U.S. NUCLEAR REGULATORY
COMMISSION
DOCKET 50-410
LICENSE NPF-69

NINE MILE POINT NUCLEAR STATION UNIT 2

UPDATED SAFETY ANALYSIS REPORT

OCTOBER 2016

REVISION 22

LIST OF EFFECTIVE FIGURES

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REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

5.1 SUMMARY DESCRIPTION

The reactor coolant system (RCS) includes those systems and components which contain or transport fluids coming from, or going to, the reactor core. These systems and the reactor vessel form the reactor coolant pressure boundary (RCPB). This chapter provides information regarding the RCS and pressure-containing appendages out to and including the outside isolation valves. The following group of components is defined as the RCPB.

The RCPB includes all pressure-containing components such as pressure vessels, piping, pumps, and valves, which are:

1. Part of the RCS, or

2. Connected to the RCS, up to and including any or all of the following:

    a. The outside containment isolation valve in piping which penetrates the primary reactor containment.

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

    c. The RCS safety/relief valves (SRV).

Section 5.4 also discusses various subsystems closely allied to the RCPB.

The nuclear system pressure relief system protects the RCPB from damage due to overpressure. To protect against overpressure, SRVs are provided that can discharge steam from the nuclear system to the suppression pool. The pressure relief system also acts to automatically depressurize the nuclear system in the event of a loss-of-coolant accident (LOCA) 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. Various loading combinations are considered in the vessel design. The vessel meets the requirements of applicable codes

and criteria described in Sections 3.2, 5.2.1, and Appendix 5A. 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.

Venturi-type main steam line (MSL) flow restrictors are installed in each MSL inside the primary containment. The restrictors are designed to limit the loss of coolant resulting from a main steam line break (MSLB) 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 isolation valves (MSIV) to close. This action protects the fuel barrier.

Two isolation valves are installed on each MSL, one located inside and the other located outside the primary containment. In the event that a MSLB occurs inside the containment, closure of the isolation valve outside the primary containment acts to seal the primary containment itself. The MSIVs automatically isolate the RCPB in the event a pipe break occurs downstream of the inside isolation valves. This action limits the loss of coolant and the release of radioactive materials from the nuclear system. Details of the MSIVs are given in Section 5.4.5.

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 automatically initiated upon receipt of a low reactor water level signal or manually by the Operator. Water is pumped to the core by a steam turbine-driven pump.

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 (e.g., hot standby). One mode of RHR operation allows the removal of heat from the primary containment following a LOCA. Another operational mode of the RHR system is low-pressure coolant injection (LPCI). The LPCI mode of operation is part of the

emergency core cooling system (ECCS) used during a postulated LOCA. This operation is described in Section 6.3.

The reactor water cleanup (RWCU) 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 RCS and its various components are found in Table 5.4-1.

5.1.1 Schematic Flow Diagram

Schematic flow diagrams of the RCS denoting all major components, principal pressures, temperatures, flow rates, and coolant volumes for normal steady state operating conditions at rated power are presented on Figures 5.1-1a and 5.1-1b.

5.1.2 Piping and Instrumentation Diagram

Piping and instrumentation diagrams (P&IDs) covering the systems included within the RCS and connected systems are presented as follows:

System	Figure
Nuclear boiler	5.1-2
Main steam	10.1-3
Feedwater	10.1-6
Recirculation	5.4-2
Reactor core isolation cooling	5.4-9
Residual heat removal	5.4-13
Reactor water cleanup	5.4-16

5.1.3 Elevation Drawing

Elevation drawings showing the general arrangement of the reactor and coolant system in relation to the containment are shown on Figures 1.2-6 through 1.2-12.

5.2 INTEGRITY OF REACTOR COOLANT PRESSURE BOUNDARY

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

5.2.1 Compliance with Codes and Code Cases

5.2.1.1 Compliance with 10CFR50, Section 50.55a

Table 3.2-4 shows compliance with 10CFR50. Code editions, applicable addenda, and component dates are in accordance with 10CFR50.55a.

5.2.1.2 Applicable Code Cases

The reactor pressure vessel (RPV) and appurtenances, and the RCPB piping, pumps, and valves are 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.

Regulatory Guides (RG) 1.84 and 1.85 provide a list of ASME design and fabrication code cases that have been approved by the regulatory staff. Code cases on this list may be used for design, fabrication, or installation until annulled. Annulled cases are considered active for equipment that has been contractually committed to fabrication prior to the annulment. The various ASME Code cases that were applied to components in the RCPB are listed in Table 5.2-1.

5.2.2 Overpressure Protection

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

5.2.2.1 Design Basis

Overpressure protection is provided in conformance with 10CFR50 Appendix A, General Design Criterion (GDC) 15.

SRVs are in conformance with ASME Section III, Article NB-7000. Preoperational and startup instructions are given in Chapter 14.

5.2.2.1.1 Safety Design Bases

The nuclear pressure-relief system has been designed:

1. To prevent overpressurization of the nuclear system that could lead to the failure of the RCPB.

2. To provide automatic depressurization for small and intermediate breaks in the nuclear system occurring with maloperation of the HPCS system, so that the LPCI and the low pressure core spray (LPCS) systems can operate to protect the fuel barrier.

3. To permit verification of its operability.

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

5.2.2.1.2 Power Generation Design Bases

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

1. Discharge to the containment suppression pool.

2. Correctly reclose following operation so that maximum operational continuity can be obtained (Section 1.10, TMI Item II.K.3.16.).

5.2.2.1.3 Discussion

The ASME Boiler and Pressure Vessel Code requires that each vessel designed to meet Section III be protected from overpressure under upset conditions as discussed in Section S.3 of GESTAR II[1].

