figure 5.4-9> for component identification.

c. Verify power is available to all components.

d. Verify that the two RCIC steam isolation valves have been properly sequenced open.

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

f. Verify that the two isolation signal logic “reset” devices have been reset.

g. Verify that the water leg pump is running. If essential power to the water leg pump fails, the RCIC system pump can be started up as defined in <Section 5.4.6.2.5> and run until the power is restored.

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

i. Verify that water is available in the condensate storage tank.

j. Verify that oil is available in RCIC turbine oil reservoir; and that the turbine and pump are ready to run as defined by the turbine and pump technical manuals.

k. During extended periods of operation and when the normal water level is again reached, the HPCS system may be manually tripped and the RCIC system flow controller may be adjusted and switched to manual operation. This prevents unnecessary cycling of the two systems. Subsequent starts of RCIC turbine and pump must be operator controlled until rated flow is reached by use of the trip throttle valve or manually initiated if F045 is first closed. If the operator leaves the RCIC in the automatic mode, the system will trip at the high water level and restart automatically at subsequent low water level.

l. Adjust flow controller set point as required to maintain desired reactor water level.

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

n. Close the steam supply valve to turbine F045.

o. Reset the turbine trip throttle valve.

p. Operate the RCIC gland seal compressor for one hour after turbine shutdown.

q. (Deleted)

r. Verify that valves F004, F005, F025, and F026 reopen automatically after valve F045 was closed.

s. Verify that the system is in the standby configuration per <Figure 5.4-9>.

5.4.6.2.5.2 Test Loop Operation

This operating mode is manually initiated by the operator. Test loop operation is performed in the following manner:

a. Verification made in Steps a. through j. of <Section 5.4.6.2.5.1> shall be completed.

b. All motor operated valves shall be positioned as shown on <Figure 5.4-9>.

c. Position F022 to simulate reactor pressure, based on valve position.

d. Start gland seal system.

e. (Deleted)

f. Open F059.

g. Open F045.

h. Verify that valves F004, F005, F025, and F026 automatically closed after valve F045 opened.

i. Observe turbine RPM on speed indicator.

j. Turn RMS switch for F019 to open position and release. Observe that valve F019 cycles fully open and closed by watching position lights. Also observe turbine speed indicator to verify speed increases during this cycling. An increase in speed confirms that the minimum flow line valves and electrical logic properly function.

k. Further adjust F022 to simulate reactor pressure plus line losses to reactor pressure, based on actual discharge pressure.

l. While turbine is running, check and record the following:

    1. Pump suction pressure

    2. Pump discharge pressure

    3. Turbine steam exhaust pressure

    4. Turbine steam inlet pressure

    5. Pump flow

    6. Turbine speed

m. Close F059.

n. Close F022.

o. When the test is completed, manually trip the turbine.

p. Follow Steps n. through s. of <Section 5.4.6.2.5.1>.

5.4.6.2.5.3 (Deleted)

5.4.6.2.5.4 Limiting Single Failure

The most limiting single failure with the RCIC system and its HPCS backup system is the failure of HPCS. With an HPCS failure, if the capacity of RCIC system is adequate to maintain reactor water level, the operator follows <Section 5.4.6.2.5.1>. If, however, the RCIC capacity is inadequate, then <Section 5.4.6.2.5.1> still applies. Additionally, the operator may also initiate the ADS system described in <Section 6.3.2>.

5.4.6.3 <u>Performance Evaluation</u>

The analytical methods and assumptions in evaluating the RCIC system are presented in <Chapter 15> and <Appendix 15A>. The RCIC system provides the flows required from the analysis <Figure 5.4-10> within a 30-second interval based upon considerations noted in <Section 5.4.6.2.4>.

The RCIC system provides vessel makeup flow for varying reactor vessel pressures from 1,215 psia to 165 psia. Rated flow is required up to 1,118 psia, based on operation of the safety relief valves in the relief and low-low-set modes during the vessel isolation transients for which RCIC is designed. The required pump NPSH for this flow is 22 feet. The available NPSH for the RCIC pump for this most limiting operating condition (suction from the suppression pool) is greater than 30 feet.

This most limiting NPSH is based on the pump operating flow of 716 gpm, maximum suppression pool water temperature of 140°F, minimum suppression pool water level at Elevation 589’-0” (minimum drawdown level conservatively assumed), and with suction strainer differential pressure of 0.02 feet corresponding to the strainer fully loaded (i.e., maximum postulated loading resulting from LOCA-generated and pre-LOCA debris materials).

The RCIC pump performance curves are provided as <Figure 5.4-21>.

5.4.6.4 Preoperational Testing

The preoperational and initial startup test programs for the RCIC system are presented in <Chapter 14>.

5.4.6.5 Safety Interfaces

The balance of plant/GE nuclear steam supply system safety interfaces for the reactor core isolation cooling system are:

a. Preferred water supply from the condensate storage tank.

b. All associated wire, cable, piping, sensors, and valves which lie outside the nuclear steam supply system scope of supply.

c. Water supply for testable check.

d. Air supply for solenoid actuated valves.

5.4.7 RESIDUAL HEAT REMOVAL SYSTEM

5.4.7.1 Design Bases

The RHR system is comprised of three independent loops. Each loop contains its own motor-driven pump, piping, valves, instrumentation and controls. Each loop has a suction source from the suppression pool and is capable of discharging water to the reactor vessel via a separate nozzle, or back to the suppression pool via a full flow test line. In addition, the A and B loops have heat exchangers which are cooled by emergency service water. Loops A and B can also take suction from the reactor recirculation system suction or fuel pool, and can discharge into the reactor via the feedwater line, fuel pool cooling discharge, or to the containment spray spargers. The RHR system is located indoors and is not subject to cold weather. Water for the RHR heat exchangers is discussed in <Section 9.2.1>.

5.4.7.1.1 Functional Design Basis

The RHR system has five subsystems, each of which has its own functional requirements. Each subsystem shall be discussed separately to provide clarity.

5.4.7.1.1.1 Residual Heat Removal Mode (Shutdown Cooling Mode)

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, reducing reactor outlet temperature to 125°F within 20 hours after the control rods have been inserted. This will permit refueling when the maximum emergency service water temperature is 70°F, the core is "mature" and the 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 corresponds to a cooldown rate of 100°F per hour with one heat exchanger loop in service.