Setpoints of the 18 SRVs are listed in Table 5.2-2. These setpoints satisfy the ASME Code specifications for safety valves, because all valves open at less than the nuclear system design pressure of 1,250 psig.

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

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

1. Meet requirements of ASME Section III.

2. Qualify for the rated nameplate capacity credit for the overpressure protection function.

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

The SRV discharge piping is designed, installed, and tested in accordance with ASME Section III.

5.2.2.1.4 Safety/Relief Valve Capacity

The SRV capacity is adequate to limit the primary system pressure, including transients, to the requirements of the ASME Boiler and Pressure Vessel Code, Section III, Division 1, 1971 Edition up to and including Winter 1972 Addenda. The essential

ASME requirements which are all met by this analysis are discussed as follows.

It is recognized that the protection of vessels in a nuclear power plant is dependent upon many protective systems to relieve or terminate pressure transients. Installation of pressure-relieving devices may not independently provide complete protection. The safety valve sizing evaluation assumes credit for operation of the scram protective system which may be tripped by either one of two sources, i.e., a direct or flux trip signal. The direct scram trip signal is derived from position switches mounted on the MSIVs, or the turbine stop valves, or from pressure switches mounted on the dump valve of the turbine control valve (TCV) hydraulic actuation system. The analytical limit for the position switch settings is 85 percent fully open for MSIVs and 90 percent fully open for the turbine stop valves. The pressure switches are actuated when a fast closure of the TCVs is initiated. Further, no credit is taken for power operation of the SRVs in the relief mode. Credit is taken for the dual-purpose SRVs in the safety mode.

The rated capacity of the SRVs is sufficient to prevent a rise in pressure within the protected vessel of more than 110 percent of the design pressure (1.10 x 1,250 psig = 1,375 psig) for events defined in Chapter 15 and Appendix A.

Full account is taken of the pressure drop on both the inlet and discharge sides of the valves. All SRVs discharge into the suppression pool through a discharge pipe from each valve which is designed to achieve sonic flow conditions through the valve, thus providing flow independence to discharge piping losses. Additional measures to counteract the effects of backpressure in the SRV discharge lines are discussed in Sections 5.2.2.2.3 and 5.2.2.4.1.

The method described in Reference 5 shows that sonic flow is achieved through a SRV with the following dimensions:

Nozzle Bore	4.84 in (d_o)
Valve Discharge Diameter (at outlet flange)	~10 in (d_1)
Valve Inlet Diameter	~8 in (based on sweepolet)

The SRV steam flows from the steam line, a large reservoir, through the sweepolet (same diameter as inlet flange), into the valve nozzle, and out through the outlet flange. The nozzle has a short flow length, and it acts as a standard nozzle or venturi tube. Values of critical pressure ratio, r_c, are found as a function of d_o/d_1 and k on page A-21 of the reference. The value of r_c, the ratio of discharge backpressure, P_2, to inlet pressure, P_1, decreases as d_o/d_1 decreases.

The value of d_o/d_1 is minimized when the outlet valve flange diameter is used as the value of d_1. In this case, d_o/d_1 = 4.84/10 = 0.484.

The critical pressure for sonic flow occurs where r_c = 0.553 (using k = 1.3 for steam) and, therefore, sonic flow occurs when $P_2 \leq (0.553) P_1$ (P_1, P_2, in psia).

SRV discharge lines are required to be designed and configured so that the discharge backpressure at the valve outlet is not greater than 40 percent of the inlet pressure using pressures measured in psig. For absolute pressure, the corresponding limit is less than 41 percent of inlet across the range of operating conditions.

This limit and discharge line design ensures that r_c will not be exceeded. Therefore, sonic flow is ensured.

Table 5.2-3 lists the systems that could initiate during the design basis overpressure event.

5.2.2.2 Design Evaluation

5.2.2.2.1 Method of Analysis

The model used to analyze overpressurization is provided in Section S.2.3 of GESTAR II[1].

5.2.2.2.2 System Design

Based on a representative equilibrium fuel cycle, a parametric study was conducted to determine the required steam flow capacity of the SRVs based on the following assumptions. Cycle-specific information is covered in Appendix A, Section A.5.2.2.2.2. Operation with a single recirculation system loop in operation, or with one MSIV out of service, has also been evaluated. See Appendices 15B and 15D, respectively.

Operating Conditions

The operating conditions are:

1. Operating power = 4,068 MWt (102 percent of nuclear boiler rated power).

2. Vessel dome pressure = 1,022 psig.

3. Steam flow = 18.07 x 10^6 lb/hr (102.4 percent of nuclear boiler rated steam flow).

These conditions are the most severe because maximum stored energy exists at these conditions. At lower power conditions, the transients would be less severe. See Appendix A Section A.5.2.2.2.2 for operating conditions that apply under TRACG methods for Cycle 15 and forward.

A transient analysis study has been performed for a typical BWR to investigate the effects of increasing the initial reactor

pressure on the peak transient vessel pressure. Two models, one from the REDY and one from the ODYN codes, were used in the study. The model in the REDY code is more conservative than that in the ODYN code. The conclusion, even for the more conservative model, was that increasing the initial operating pressure up to the high-pressure scram setpoint (analytical upper limit of 1,086 psig) results in an increase of the peak system pressure of less than half the initial pressure increase for the overpressure design transient (i.e., all MSIV closure with indirect high neutron flux scram). The same general trend is expected to exist for Nine Mile Point Nuclear Station - Unit 2 (Unit 2). Since there is a significant margin (greater than 50 psi with two SRVs out of service (OOS) by comparing the peak vessel pressure with the ASME Code limit of 1,375 psig) for Unit 2, no safety concern would result from the above-assumed initial dome pressure.

Transients

The overpressure protection system must accommodate the most severe pressurization event described in Section S.3 of GESTAR II[1]. Table 5.2-4 lists the sequence of events for this worst-case transient, the MSIV closure with flux scram, based on the installed SRV capacity, a representative equilibrium fuel cycle, and two SRVs OOS.