When the shutdown cooling mode is used, the operator controls the cooldown by throttling the reactor coolant flow through the heat exchanger using E12-F003. The operator determines the cooldown rate by monitoring reactor coolant temperature change with time. The operator will monitor this temperature at intervals not to exceed 30 minutes.

Assuming the operator does not flush the RHR loop while the reactor vessel is in the initial shutdown stage, and assuming that two hours are used for flushing, the minimum time required to reduce vessel coolant temperature to 212°F is depicted by <Figure 5.4-11>.

b. The design basis for the most limiting single failure for the RHR system (shutdown cooling mode) is that one exchanger loop is lost and the plant is then shut down using the capacity of a single RHR heat exchanger loop and related service water capability. <Figure 5.4-12> shows the nominal time required to reduce vessel coolant temperature to 212°F using one RHR heat exchanger loop.

c. If the nonsafety-grade main condenser is not available for reactor shutdown, the safety-grade safety/relief valves are used to depressurize the reactor while the safety grade RCIC system can supply makeup water. Below approximately 135 psig reactor pressure, the safety grade RHR shutdown cooling mode comes in to bring the reactor to the cold shutdown condition.

5.4.7.1.1.2 Low Pressure Injection (LPCI) Mode

The functional design basis for the LPCI mode is to pump a total of 7,100 gpm of water per loop using the separate pump loops from the suppression pool into the core region of the vessel for a design LOCA. Injection flow commences at 225 psi differential pressure between the reactor vessel and the suppression pool air volume outside the drywell.

The initiating signals are: vessel level 1.0 foot above the active fuel or drywell pressure greater than or equal to 2.0 psig. The pumps will attain rated speed in 27 seconds and injection valves to the position required to pass 7100 gpm flow in 37 seconds.

5.4.7.1.1.3 Suppression Pool Cooling Mode

The functional design basis for the suppression pool cooling mode is that it shall have the capacity to ensure that the suppression pool temperature immediately after a blowdown shall not exceed 170°F when the reactor pressure is above 135 psig.

5.4.7.1.1.4 Containment Spray Cooling Mode

The functional design basis for the containment spray cooling mode is to have two redundant means to spray into the containment and suppression pool vapor space, reducing internal pressure to below design limits with bypass leakage from all leakage paths from drywell to containment.

5.4.7.1.1.5 Reactor Steam Condensing Mode

This mode is not used at the Perry Nuclear Power Plant.

5.4.7.1.2 Design Basis for Isolation of RHR System from Reactor Coolant System

The low pressure portions of the RHR system are isolated from full reactor pressure whenever the primary system pressure is above the RHR system design pressure. See <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 line pressure rating. (See <Section 5.2.5> for an explanation of the leak detection system and the isolation signals.)

The RHR pumps are protected against damage from a closed discharge valve by means of automatic minimum flow valves, which open on low main line flow and close on high main line flow.

5.4.7.1.3 Design Basis for Pressure Relief Capacity

The relief valves in the RHR system are sized on one of two bases:

a. Thermal relief only

b. Valve bypass leakage only

Transients are treated by Item a., above; Item b. has resulted from an excessive leak past isolation valves. Valves E12-F055A and B are set at 485 psig to maintain upstream piping within the system design pressure of 500 psig. Valves 1E12-F005, F017A, B and C, and F025A, B and C are set at or below the system design pressure <Figure 5.4-14>.

Relief valve capacity, nominal set points, set point tolerances and ASME class ratings are provided in <Table 5.4-4>.

The vulnerability of the RHR system to malfunctions that could result in overpressurization of low pressure piping is discussed below, and in <Section 5.4.7.2.7>.

The RHR system is connected to higher pressure piping at shutdown suction, shutdown return, LPCI injection, and head spray. In general, pressure interlocks prevent opening valves to the low pressure suction piping where the reactor pressure is above the shutdown range. These same interlocks initiate valve closure on increasing reactor pressure. In addition, a high pressure check valve will close to prevent reverse flow if the pressure should increase. Relief valves in the discharge piping are sized to account for leakage past the check valve. Shutdown suction has two gate valves, F008 and F009, in series which have independent pressure interlocks to prevent opening at higher inboard pressure for each valve. No single active failure or operator error will result in overpressurization of the low pressure piping. In the event of leakage past F008 and F009, the leakage is directed through a passive leakoff line to the suppression pool to alleviate the pressure. PT-N057 provides indication and alarm to the control room operator if the leakoff line is isolated and results in pressurization.

The shutdown return line has a swing check valve, F050, to protect it from higher vessel pressures. Additionally, a globe valve, F053, is located in series and has pressure interlock to prevent opening at high inboard pressures. No single active failure or operator error will cause overpressurization of the lower pressure piping.

The LPCI injection line has a piston check valve, F041, and a high-low pressure interlock, to protect it from higher vessel pressure. No single active failure or operator error will cause overpressurization of the lower pressure piping. The head spray line has a swing check valve, F019, to protect it from higher vessel pressure. Additionally, a gate valve, F023, is located in series and has pressure interlocks to prevent opening at higher inboard pressure. No single active failure or operator error will cause overpressurization of the lower pressure piping.

5.4.7.1.4 Design Basis with Respect to General Design Criterion 5

The RHR system for this unit does not share equipment or structures with any other nuclear unit.

5.4.7.1.5 Design Basis for Reliability and Operability

The design basis for the shutdown cooling mode of the RHR system is that this mode is controlled by the operator from the control room. The only operation performed outside of the control room for a normal shutdown is manual operation of local flushing water admission valves, which is the means for providing clean water to the shutdown portions of the RHR system.

Two separate shutdown cooling loops are provided. Although both loops are required for shutdown under normal circumstances, the reactor coolant can be brought to 212°F in less than 20 hours with only one loop in operation. 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 redundancy, flooding protection, piping protection power separation, etc. required of such systems (see <Section 6.3> for an explanation of the design bases for ECCS systems). Shutdown suction and discharge valves are required to be powered from both offsite and standby emergency power for purposes of isolation and shutdown following a loss of offsite power. If either of the two shutdown supply valves fails to operate, an operator is sent out to operate the valve manually. If this is not feasible and the plant must be shut down as soon as possible, the alternate shutdown method must be employed. In this procedure, water is drawn from the suppression pool and pumped into the reactor using either the LPCI "C" system or the LPCS system. The vessel water is allowed to overflow the steamlines and discharge back to the suppression pool via the ADS valve discharge lines. Simultaneously, the associated Divisional RHR "A" or "B" system is placed in suppression pool cooling mode. This method effectively transfers sensible and decay heat from the vessel to the pool and finally to the emergency service water via the RHR heat exchanger. See <Section 15.2.9> for further discussion on alternate shutdown.

The ADS valves, which are of the Dikkers type, were operationally tested as described in NEDE-24988-P for the most limiting conditions for the alternate shutdown cooling mode. The discharge fluid was single phase liquid water. In addition, the loads from the liquid discharge on the valve and discharge piping were considerably lower than the design basis loads. Both valve and discharge piping integrity were verified following the liquid discharge testing.

The test results also showed that there is a considerable excess of total valve capacity to pass the required flow rate for alternate shutdown cooling, i.e., only 1 or 2 valves are required to pass the needed flow. Consequently, the hydraulic losses will result in a system pressure which is low enough to allow the low pressure pumps to inject sufficient suppression pool water into the vessel. As shown in <Figure 6.3-77>, the additional head required to flood the steamlines will have a negligible effect on the pump flow rate.

The time required to achieve cold shutdown using the alternate shutdown cooling mode can be less than the time to achieve cold shutdown using the normal shutdown cooling mode of the RHR system. However, the recommended rate of cooldown is 100°F per hour.

5.4.7.1.6 Design Basis for Protection from Physical Damage

The design basis for protection from physical damage caused by internally generated missiles, pipe breaks and seismic effects is discussed in <Section 3.5>, <Section 3.6>, and <Section 3.7>, respectively.

5.4.7.2 System Design

5.4.7.2.1 System Diagrams

All of the components of the RHR system are shown in the P&ID, <Figure 5.4-13>. A description of the controls and instrumentation is presented in <Section 7.3.1.1>.

The process diagram and data for the RHR system is shown in <Figure 5.4-14>. All of the sizing modes of the system are shown in the process data. The Functional Control Diagram (FCD) for the RHR system is provided in <Section 7.3>.

Interlocks are provided:

a. To prevent opening vessel suction valves above the suction line or the discharge line design pressure.

b. To prevent inadvertent opening of containment spray valves without at least high drywell pressure for more than ten minutes following DBA.

c. (Deleted)

d. To prevent pump start when suction valve(s) is not open.

5.4.7.2.2 Equipment and Component Description

a. System Main Pumps

The RHR main system pumps are motor-driven deepwell pumps with mechanical seals and cyclone separators. The motors are air cooled by the ventilating system. The pumps are sized on the basis of the LPCI mode (Mode A) and the minimum flow mode (Mode G) of process data in <Figure 5.4-14>. Design pressure for the pump suction structure is 215 psig with a temperature range from 40° to 360°F. Design pressure for the pump discharge structure is 500 psig. The bases for the design temperature and pressure are maximum shutdown cut-in pressures and temperature, minimum ambient temperature, and maximum shutoff head. The pump pressure vessel is carbon steel, the shaft and impellers are stainless steel. A comparison between the available and the required NPSH can be obtained from the pump characteristic curves provided in <Figure 5.4-15> and <Figure 5.4-14> (Note 8).

The RHR pumps require a minimum NPSH of 4 feet at the design flow of 7100 gpm and 6.2 feet at the maximum design runout flow of 8520 gpm. The minimum NPSH requirement specified is at a location of 3 feet above the pump mounting flange, which is approximately 1.25 inches above the pump suction nozzle centerline.

Final design calculations, based on the <Regulatory Guide 1.1> position, indicate an available NPSH in each RHR loop sufficient to ensure pump performance capable of accomplishing the required safety functions. During the post-LOCA mode, with all ECCS pump flows at maximum, the suppression pool water level may reach a design minimum 589’-0” elevation, reducing the static pressure head at the RHR pump suction nozzle, located at Elevation 571’-7”, to a minimum of 17’-5”. At this minimum condition, assuming a maximum pumped water temperature of 185°F, minimum containment pressure of 14.7 psia, suction line losses and a maximum pressure drop of 3.5 feet across the suction strainer, corresponding to the strainer fully loaded (i.e., maximum postulated loading resulting from LOCA-generated and pre-LOCA debris materials), the minimum available NPSH at the manufacturers reference point is greater than 26 feet. Other NPSH requirement identified by the manufacturer are less severe and result in larger NPSH margins.

b. Heat Exchangers

The RHR heat exchangers are sized on the basis of the duty for containment cooling mode (Mode B-1 of the process flow diagram). All other uses of these exchangers require less cooling surface.

Original design information:

Flow rates are 7,100 gpm (design), 7,800 gpm (maximum), on the shell side and 7,300 gpm (rated) on the tube side (emergency service water side). Rated inlet temperature is 185°F shell side and 85°F tube side. The overall heat transfer coefficient is 200 Btu/hr-ft^2-°F. Each loop's heat exchangers contain 15,622 ft^2 of effective surface. Design temperatures of the shell side are 40° to 480°F. Design temperatures of the tube side are 32° to 480°F. Design pressure is 500 psig on both sides, design fouling factors are .0005 shell side and .002 tube side. The construction materials are carbon steel for the pressure vessel with stainless steel tubes and stainless steel clad tubesheet. Temperature elements have been installed on the inlet and outlet piping of the RHR heat exchangers for performance testing.

Current design basis calculations evaluate heat exchanger performance using parameters that differ from the original values shown above. These analyses are performed to confirm that adequate heat transfer will occur when altered conditions are present (e.g., lower tube side cooling flow and altered tube fouling factor values). Therefore, while several of the associated heat exchanger parameters are different than listed above, they are acceptable since the heat removal capability of the heat exchanger under those conditions has been confirmed to be adequate.

c. Valves

All of the directional valves in the system are conventional gate, globe, flow control, and check valves designed for nuclear service.

The injection valves, reactor coolant isolation valves, and pump minimum flow valves are high - speed valves, as operation for LPCI injection or vessel isolation requires. Valve pressure ratings are, as necessary, to provide the control or isolation function, i.e., all vessel isolation valves are rated as Class 1 nuclear valves rated at the same pressure as the primary system.

d. ECCS Portions of the RHR System

The ECCS portions of the RHR system include those sections described through Mode A-1 of <Figure 5.4-14>.