Safety/Relief Valve Transient Analysis Specification

1. Valve groups: Spring-action safety mode - 5 groups

2. Spring pressure setpoint (maximum safety limit) and number of valves per group:

Group 1:	1,200 psig	2 SRVs
Group 2:	1,210 psig	4 SRVs
Group 3:	1,221 psig	4 SRVs
Group 4:	1,231 psig	4 SRVs
Group 5:	1,241 psig	4 SRVs

The assumed spring setpoints in the analysis are 3 percent above the actual nominal setpoints to account for initial setpoint errors and any instrument setpoint drift that might occur during operation. Conservative SRV response characteristics are also assumed.

Safety/Relief Valve Capacity

Sizing of the SRV 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.

Reference 7 provides sufficient information and documentation to show compliance with all requirements of Article NB-7000 of the ASME Boiler and Pressure Vessel Code, Section III, Nuclear Power Plant Components, Division 1, 1971 Edition, with Addenda to and

including Winter 1972, in the area of overpressure protection design of the Unit 2 nuclear pressure vessel and other RCPB components. The effects of valve capacity on the pressure transients are shown also.

The overpressure protection analysis also includes the simulation of anticipated transient without scram (ATWS) recirculation pump trip (RPT) on high reactor pressure.

5.2.2.2.3 Evaluation of Results

Safety/Relief Valve Capacity

The required SRV capacity is determined by analyzing the pressure rise from a MSIV closure with flux scram transient as documented in Section S.3 of GESTAR II[1]. Results of this analysis are shown on Figure 5.2-1.

Pressure Drop in Inlet and Discharge

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

Cycle-specific evaluation is covered in Appendix A, Section A.5.2.2.2.3.

5.2.2.3 Piping and Instrument Diagrams

Figures 5.2-2 and 10.1-3 show the schematic location of pressure-relieving devices for:

1. Reactor coolant system.

2. Primary side of the auxiliary or emergency systems interconnected with the primary system.

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

The schematic arrangements of the SRVs are shown on Figures 5.2-2 and 5.2-3.

5.2.2.4 Equipment and Component Description

5.2.2.4.1 Description

The nuclear pressure relief system consists of 18 SRVs located on the MSLs between the reactor vessel and the first isolation valve

within the drywell. These valves protect against overpressure of the nuclear system.

The SRVs provide three main protection functions:

1. Overpressure Relief Operation The valves open automatically using a pneumatic actuator to limit a pressure rise.

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

3. Depressurization Operation The automatic depressurization system (ADS) valves open automatically as part of the 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 5.2-4.

Chapter 15 discusses the events that are expected to activate the primary system SRVs. Chapter 15 also summarizes the number of valves expected to operate during the initial blowdown of the valves and the expected duration of this first blowdown. For several of the events it is expected that the lowest set SRV will reopen and reclose as generated heat drops into the decay heat characteristics. The pressure increase and relief cycle continues 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 main control room is recommended to minimize the total number of these discharges, with the intent of achieving extended valve seat life. The design and position indication of the SRVs comply with the requirements of NUREG-0737, as discussed in Section 1.10.

A schematic of the SRV is shown on Figure 5.2-5. It is opened by either of two modes of operation:

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

2. The power-actuated (relief) mode of operation which consists of using an auxiliary actuating device consisting of a pneumatic piston/cylinder and mechanical linkage assembly which opens the valve by overcoming the spring force, even with valve inlet pressure equal to 0 psig.

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

For overpressure SRV operation (self-actuated or spring lift mode), the spring load establishes the SRV 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 SRV is a backup to the relief function described in the following paragraph. The spring-loaded valves are designed and constructed in accordance with ASME Section III, Subsubarticle 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 that 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) does not exceed 0.1 sec. The maximum elapsed time between signal to actuator and full open position of the valve does not exceed 0.25 sec.

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

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

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

1. 340°F for 3 hr, at drywell pressure ≤45 psig.

2. 320°F for an additional 3-hr period, at drywell pressure ≤45 psig.

3. 250°F for an additional 18-hr period, at 25 psig.

4. Duration of operability is 2 days at 200°F and 20 psig, following which the valves remain fully open or closed for 97 days, provided air and power supply is

available. No power/air supply is required to keep the valve closed.

The ADS uses seven selected SRVs to depressurize the reactor system to below the shutoff head of the low-pressure ECCS. ADS is not required for large breaks but is necessary following a small break with failure of the high-pressure inventory makeup systems. The analytical results for this event have changed over the years due to changes in the LOCA models. Currently, for this event, using the specified number of ADS valves, it has been demonstrated in Section 6.3.3 that the calculated peak cladding temperature (PCT) is well below the 2200°F limit. These calculations are in strict accordance with 10CFR50 Appendix K, including the single failure criteria.

Each of the SRVs 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 45 psig. The accumulator capacity is sufficient for each ADS valve to provide one actuation against the maximum drywell design pressure with the reactor pressure of 0 psig.

Each SRV discharges steam through a discharge line to a point below the minimum water level in the suppression pool.

The SRV discharge lines (SRVDL) are classified as Quality Group C and Category I. SRVDL piping from the SRV to the suppression pool consists of two parts. The first is attached at one end to the SRV and attached at its other end to a pipe anchor. The main steam piping, including this portion of the SRV discharge piping, is analyzed as a complete system. The second part of the SRV discharge piping extends from the anchor to the suppression pool. Because of the upstream anchor on this part of the line, it is physically decoupled from the main steam header and is therefore analyzed as a separate piping system.

The SRV 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 may cause excessive pressure at the valve discharge when the valve is opened again. For this reason, two vacuum relief valves are provided on each SRVDL to prevent drawing an excessive amount of water up into the line as a result of steam condensation following termination of relief operation.