The route includes suppression pool suction strainer, suction piping, RHR pumps, discharge piping including heat exchangers, injection valves, and drywell piping into the vessel nozzles and core region of the reactor vessel.

Pool cooling components include pool suction strainer, suction piping, pumps, heat exchangers, and pool return lines.

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 <Section 7.3> and <Section 7.4>.

5.4.7.2.4 Applicable Codes and Classifications

Codes and classifications applicable to the RHR system are listed in <Table 5.4-5>.

5.4.7.2.5 Reliability Considerations

The residual heat removal system has included the redundancy requirements of <Section 5.4.7.1.5>. Two completely redundant loops have been provided to remove residual heat, each powered from a separate emergency bus. With the exception of the common shutdown line, all mechanical and electrical components are separate. Either loop is capable of shutting down the reactor within a reasonable length of time.

5.4.7.2.6 Manual Action

a. Residual Heat Removal (Shutdown Cooling Mode)

In shutdown operation, when vessel pressure is 135 psig or less, the suppression pool suction valve may be closed for the initial shutdown loop. Locally operated flushing valves are then opened and the stagnant water is flushed to radwaste via valves E12-F040 and F049 operated from the control room. The system may also be flushed with suppression pool water through the suppression pool suction and return lines. At the end of this nominal flush, the lower half of the shutdown loop may be prewarmed by opening vessel suction valves, with effluent directed through the pump to radwaste. The radwaste effluent valves are closed when increasing temperature is noted at the heat exchanger inlet. If required to assist in the warmup, the heat exchanger shell side vent valves may be operated from the control room. The pump is then started at a regulated flow through the return valve E12-F053 and cooldown of the vessel is in progress. Service water flow is initiated prior to starting flow through the heat exchanger. Cooldown rate is subsequently controlled via valves E12-F053 (total flow) and E12-F048 (heat exchanger bypass flow) and E12-F003 (exchanger flow). All operations are performed from the control room except for opening and closing of local flush water valves.

The manual actions required for the most limiting failure are discussed in <Section 5.4.7.1.5>.

5.4.7.2.7 Outline of Operating Procedures

The following description provides a brief outline of the operating procedures required to bring the plant to a cold shutdown condition from hot standby and procedures for plant startup from cold shutdown.

a. Hot Standby to Cold Shutdown

1. Normal Mode: Condenser Available as a Heat Sink

(a) Cooldown from rated pressure to ≤135 psig[1] will use turbine bypass valves with RHR shutdown cooling mode isolated from the RPV.

(b) Flush of RHR system, if required, will be accomplished with condensate transfer pumps; maximum supply pressure is ≤135 psig[1].

(c) Cooldown from ≤135 psig[1] to ≤200°F will use RHR in shutdown cooling mode. Protection to the low pressure piping is offered by isolation valves F008 and F009 at ≤135 psig[1], and closure of check valves F050 and F019.

NOTE:

[1] This pressure will be less than the auto-isolation setpoint for F008 and F009.

2. Off-Normal Mode: Condenser Not Available as a Heat Sink

(a) Safety/relief valves and RCIC will be used as necessary to maintain or reduce RPV pressure below the first safety/relief valve auto-setpoint.

(b) If required by the plant operating procedures/ instructions, initiate the Residual Heat Removal System in the suppression pool cooling mode to maintain the suppression pool temperature within the licensing bases.

(c) Cooldown will continue until RPV pressure is ≤135 psig[(1)], when at least one loop of RHR will be lined up for shutdown cooling mode as needed.

(d) Cooldown from ≤135 psig[(1)] to ≤200°F will be essentially the same as in Item a.l.

b. Cold Shutdown to Hot Standby

1. Normal Mode: Condenser Available as Heat Sink

Normally, RHR operation in the shutdown cooling mode will be secured prior to withdrawing control rods. In any event, it will always be secured prior to RPV water temperature exceeding 200°F.

Heatup to full pressure will use turbine bypass valves.

NOTE:

[(1)] This pressure will be less than the auto-isolation setpoint for F008 and F009.

2. Off-Normal Mode: Condenser Unavailable as a Heat Sink

Normally, RHR operation in the shutdown cooling mode will be secured prior to withdrawing control rods. In any event, it will always be secured prior to RPV water temperature exceeding 200°F.

If heatup beyond 200°F is desired, pressure and level will be controlled with RCIC and/or RWCU.

5.4.7.3 Performance Evaluation

Thermal performance of the RHR heat exchangers is based upon containment cooling with excess capability for the shutdown mode. Because shutdown is usually a controlled operation, maximum emergency service water temperature less 10°F is used as the cooling water inlet temperature. These are nominal design conditions. If the emergency service water temperature is higher, the exchanger capabilities are reduced and the shutdown time is longer and vice versa.

5.4.7.3.1 Shutdown with All Components Available

No typical curve is included here to show vessel cooldown temperatures versus time due to the infinite variety of such curves that may be due to:

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 condition of fouling of the exchangers

c. Operator use of one or two cooling loops

d. Coolant water temperature

e. System flushing time

Since the exchangers are designed for the fouled condition with relatively high emergency service water temperature, the units have excess capability to cool when first cut in at high vessel temperature. Total flow and mix temperature must be controlled to avoid exceeding 100°F per hour cooldown rate. (See <Section 5.4.7.1.1.1> for 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: instrument, set point, logic element, pump, heat exchanger, valve, and 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 in accordance with system data sheets and process data. Logic elements are tested electrically while 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 differences available with pool cooling.

5.4.8 REACTOR WATER CLEANUP SYSTEM

The reactor water cleanup system (RWCS) is classified as a primary power generation system (not an engineered safety feature), a small part of which is part of the reactor coolant pressure boundary (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 RWCS may be operated at any time during planned reactor operations, or it may be shut down if water quality is within limits.

5.4.8.1 Design Basis

5.4.8.1.1 Safety Design Basis

The RCPB portion of the RWCS meets the requirements of <Regulatory Guide 1.26> and <Regulatory Guide 1.29> in order 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 RWCS from the RCPB.