The effects of flow-induced SRVDL backpressure on the performance of the SRV are counteracted by: (a) the proper sizing of the SRVDL to ensure that the steady state backpressure does not exceed 40 percent of the SRV inlet pressure (steady state maximum, 550 psig) to assure the SRV rated flow capacity; and (b)

the orificed vent line backpressure control scheme to assure proper performance of the SRV.

When the valve is self-pressure actuated, flow-induced backpressure is developed in the SRVDL and in the SRV body bowl cavity that is located above the disc and piston. Backpressure in this area tends to close the valve. To control this effect, an orificed vent line is attached to the SRV and sized to permit flow from that cavity area. This vent line controls the effective backpressure buildup and maintains the required force balance needed to keep the SRV open and to permit proper blowdown (reclosure).

Each valve was full flow at set-pressure, and blowdown tested by the manufacturer to assure that it would function under the plant-specific SRVDL backpressures.

When the valve is actuated by the electropneumatic actuator assembly in the relief mode of operation, there is no backpressure effect on the performance of the valve since the actuator controls the opening and closing functions of the valve. The actuator and related linkages are sized to withstand the pressure forces created by the SRVDL-induced backpressure.

The SRVs are located on the MSL piping, rather than on the reactor vessel top head, 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.

The nuclear pressure relief system automatically depressurizes the nuclear system sufficiently to permit the LPCI and the LPCS system to operate as a backup for the HPCS system. Further descriptions of the operation of the automatic depressurization feature are found in Sections 6.3 and 7.3.1.1.

5.2.2.4.2 Design Parameters

The specified operating transients for components within the RCPB are given in Section 3.9B.1. Refer to Section 3.7B for discussion of the input criteria for design of Category I structures, systems, and components.

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

5.2.2.4.3 Safety/Relief Valve

The discharge area of the valve is 18.4 sq in and the coefficient of discharge K(D) is equal to 0.873 (K=0.9 K(D)).

The SRV discharge coefficient is established by test[6] and the flow capacity is calculated[6].

$$W = 51.5 \times A \times P \times K$$

$$W = 51.5 \times (4.843)^2 \frac{(\pi)}{4}(1.03 \times 1,205 + 14.7)(0.786)$$

$$W = 936,454 \frac{lb}{hr}$$

Where:

W = Steam flow $\frac{lb}{hr}$

A = Nozzle area, in^2

P = 1.03 x set-pressure + 14.7 psia

K = Average coefficient (Reference 6)

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. The design pressure and temperature for the SRV discharge piping are 570 psig and 485°F.

SRV cyclic testing has demonstrated an expected service life of at least 60 actuation cycles between required maintenance. Discharge of pipeline debris through the valve will, however, adversely affect seat leakage. For a schematic cross section of the valve, see Figure 5.2-5.

5.2.2.5 Mounting of Pressure Relief Devices

The SRVs are located on the main steam piping header. The mounting consists of a special, forged-contoured nozzle and an oversized flange connection. This provides a high-integrity connection that withstands the thrust, bending, and torsional loadings to which the main steam pipe and relief valve discharge pipe are subjected, including:

1. Thermal expansion effects of the connecting piping.

2. Dynamic effects of the piping due to safe shutdown earthquake (SSE) and suppression pool loads.

3. Reactions due to transient unbalanced wave forces exerted on the SRVs 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 are self-equilibrated by the valve discharge piping.)

4. 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.

5.2.2.6 Applicable Codes and Classification

The vessel overpressure protection system is designed to satisfy the requirements of ASME Section III. The general requirements for protection against overpressure of ASME Section III recognize that reactor vessel overpressure protection is one function of the reactor protective systems and allows the integration of pressure relief devices with the protective systems of the nuclear reactor. Hence, credit is taken for the scram protective system as a complementary pressure protection device. The NRC has also adopted the ASME Codes as part of their requirements in 10CFR50.55a.

5.2.2.7 Material Specification

Material specifications of pressure-retaining components of SRVs are reported in Table 5.2-5.

5.2.2.8 Process Instrumentation

Overpressure protection process instrumentation is listed in Table 1.7-1, Figure 5.2-4 and shown on P&IDs, Figures 5.1-2 and 10.1-3.

5.2.2.9 System Reliability

The SRVs have a high reliability as they are designed to satisfy the requirements of ASME Section III. The consequences of failure are discussed in Sections 15.1.4 and 15.6.1.

5.2.2.10 Inspection and Testing

The inspection and testing applicable to SRVs uses a quality assurance (QA) program that complies with Appendix B of 10CFR50. The SRVs were tested at the vendor's shop in accordance with quality control procedures to detect defects and to prove operability prior to installation. The following tests were conducted:

1. Hydrostatic test at ASME-specified test conditions.

2. Seat leakage measurements are made with steam during the set-pressure test.

3. Set-pressure test: Each valve is pressurized with saturated steam, with the pressure rising to the valve

set-pressure. The valve must open at the nameplate set-pressure ±1 percent.

4. Response time test: Each SRV is tested to demonstrate acceptable response time.

The equipment specification requires certification from the valve manufacturer that design and performance requirements have been met, including capacity and blowdown requirements. The setpoints are adjusted, verified, and indicated on the valves by the vendor. Specified manual and automatic actuation relief mode of each SRV is verified during the preoperational test program.

QA review of certification and/or witnessing or inspection of parts and tests performed by the manufacturer assure compliance to the specification requirements. In addition, ASME-certified Boiler Inspectors perform independent inspections and reviews for manufacturer's compliance to ASME requirements. Customer reviews and inspections of the manufacturer further assure achievement of the compliance needed.

The valve throat bore diameter and valve lift are measured and recorded to confirm compliance with ASME capacity.

The equipment specification includes all of the design requirements necessary for the operation of the valve in its expected normal and abnormal environments. The valve actuator, which includes the pneumatic cylinder and the electrically-operated nitrogen supply solenoids, was environmentally qualified by test to IEEE-323-71.

The components of the electropneumatic SRV operator, subjected to environmental test conditions required by IEEE-323, are similar to the components of the electropneumatic SRV operator provided with the SRV. However, the seismic testing of the environmentally-aged operator was not sequentially performed, as required by IEEE-323. Seismic testing of a fully-assembled SRV, including another electropneumatic operator, was performed separately. Testing to demonstrate operability under the 1- to 100-day accident environment condition was limited to 3 days.

The valve actuator is the only part of the SRV that sees environmental conditions more severe than those induced by the process steam when the SRV is opened. In addition, the actuator consists of elastomeric parts that are susceptible to degradation induced by environment conditions. Hence, the actuator was subjected to the following sequence of tests to qualify for normal and abnormal service conditions:

<u>Aging</u> - The actuator was subjected to radiation, thermal, and mechanical aging that was equivalent to 5 yr of service life plus the accident radiation dosage.

Abnormal Transient - The actuator was subjected to the abnormal ambient conditions, as shown in Figure 5.2-5a. The peak abnormal temperature transient was repeated to provide adequate confidence. The temperature was also increased over the FSAR table by 10°F to add margin. The test was cut off after the fourth day, as the solenoid valves capability to function in the steady state energized condition with an ambient temperature of 250°F was demonstrated, and the solenoid valves were not required to change state (i.e., from opening to closing or vice versa) after the test period.

The valve assembly was seismically qualified by test to IEEE-344-75.

Three significant SRV test programs, using prototype units which are fully representative of the production units, demonstrated satisfactorily the proof testing under environmental conditions with specified time periods. The test programs were:

Life Cycle Test - This test consisted of 300 powered and safety actuations of a valve assembly with saturated steam pressurizing the inlet nozzle/disc.

Pneumatic Actuator Environmental Test - This test, which includes the electrically-operated solenoids and the air cylinder, consists of a minimum of 1,000 pneumatic cylinder actuations that meet the environmental requirements of IEEE-323-71.

Seismic Test - This test consisted of subjecting a valve assembly to a series of operating basis earthquake (OBE) and SSE with moments applied to the inlet and outlet flanges that met the seismic testing requirements of IEEE-344-75.

A randomly-selected production SRV was life-cycle tested to verify there were no differences in operation characteristics between the prototype valve and the production valves. This test consisted of 450 powered and safety actuations of a valve assembly with saturated steam pressurizing the inlet nozzle/disc. The results met all the requirements.

The SRV performance is monitored by the following three procedures:

1. Acoustic monitors and thermocouples are installed in the discharge line of each SRV. A noise level or temperature increase in the discharge piping will indicate steam leakage/flow across the nozzle seat. The temperature change cannot be used for a quantitative steam flow.

2. The SRVs are fully tested during the startup of the reactor prior to turnover for commercial operation.

3. Included with the SRV is an instruction manual provided to the customer. Within this manual will be recommended periodic maintenance programs as recommended by the manufacturer based upon his experience.

During the inservice (operational) phase, all main steam SRVs will be subject to the following tests and inspections in accordance with the IST program:

During every refueling outage, a sample of the installed valves will be tested for verification of set-pressures, opening and closing using the pneumatic power actuator, testing of all bolted closures, and testing of pneumatic actuator leakage. Valve sample size will be in accordance with the IST program plan.

After the preceding testing, the valves will undergo preventive maintenance in accordance with an approved procedure.

All disassembled valves will be inspected for wear, damage, and erosion. All gaskets, seals, and parts will be replaced as needed in accordance with inspection results. Valves will be relapped, as required, and lubricated. All disassembled valves will be retested, and appropriate adjustments will be made prior to use.

It is not feasible to test the SRV setpoints while the valves are in place. The valves are mounted on 1,500-lb primary service rating flanges. They can be removed for maintenance or bench checks and reinstalled during normal plant shutdowns. SRVs are tested and adjusted in accordance with the Code, 10CFR50.55a, Technical Specifications, and the IST program. The external surface and seating surface of all SRVs are 100-percent visually inspected when the valves are removed for maintenance or bench tests. Valve operability was verified during the preoperational test program as discussed in Chapter 14.

A discussion of SRV operability testing for two-phase flow, in accordance with NUREG-0737, is provided in Section 1.10, Task II.D.1.

5.2.3 Reactor Coolant Pressure Boundary Materials

5.2.3.1 Material Specifications

Table 5.2-5 lists the principal pressure-retaining components and materials and the appropriate material specifications for the RCPB components.

5.2.3.2 Compatibility with Reactor Coolant

5.2.3.2.1 PWR Chemistry of Reactor Coolant

Not applicable to boiling water reactors (BWRs).

5.2.3.2.2 BWR Chemistry of Reactor Coolant

Materials in the RCS are primarily austenitic stainless steel, carbon 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 unusual materials in the coolant. Chloride limits are specified to prevent stress corrosion cracking of stainless steel[2].

The water quality requirements are supported by General Electric Company (GE) stress corrosion test data summarized as follows:

1. 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 a pH of 7. Test specimens were bent beam strips stressed over their yield strength. After 2,100-hr exposure, no cracking or failures occurred.

2. 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 a pH of 7. Uniaxial tensile test specimens were stressed at 125 percent of their 550°F yield strength. No cracking or failures occurred at 15,000-hr exposure.