5.4.8.1.2 Power Generation Design Basis

The reactor water cleanup system:

a. Removes solid and dissolved impurities from reactor coolant 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 or radwaste.

c. Minimizes temperature gradients in the main recirculation piping and reactor pressure vessel (RPV) during periods when the main recirculation pumps are unavailable. The operation of the RWCU system with the heat exchangers bypassed and the reactor temperature above 435°F is prohibited since the feedwater piping is not analyzed for temperatures above 435°F.

d. Minimizes the RWCS heat loss.

e. Enables the major portion of the RWCS to be serviced during reactor operation.

f. Prevents the standby liquid reactivity control material from being removed from the reactor water by the cleanup system when required for shutdown.

5.4.8.2 System Description

The system is capable of taking a suction from the inlet of each reactor main recirculation pump and from the reactor pressure vessel bottom head. The process fluid is circulated with the cleanup pumps through a regenerative and nonregenerative heat exchanger for cooling, through the 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 or radwaste <Figure 5.4-16>, <Figure 5.4-17>, and <Figure 5.4-18>.

The major equipment of the reactor water cleanup 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 5.4-3>.

The temperature of the filter demineralizer units is limited by the resin operating temperature. Therefore, the reactor coolant must be cooled before being processed in the filter demineralizer units. The regenerative heat exchanger transfers heat from the 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 building closed cooling water system.

The filter demineralizer units <Figure 5.4-19> are pressure precoat type filters using filter aid and mixed ion-exchange resins. Spent resins are not regenerable and are sluiced from the filter demineralizer unit to a backwash receiving tank from which they are transferred to the radwaste system for processing and disposal. To prevent resins from entering the reactor recirculation system in the event of failure of a filter demineralizer resin support, a strainer is installed on the filter demineralizer unit. Each strainer and filter demineralizer vessel has a control room alarm that is energized by high differential pressure. Upon further increase in differential pressure from the alarm point, the filter demineralizer will automatically isolate.

Analysis will be performed on reactor water cleanup effluent grab samples for chloride with approved Chemistry Instructions. pH analysis will be performed if conductivity is ≥1.0 μmho/cm during power operation. Water conductivity for reactor water cleanup effluent will be monitored and recorded continually on the sampling panel.

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 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.

In the event of 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. High conductivity is annunciated in the control room. The influent sample point is also used as the normal source of reactor coolant grab samples. Sample analysis also indicates the effectiveness of the filter demineralizer units.

The suction line (RCPB portion) of the RWCS contains two motor operated isolation valves which automatically close in response to signals from RPV low water level, leak detection system, actuation of the standby liquid control system (SLCS), and nonregenerative heat exchanger high outlet temperature. <Section 7.6> describes the leak detection system requirements, which are summarized in <Table 5.2-9>. 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, should the SLCS be in operation, and prevents damage of the filter demineralizer resins due to high temperature. The

RCPB isolation valves may be remote 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 gate valve on the return line to the reactor provides long term leakage control. Instantaneous reverse flow isolation is provided by check valves in the RWCS piping.

Operation of the reactor water cleanup system is controlled from the main control room. Resin-changing operations, which include backwashing and precoating, are controlled from a local control panel.

The control logic for the reactor water cleanup system is discussed in <Section 7.3.1>.

5.4.8.3 System Evaluation

The reactor water cleanup system, in conjunction with the condensate treatment system and the fuel pool cooling and cleanup system, maintains reactor water quality during all reactor operating modes (startup, run, refueling, and shutdown).

This type of “pressure precoat” cleanup system was first put into operation in 1971 and is in use in all subsequently started operating BWR plants. Operating plant experience has shown that the RWCS, 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 when the cooling capacity of the regenerative heat exchanger is reduced due to partially bypassing a portion of the return flow to the main condenser or radwaste. The control requirements of the RCPB isolation valves are designed to the requirements of <Section 7.3.1>. The component design data (flowrates, pressure and temperature) are presented in

<Table 5.4-3>. All components are designed to the requirements of <Section 3.2> according to the requirements of the P&ID's, <Figure 5.4-16> and <Figure 5.4-19>.

5.4.9 MAIN STEAM LINE AND FEEDWATER PIPING

5.4.9.1 Design Bases

The main steam line and feedwater piping meet the following criteria:

a. The main steam lines have been designed to conduct steam from the reactor vessel over the full range of reactor power operation.

b. The feedwater lines have been designed to conduct water to the reactor vessel over the full range of reactor power operation.

5.4.9.2 Description

The main steam piping is described in <Section 10.3>. Main steam and feedwater piping is shown in <Figure 5.1-3>.

The feedwater piping consists of two 20-inch outside diameter lines. Each line penetrates the containment and drywell and branches into three 12-inch lines which connect to the reactor vessel. Each 20-inch line includes three containment isolation valves consisting of one check valve inside the drywell, and one motor operated gate valve and one check valve outside the containment. The design pressure and temperature of the feedwater piping between the reactor and maintenance valves (N27-F560A, N27-F560B) are 1,250 psig and 575°F, respectively. From the maintenance valve to the check valve outside containment, the design pressure and temperature are 1,250 psig and 575°F, respectively. From the check valve to the motor-operated gate valve, the design temperature and pressure is 550°F and 1,250 psig. The Seismic

Category I design requirements are placed on the feedwater piping from the reactor up to and including the outboard check valves, including branch piping of 2-1/2 inch or larger nominal piping. The Seismic Category 2 design requirements are placed on the feedwater piping from the outboard check valves up to and including the outboard isolation valves.

In order to meet the requirements of <NUREG-0619>, PNPP's design has RWCU flow routing through the feedwater piping in the tunnel. PNPP is also committed to have the noncladded feedwater nozzles and triple-sleeve with dual piston ring type feedwater spargers.

The materials used in the piping are in accordance with the applicable design code and supplementary requirements described in <Section 3.2>.

The general requirements of the feedwater system are described in <Section 7.1.2>, <Section 7.7.1>, and <Section 10.4.7>.

5.4.9.3 Safety Evaluation

Differential pressure on reactor internals under the assumed accident condition of a ruptured steam line is limited by the use of flow restrictors and by the use of four main steam lines. Main steam and feedwater piping is designed in accordance with the requirements defined in <Section 3.2>. Design of the piping in accordance with these requirements ensures meeting the safety design bases.

5.4.9.4 Inspection and Testing

Inspection and testing are performed as discussed in <Section 3.9> and <Chapter 14>. Inservice inspection is considered in the design of the main steam and feedwater piping. This consideration assures adequate working space and access for the inspection of selected components.

5.4.10 PRESSURIZER

This section is not applicable to PNPP.