When conductivity is in its normal range, pH, chloride, and other impurities affecting conductivity are also within their normal range[2]. When conductivity becomes abnormal, chloride measurements are made to determine whether they are also out of their normal operating values. Conductivity may be high due to the presence of a neutral salt which does not have an effect on pH or chloride. In such a case, high conductivity alone is not a cause for shutdown. In some types of water-cooled reactors, conductivities are high because of the purposeful use of additives. In BWRs, however, the only additive is a small amount of zinc, and where near-neutral pH is maintained, conductivity provides a good and prompt measure of the quality of the reactor water. Significant changes in conductivity provide the Operator 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 out-of-specification condition include operation of the RWCU system or 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.

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

Normal Plant Operation

The BWR system water chemistry is described by following the system cycle shown on Figure 5.2-6. 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. The process consists of filtration and 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 and enters the reactor vessel at an elevated temperature and with a chemical composition typically 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 are formed. Due to steam generation, stripping of these gases from the water 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) oxygen in the condensate. A minimum oxygen concentration level of 30 ppb in the condensate and feedwater systems is maintained by the oxygen in feedwater injection system (Section 10.4.11), which injects oxygen into the suction side of the condensate pumps.

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.

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

equivalent to feedwater in 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.

<u>Plant Conditions Outside Normal Operation</u>

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

<u>Plant Depressurized and Reactor Vented</u> During certain periods such as refueling and maintenance outages, the reactor is vented to the atmosphere. Under these circumstances the reactor cools and the oxygen concentration increases to a maximum value of 8 ppm (Table 5.2-6). Equilibrium between the atmosphere above the reactor water surface, the CRD cooling water input, any residual radiolytic effects, and the bulk reactor water is established after some time. No other changes in water chemistry of significance take place during this plant condition because no appreciable inputs take place.

<u>Plant Transient Conditions - Plant Startup/Shutdown</u> During these conditions, no significant changes in water chemistry other than oxygen concentration take place.

1. <u>Plant Startup</u> 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 decreases rapidly as a function of water temperature increase resulting in correspondingly reduced oxygen solubility in water. The oxygen level reaches a minimum of about 20 ppb (0.02 ppm) at a coolant temperature of about 380°F, at which point an increase takes place due to significant radiolytic oxygen generation. Up to this point, the oxygen is

degassed from the water and is displaced to the steam dome above the water surface.

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

2. <u>Plant Shutdown</u> Upon plant shutdown following power operation, the radiolytic oxygen generation essentially ceases as the fission process is terminated. Because oxygen is no longer generated, while some steaming still takes place due to residual energy, the oxygen concentration in the coolant decreases to a minimum value determined by steaming rate temperature. If venting is performed, a gradual increase to essentially oxygen saturation at the coolant temperature takes place, but does not exceed a maximum value of <8 ppm oxygen.

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

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

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

Grab samples are provided, for the locations shown in Table 9.3-1, for special and noncontinuous measurements such as pH, oxygen, chloride, and radiochemical measurements. The relationship of chloride concentration to specific conductance measured at 25°C for chloride compounds such as sodium chloride

and hydrochloric acid can be calculated, as shown on Figure 5.2-7. Conductance values for these compounds essentially bracket those for 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 are quickly 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 per the Technical Requirements Manual (TRM). For the higher than normal limits of <1 umho/cm, more frequent sampling and analyses are invoked by the coolant chemistry surveillance program.

The chemistry of the RCS system shall be maintained within the limits specified in TRM Section 3.4.1 at all times.
The primary coolant conductivity monitoring instrumentation is listed in Table 9.3-1 and is further described in Section 9.3.2.5.

<u>Water Purity During Condenser Leakage</u> The design of the Unit 2 condensate demineralizer (CND) system is in conformance with RG 1.56. The operational aspects of the CND system are in accordance with the operational guidance of RG 1.56.

5.2.3.2.3 Compatibility of Construction Materials with Reactor Coolant

Construction materials exposed to the reactor coolant consist of the following:

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

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

3. Carbon steel and low-alloy steel.

4. Some 400 series martensitic stainless steel (all tempered at a minimum of 1,100°F).

5. Colmonoy and Stellite hardfacing material.

All of these construction materials are highly resistant to stress corrosion in the BWR coolant. General corrosion on all materials, except carbon and low-alloy steel, is negligible.

Conservative corrosion allowances are provided for all exposed surfaces of carbon and low-alloy steels.

Contaminants in the reactor coolant are controlled to very low limits by the reactor water quality specifications. No detrimental effects will occur to 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

Construction materials exposed to external insulation are:

1. Solution-annealed austenitic stainless steels. Types 304, 304L with 0.035-percent maximum of carbon content, 316 and 316K with 0.02-percent weight maximum of carbon content.

2. Carbon and low-alloy steel.

Two types of external insulation are employed on BWRs. Stainless steel reflective metal insulation used does not contribute to any surface contamination and has no effect on construction materials. Similarly, the fibrous (nonmetallic) insulation is encapsulated in metal sheeting which prevents direct contact with the RCS materials. In addition, the fibrous insulation used is assessed to meet the requirements of RG 1.36, and has the proper ratios of leachable sodium and silicate ions to chloride and fluoride ions.

Since the only additive is a small amount of zinc in the BWR coolant, leakage would basically expose materials to high-purity, demineralized water. Exposure to demineralized water would cause no detrimental effects.

5.2.3.2.5 Monitoring BWR Structural Components Exposed to Reactor Coolant

A crack arrest verification (CAV) system is installed in Unit 2 to provide indication and to record the performance of plant materials in the primary coolant environment. The CAV is an in-line system which exposes precracked fracture mechanics test specimens made from BWR structural materials such as Inconel 182, Inconel 600, and Type 304 stainless steel. At Unit 2, the CAV system consists of three specimens installed in a test vessel and load frame, a water chemistry monitoring station, and electrochemical potential (ECP) monitor. Changes in crack length are detected by measuring variations in voltage drop across each

crack. In addition to providing information related to crack growth, the CAV records key water chemistry parameters such as levels of dissolved oxygen, primary coolant conductivity and water temperature.

The CAV system is located in the secondary containment of the reactor building and is nonsafety related. Suction flow is taken from the recirculation (2RCS) sample line (Figure 5.4-2b), and discharge flow is connected to the RWCU (2WCS, Figure 5.4-16a). The crack extension rate test (CERT) pressure vessel for the test specimens is designed and fabricated per ASME Section VIII, Division 1, 1986 Edition. Valves, fitting, tubing, seals and other pressure-retaining components meet ANSI B31.1 standards.

5.2.3.3 Fabrication and Processing of Ferritic Materials

5.2.3.3.1 Fracture Toughness

Materials in the RCPB, other than the RPV, are required by 10CFR50.55a and Appendix G to meet the fracture toughness requirements of ASME Section III, NB-2300. These fracture toughness requirements for ferritic piping, valve, bolting, and pump materials are met as follows:

1. Piping and weld filler materials are in accordance with ASME Section III, NB-2300, 1974 Edition; field weld filler materials are to 1974 Edition.

2. Valves are in accordance with ASME III, NB-2300, as follows:

    a. Motor-operated valves (MOVs), Winter 1975, except as noted in Appendix 5A, 1971 Edition and Winter 1973 Addenda.

    b. MSIVs, Summer 1977.

    c. Carbon Steel Manual Valves, Winter 1973.

3. Materials for bolts with diameters exceeding 1 in meet the 25-mil lateral expansion requirement of ASME Section III, NB-2300, of the same code date as the associated equipment. In addition, bolting greater than 1 in is required to meet a minimum of 45 ft-lb absorbed energy.

4. There are no ferritic pumps in the RCPB.

The fracture toughness properties of the RPV are discussed in Section 5.3.1 and Appendix 5A.

5.2.3.3.2 Control of Welding

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

RG 1.50 delineates preheat temperature control requirements and welding procedure qualifications supplementing those in ASME Sections III and IX.

The use of low-alloy steel is restricted to the RPV. Other ferritic components in the RCPB are fabricated from carbon steel materials. Preheat temperatures employed for welding of low-alloy steel meet or exceed the recommendations of ASME Section III, Appendix D. Components were either held for an extended time at preheat temperature to assure removal of hydrogen, or preheat was maintained until postweld heat treatment. The minimum preheat and maximum interpass temperatures were specified and monitored.

Control of Electroslag Weld Properties (Regulatory Guide 1.34)

No electroslag welding was performed on RCPB components.

Welder Qualification for Areas of Limited Accessibility (Regulatory Guide 1.71)

Qualification for areas of limited accessibility is discussed in Section 5.2.3.4.2.

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

RPV specifications require that all low-alloy steel be produced to fine grain practice. The requirements of this regulatory guide are not applied to BWR vessels.

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 Subsubarticle NB-2550 of ASME Section III. These RCPB components met the requirements of ASME Codes existing at the time of placement of the order. At the time of placement of the orders, 10CFR50 Appendix B requirements and the ASME Code requirements assured adequate control of quality for the products.

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 that are not immediately used are placed in storage ovens that are maintained at or above 250°F (150°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 that are damaged, wet, or contaminated are discarded. If any electrodes are inadvertently left out of the ovens for more than one shift, they are discarded or reconditioned in accordance with the manufacturer's instructions.

5.2.3.4 Fabrication and Processing of Austenitic Stainless Steels

5.2.3.4.1 Avoidance of Stress Corrosion Cracking

Avoidance of Significant Sensitization

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

All austenitic stainless steel was purchased in the solution heat-treated condition in accordance with applicable ASME and ASTM specifications. Cooling rates from solution heat-treating temperatures were required to be rapid enough to prevent sensitization.

Materials have been selected to minimize the possibility of intergranular stress corrosion cracking (IGSCC). Except as discussed below, all wrought austenitic stainless steel in the RCPB is required to be low carbon Type 304L or 316L. Recirculation system piping is Type 316K with 0.020-percent maximum carbon. Cast austenitic stainless steels are required to contain 5-percent minimum delta ferrite. Since these materials have all been demonstrated to be highly resistant to oxygen-assisted stress corrosion in the as-installed condition, the RCPB is in complete compliance with NUREG-0313, Revision 2.

Portions of the RCPB which normally operate below 200°F (e.g., instrument lines beyond the primary containment penetration and local drain valves off the reactor recirculation system) are wrought austenitic stainless steel but are not L-grade based on Item 1 below.

In addition to the RCPB, the balance-of-plant ASME III Class 1, 2 and 3 piping and components 4-in nominal pipe diameter and larger comply with NUREG-0313, Revision 2, except as noted below:

1. Where the normal operating temperature is 200°F or less, the line has not necessarily been revised to comply with NUREG-0313, Revision 2. Test data and

analysis of actual in-service failures demonstrate that there is an insignificant risk of IGSCC in systems which normally operate at temperatures of 200°F or less in normal BWR environments.

2. Where the normal operating temperature exceeds 200°F for an extremely short period of time (less than 1 percent of the total design life of the plant), the line has not necessarily been revised to comply with NUREG-0313, Revision 2. In these cases, the total length of time in which these lines are exposed to temperatures greater than 200°F is insignificant with respect to the service conditions that cover IGSCC.

3. In accordance with NUREG-0313, Revision 2, the weld (RCS 64-00-FWA-007) has been categorized as having been extensively repaired and therefore classified as a Category D weld. This weld will receive augmented examination during the first refueling outage and every second refueling outage thereafter.

4. The piping welds subject to NUREG-0313, Revision 2, have been classified per the Unit 2 ISI Program Plan. They will be examined in accordance with the frequency and selection criteria of the Unit 2 ISI Program Plan using PDI examination techniques.