5.4.11 PRESSURIZER RELIEF DISCHARGE SYSTEM

This section is not applicable to PNPP.

5.4.12 VALVES

5.4.12.1 Safety Design Bases

Line valves such as gate, globe and check valves are located in the RCPB 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 required in <Section 3.9> 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, will endure the 40-year plant life under the environmental conditions applicable to the particular system when appropriate maintenance is 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 have been shop tested by the manufacturer for performability. The tests and analyses discussed in <Section 3.9> are performed to ensure the operability of active valves. Pressure retaining parts are subject to the testing and examination requirements of Section III of the ASME code. To minimize internal and external leakage past seating surfaces, maximum allowable leakage rates are stated in the design specifications for both backseat and mainseat for gate and globe valves.

Valve construction materials are compatible with the maximum anticipated radiation dosage for the service life of the valves as required in the design specifications.

5.4.12.4 Inspection and Testing

Valves serving as containment isolation valves and which must remain closed or open during normal plant operation may be partially exercised during this period to assure their operability at the time of an emergency or faulted condition. Other valves, serving as a system block or throttling valves, may be exercised when appropriate.

Leakage from critical valve stems (for MSIVs see <Section 5.4.5.4>) is monitored by use of double-packed stuffing boxes with an intermediate lantern leakoff connection for detection and measurement of leakage rates.

Motors used with valve actuators have been furnished in accordance with applicable industry standards. Each motor actuator has been 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 have additionally been tested to demonstrate adequate stem thrust (or torque) capability to open (or close) the valve within the specified time at specified differential pressure. Tests verified no mechanical damage to valve components during full stroking of the valve. Suppliers were required to furnish assurance of acceptability of the equipment for the intended service based on any combination of:

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.

Diagnostic testing is used to further assure proper Motor Operated Valve (MOV) operation. Safety-Related Containment Isolation MOV's are tested for installed operating characteristics which include, but is not limited to, timing, current, switch position, thrust, and/or torque.

MOV torque/thrust calculations are prepared in accordance with Plant Procedures, Industry, and Perry Plant specific information. These calculations, or "Windows," are used to determine the minimum and maximum thrust and/or torque required for the MOV to perform under its normal and design basis condition. The Windows are used as acceptance criteria for the diagnostic testing of the MOV.

The MOV limit switch settings are determined and set as required per maintenance and diagnostic test instructions to assure proper open and closing operations, torque switch bypass features, and stroke times consistent with the Tech Spec requirements.

Inservice inspection of the RCPB is discussed in <Section 5.2.4>.

5.4.13 SAFETY AND RELIEF VALVES

5.4.13.1 Safety Design Bases

a. Overpressure protection has been provided at isolatable portions of RCPB systems in accordance with the requirements set forth in the ASME Code, Section III for Class 1, 2 and 3 components. <Section 5.2.2> discusses RCPB safety/relief valves.

b. The valves are designed in accordance with the requirements listed in <Table 3.2-1>.

c. The design loading, design procedure and acceptability criteria are described in <Section 3.9>.

d. The design and installation details for the mounting of pressure relief devices are described in <Section 3.9>.

5.4.13.2 Description

Safety or pressure relief valves are designed and constructed in accordance with the same code class as that of the line valves in the system.

5.4.13.3 Safety Evaluation

The use of pressure relieving devices will assure that overpressure will not exceed 10 percent above the design pressure of the system. The number of pressure relieving devices on a system or portion of a system has been determined on this basis.

In accordance with ASME code requirements, all safety valves are constructed so that failure of any part cannot obstruct the free discharge of steam or water from the valve.

5.4.13.4 Inspection and Testing

The valves are inspected and tested in accordance with the requirements of the applicable ASME code. In addition, shop performance tests are performed on the valves to ensure their operability in accordance with specification requirements.

No provisions are to be made for inline testing of spring loaded safety/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. Leakage past seating surfaces during normal plant operation is detected by visual examinations or by measuring an increase in discharge line temperature.

5.4.14 COMPONENT SUPPORTS

Support elements are provided for those components included in the RCPB and the connected systems.

5.4.14.1 Safety Design Bases

Design loading combinations, design procedures and acceptability criteria are as described in <Section 3.9.3>. Flexibility calculations and seismic analysis for Class 1, 2 and 3 components conform with the appropriate requirements of ASME Section III.

Support types and materials used for fabricated support elements conform with Sections NF-2000 and NF-3000 of ASME Code Section III. Pipe support spacing guidelines of Table 121.1.4 of ANSI B31.1 Power Piping Code are used as guidance.

5.4.14.2 Description

The use and location of rigid-type supports, variable or constant spring-type supports, snubbers, and anchors or guides are determined by flexibility and seismic/dynamic stress analyses. Component support elements are manufacturers' standard items. Direct weldment to thin wall pipe are avoided where possible.

5.4.14.3 Safety Evaluation

The flexibility and seismic/dynamic analyses performed for the design of adequate component support systems included all transient loading conditions expected by each component. Provisions were made to provide spring-type supports with spring stops where required to prevent damage from dead weight loading during hydrostatic testing.

5.4.14.4 Inspection and Testing

After completion of the installation of a support system, all hanger elements are visually examined to assure that they are in correct adjustment to their cold setting position. Upon hot startup operations, thermal growth is observed to confirm that spring-type hangers function properly between their hot and cold setting positions. Final adjustment capability is provided on all hanger or support types.

5.4.15 REFERENCES FOR SECTION 5.4

1. Ianni, P. W., "Effectiveness of Core Standby Cooling Systems for General Electric Boiling Water Reactors," APED-5458, dated March 1968.

2. General Electric Co., Atomic Power Equipment Department, "Design and Performance of General Electric Boiling Water Reactor Main Steam Line Isolation Valves," APED-5750, dated March 1969.

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	-	30	22
Elbows	3	-	22
Gate Valves	1	-	22
Discharge Line			
Straight Pipe	-	28	24
Elbows	2	-	24
Flow Control Valves	1	-	24
Gate Valves	1	-	24
Discharge Manifold			
Pipe	-	40	16
Reducer Cross	1	-	24 x 16
Contour Nozzle	4	-	16 x 12
Caps	2	-	16
Concentric Reducer	1	-	24 x 12
External Risers			
Straight Pipe	5	7	12
Elbows	5	-	12

Design Pressure (psig)/Design Temperature (°F)

Suction piping and valve up to and including pump suction nozzle	1,250/575
Pump, discharge valves, and piping between	1,650/575
Piping after discharge blocking valve up to vessel	1,550/575
Pump auxiliary piping and cooling water piping	150/125
Vessel bottom drain	1,250/575

Operation at Rated Conditions

Recirculation Pump

Flow, gpm	42,890
Flow, lb/hr	16.20×10^6
Total developed head, ft	780

TABLE 5.4-1 (Continued)

Recirculation Pump (Continued)	
Suction Pressure (static), psia	1,039
Required NPSH, ft	106
Water Temperature (max.), °F	575
Pump Brake HP (min.)	7,200
Flow velocity at pump suction, fps	43
PUMP MOTOR	
Voltage rating	13,200
Speed, rpm	1,782
Motor rating, hp	7,935
Phase	3
Frequency	60
Rotational inertia, lb-ft^2	17,900
JET PUMPS	
Number	20
Total jet pump flow x 10^6 lb/hr	104.0
Throat ID, in.	6.4
Diffuser ID, in.	14.7
Nozzle ID (five each), in.	1.30
Diffuser exit velocity, fps	26.0
Jet pump head, ft	89.67
FLOW CONTROL VALVE	
Type	Ball
Material	Stainless
Type actuation	Hydraulic
Failure mode (on loss of power or control signal)	Fail as is
CV at valve maximum position (min. required)	8,310
Valve size diameter, in.	24
RECIRCULATION BLOCK VALVE, DISCHARGE/SUCTION	
Type	Gate
Actuator	Motor operated
Material	Austenitic stainless steel
Valve size diameter, in. (discharge/suction)	24/22

TABLE 5.4-1 (Continued)

RECIRCULATION BLOCK VALVE, DISCHARGE/SUCTION (Continued)

LOW FREQUENCY MOTOR GENERATOR SET

Motor horsepower	400
Voltage	4,000
Generator frequency	15

TABLE 5.4-2

DESIGN PARAMETERS FOR RCIC SYSTEM COMPONENTS

NOTE: The limiting values for a specific event may vary based on system conditions and associated postulated failures.

a. RCIC Pump Operation (C001)

Flow Rate	Injection Flow - 700 gpm Cooling Water Flow - 16 to 25 gpm Total Pump Discharge - 725 gpm (includes no margin for pump wear)
Water Temperature Range	40°F to 141.5°F
NPSH	21 ft minimum
Developed Head	2,980 ft @ 1,192 psia reactor pressure 610 ft @ 165 psia reactor pressure
BHP, Not to Exceed	825 hp @ 2,980 feet developed head 150 hp @ 610 feet developed head
Design Pressure	1,525 psig
Design Temperature	40°F to 150°F

b. RCIC Turbine Operation

(C002)	H.P. Condition	L.P. Condition
Reactor Press (Sat. Temp.)	1,192 psia	165 psia
Turbine Inlet Pressure	1,177 psia, minimum	150 psia, minimum
Turbine Exhaust Pressure	25 psia, maximum	25 psia, maximum
Design Inlet Pressure	1,250 psig at saturated temperature	
Design Exhaust Pressure	165 psig at saturated temperature	

TABLE 5.4-2 (Continued)

c. RCIC Orifice Sizing	
Minimum Flow Orifice (D005)	Sized for 100 gpm + 10 percent with E51-F019 fully open with the turbine/pump at maximum speed
Test Return Orifice (D006)	Sized with piping arrangement to simulate pump discharge pressure required when the RCIC system is injecting design flow with the reactor vessel pressure at 165 psia. Valve E51-F022 must be throttled for system testing at a simulated 1,015 psia reactor pressure.
Leak-Off Orifices (D008, D010)	Sized for 1/8-inch diameter minimum, 3/16-inch diameter maximum.
Water Leg Pump Minimum Flow Orifice (D011)	Sized for minimum water leg pump flow and located in a pipe run of sufficient length to act as a heat sink thus permitting continuous water leg pump operation without pump overheating.
Flow Element (N001)	
Flow at Full Meter Differential Pressure:	1,000 gpm
Full Meter Differential Pressure:	410 inches water at 68°F
Normal Temperature:	40 to 170°F
System Design Pressure/Temperature:	1,525 psig/140°F
Maximum Unrecoverable Loss at Normal Flow:	4.5 psi
Installed Accuracy:	±1 percent at normal flow and normal temperature
Cooling Loop Back Pressure Orifice (D012)	Sized orifice to maintain 16 to 25 gpm to lube oil cooler based upon pump suction line pressure varying from 50 psig to minimum NPSH value (estimated size is 0.343 inch diameter).

c. (Continued)

Accuracy	
Combined Inaccuracy of Flow Element N001, Flow Transmitter N003 and Flow Indicator R606:	±2.5 percent maximum
Combined Inaccuracy of the Pressure Transmitter N004 or N050, and Pressure Indicator R601:	±2.5 percent maximum

d. Valve Operation Requirements

Steam Supply Valve (F045)	Open and/or close against full differential pressure of 1,177 psi within 15 seconds.
Pump Discharge Valve (F013)	Open and/or close against full differential pressure of 1,400 psi within 15 seconds.
Pump Minimum Flow Bypass Valve (F019)	Open and/or close against full differential pressure of 1,400 psi within 8 seconds.
RCIC Steam Supply Inboard Isolation Valve (F063)	Open and/or close against a full differential pressure of 1,177 psi within 20 seconds.
RCIC Steam Supply Outboard Isolation Valve (F064)	Open and/or close against full differential pressure of 1,177 psi within 20 seconds.
Cooling Water Pressure Control Valve (F015)	Self-contained downstream sensing control valve capable of maintaining constant downstream pressure of 125 psia. Diaphragm of pressure control valve must be of the elastomer type.

TABLE 5.4-2 (Continued)

d. (Continued)

Pump Suction Relief Valve (F017)	100 psig relief setting: 10 gpm at 10 percent accumulation.
Cooling Water Relief Valve (F018)	Sized to prevent over-pressurizing piping, valves and equipment in the coolant loop in the event of failure of pressure control valve F015.
Pump Test Return Valve (F022)	Capable of throttling control against differential pressures up to 1,100 psi and closure against differential pressure of 1,400 psi.
Pump Suction Valve, Suppression Pool (F031)	Capable of opening and closing against 75 psi differential pressure.
Testable Check Valve (F066)	System test mode bypasses this valve, and its functional capability is demonstrated separately. Therefore, valve test provisions are provided, including limit switches to indicate disc movement. The valve and valve associated equipment are capable of proper functional operation during maximum ambient conditions.
Outboard Check Valve (F065)	Is accessible during plant operation and is capable of local testing.
Turbine Exhaust Isolation Valve (F068)	Opens and/or closes against 30 psi differential pressure at a temperature of 330°F. Physically located in the line on a horizontal run, as close to the containment as practical.
Isolation Valve, Steam Warmup Line (F076)	Opens and/or closes against differential pressure of 1,177 psi with minimum travel of 4 inches per minute.
Vacuum Breaker Isolation Valves (F077 & F078)	Opens and/or closes against a differential pressure of 30 psi at a minimum rate of 4 inches per minute.
Vacuum Breaker Check Valves (F079 & F081)	Full flow and open with a minimum pressure drop (less than 0.5 psi) across them.

TABLE 5.4-2 (Continued)

d. (Continued)

Steam Exhaust Drain Pot System Isolation Valves (F004 & F005)	These valves operate only when RCIC system is shut down. They allow drainage to CRW system. They must operate against a differential pressure of 75 psi.
Condensate Storage Tank Isolation Valve (F010)	This valve isolates the condensate storage tank so that suction may be drawn from the suppression pool. Valve must operate against a differential pressure of 75 psi.
Steam Inlet Drain Pot System Isolation Valves (F025 & F026)	These valves allow for drainage of the steam inlet drain pot. They must operate against a differential pressure of 1,177 psi.
Steam Inlet Trap Bypass Valve (F054)	This valve bypasses the trap D003. It must operate against a differential pressure of 1,177 psi.
Pump Test Return Valve (F059)	This valve allows water to be returned to the condensate storage tank during RCIC system test. It must operate against a differential pressure of 1,400 psi.
Thermal Relief Valve (F090)	Size as required to protect the discharge line between valves E51-F022 and E51-F059 from thermal expansion due to abnormal ambient temperature of 212°F and water at 40°F.

e. Rupture Disc Assemblies (D001 & D002)

e. Rupture Disc Assemblies (D001 & D002)	Utilized for turbine casing protection, includes a mated vacuum support to prevent rupture disc reversing under vacuum conditions.
Rupture Pressure Flow Capacity	150 psig ±5 percent at 212°F 75,000 lb/hr at 165 psig

f. Instrumentation

For instrumentation and control definition, refer to <Chapter 7>.

TABLE 5.4-2 (Continued)

g. Condensate Storage Requirements

Total reserve storage for RCIC and HPCS systems is 150,000 gallons.

h. Piping RCIC Water Temperature

The maximum water temperature range for continuous system operation shall not exceed 140°F. However, due to potential short term operation at higher temperatures, piping expansion calculations were based on 170°F.

i. Turbine Exhaust Vertical Reaction Force

The turbine exhaust sparger is capable of withstanding a vertical pressure of 20 psi. This pressure unbalance is due to turbine steam discharge below the suppression pool water level.

j. Ambient Conditions

For various environmental conditions refer to <Figure 3.11-12>.

k. Suction Strainer Sizing

The suppression pool suction strainer shall be sized such that:

1. Pump NPSH requirements are satisfied when the strainer is fully loaded (i.e., maximum postulated loading resulting from LOCA-generated and pre-LOCA debris materials).

2. Particles over 3/32 inch diameter are restrained from passage into the pump and the head spray nozzles (refers to GE supplied components).

TABLE 5.4-3

REACTOR WATER CLEANUP SYSTEM EQUIPMENT DESIGN DATA

SYSTEM FLOW RATE (lb/hr): 154,000

MAIN CLEANUP RECIRCULATION PUMPS

Number required	2
Capacity, % (each)	50
Design temperature, °F	575
Design pressure, psig	1,410
Discharge head at shutoff, ft	600
Minimum available NPSH, ft	13

HEAT EXCHANGERS

	Regenerative	Nonregenerative
Capacity, %	100	100
Shell design pressure, psig	1,420	150
Shell design temperature, °F	575	370
Tube design pressure, psig	1,410	1,410
Tube design temperature, °F	575	575

FILTER DEMINERALIZERS

Type:	pressure precoat
Number required	2
Capacity, % (each)	50
Flow rate per unit, lb/hr	77,000
Design temperature, °F	150
Design pressure, psig	1,410

TABLE 5.4-4

RHR RELIEF VALVE DATA

Valve	Rated Capacity (gpm)	Set Pressure Maximum (psig)	ASME Class
E12F005[1]	24.4	185	Section III, Class 2
E12F017A,B[1]	22.8	200	Section III, Class 2
E12F017C[1][4]		100	Section III, Class 2
E12F025A,B,C[2]	35.6	485	Section III, Class 2
E12F055A,B[2]	138,600 lbs/hr[3] (required capacity)	485	Section III, Class 2

NOTE:

(1) The setpoint tolerance is ±3% of the nominal set pressure excluding effects on set pressure due to temperature of springs.

(2) The setpoint tolerance is ±10% of the nominal set pressure excluding effects on set pressure due to temperature of springs. The setpoint tolerance was expanded from the original ±3% tolerance as allowed by Appendix I of the ASME OM Code.

(3) Safety relief valves E12F055A,B were originally designed and certified with a capacity of 138,600 lbs/hr for steam service. Due to the elimination of the Steam Condensing Mode of RHR, these valves are used for liquid/water service as thermal relief protection for the RHR Heat Exchangers. Re-certification of E12F055A,B for water service is not required, but the valve manufacturer has supplied a calculated capacity for water of 3442 gpm.

(4) Being that the relief valve is a thermal relief valve there is no bounding flow requirement.

TABLE 5.4-5

RHR CODES AND CLASSIFICATIONS[1]

	Code	Classification
Piping, Pumps and Valves		
a. Process Side	ASME III	1/2
b. Service Water Side	ASME III	3
Heat Exchangers		
a. Process Side	ASME III	2
	TEMA	C
b. Service Water Side	ASME III	3
	TEMA	C

NOTE:

(1) IEEE Standards 279 and 308 are applicable to electrical portions of the RHR system.