For piping, in manual welds with the gas tungsten arc (GTAW) and shielded metal arc (SMAW) welding processes, the heat input was limited by weaving and welding technique restrictions. Nonweaving (stringer bead) techniques were used where possible. When required, weaving was controlled to meet the following bead width limits: for GTAW, the lesser of five times the filler wire diameter or 7/16 in; for SMAW, the lesser of 4 times the electrode core wire diameter or 5/8 in. For automatic welding, heat input was restricted to 50,000 joules/in. Interpass temperature was restricted to 350°F for all stainless steel welds. High heat welding processes such as block welding and electroslag welding were not permitted. All weld filler metal and castings were required by specification to have a minimum of 5-percent delta 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 resolution heat treated. These controls were used to avoid severe sensitization and to comply with the intent of RG 1.44.

<u>Process Controls to Minimize Exposure to Contaminants</u>

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

care is 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 meets the requirements of RG 1.44.

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

Avoidance of Hot Cracking

RG 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 that are approved by GE, other approved vendor, or Nine Mile Point Nuclear Station (NMPNS) management are required for all primary pressure boundary welds. These procedures comply with the requirements of ASME Sections III and IX and applicable regulatory guides.

The areas where Unit 2 practices comply with RG 1.31 are as follows:

1. Verification of test results to 10CFR50 requirements for reporting (in accordance with ASME Code).

2. The minimum acceptable ferrite content of 5 percent (or 5 ferrite number [FN]) in undiluted weld pads and constitutional diagram evaluations.

3. Exemption of austenitic stainless steel cladding and Type 16Cr-8Ni-2Mo filler metal from ferrite control.

4. Measurements to be made prior to production welding on each heat and each lot of filler metal and/or flux combination.

Areas where some degree of variation occurred are as follows:

1. No maximum value was placed on ferrite content; however, RG 1.31 allows waivers in situations where the maximum may be exceeded.

2. Gauge Types for Measurement - The Unit 2 RPV requirements did not specify the type of magnetic measuring gauge, its calibration, or examination

criteria to be used. Magnetic gauges were used. The type and calibration technique data may not be the same as required by RG 1.31.

3. Pad preparation technique for Unit 2 welds is in accordance with ASME Code SFA 5.9 in GE specifications which differ somewhat from AWS 5.4-74 techniques specified by RG 1.31. The ferrite variation due to these different techniques for pad deposition is not significant.

The variations that exist are considered minor in respect to accomplishing the basic purpose of RG 1.31. The specifications go beyond the regulatory guide requirements in that low carbon grades of filler metal were specified for improved corrosion resistance, and the need for a minimum ferrite content was recognized and actions implemented to assure its presence before RG 1.31 was published in any of its revisions. The main purpose of RG 1.31 was to provide added assurance by testing and documentation that the specified ferrite actually was present in the filler metal used in production. The procedures accomplished this even though there may have been slight differences in technique.

<u>Electroslag Welds (Regulatory Guide 1.34)</u>

Electroslag welding was not employed for RCPB components.

<u>Welder Qualification for Areas of Limited Accessibility (Regulatory Guide 1.71)</u>

RG 1.71 states 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 ASME Sections III and IX. It also requires additional performance qualifications for welding in areas of limited access. All ASME Section III welds were fabricated in accordance with the requirements of ASME Sections III and IX. There are few restrictive welds involved in the fabrication of BWR components. Welder qualification for welds with the most restrictive access was accomplished by mockup welding. Mockups were examined by radiography or sectioning.

5.2.4 Inservice Inspection and Testing of Reactor Coolant Pressure Boundary

This section discusses the Unit 2 ISI program and IST program for ASME Class 1 components. The ISI program and the IST program were developed to meet the requirements of 10CFR50.55a and the inservice inspection and testing Codes.

5.2.4.1 System Boundary Subject to Inspection

The RPV, system piping, pumps, valves, and components within the RCPB defined as ASME Class 1 are designed and fabricated to permit compliance with ASME Section XI as required by 10CFR50.55a(b). The examination procedures required for ISI have been considered in the design of components, weld joint configurations, and system arrangement to assure access for inspection. Where required, access is provided for a volumetric and surface examination of pressure-retaining welds from the external surface. Periodic design reviews and onsite audits are performed throughout the design and erection phase to assure that these objectives are being met.

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

1. Reactor pressure vessel.
2. Main steam lines.
3. Reactor feedwater lines.
4. Reactor recirculation lines.
5. RHR system lines.
6. HPCS and LPCS lines.
7. RCIC system lines.
8. Core DP line.
9. RWCU lines.
10. CRD housing.
11. RPV vent lines.
12. Reactor drain line.
13. Standby liquid control line.

5.2.4.2 Provisions for Access to the Reactor Coolant Pressure Boundary

5.2.4.2.1 Reactor Pressure Vessel

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

1. The shield wall and vessel insulation behind the shield wall are erected away from the RPV outside surface. Access ports are located at each RPV nozzle and at the base of the shield wall. The annular space between the RPV 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 at the base of the shield wall or through removable insulation panels at the top of the shield wall.

2. Access to the RPV 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.

3. The vessel flange area and vessel closure head can be examined during normal refueling outages using manual ultrasonic methods. 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. Removal of these components also enables examination of the remaining internal RPV components utilizing remote visual techniques. The examination of the flange-to-vessel weld can be performed manually from the flange seal surface.

4. 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 RPV studs are accessible for required examinations during refueling either in place or when removed.

5. Openings in the RPV support skirt provide access for manual or automated ultrasonic methods for examination of the RPV meridional and circumferential welds within the support skirt. Welds that are inaccessible will be identified in the ISI program plan.

5.2.4.2.2 Pipe, Pumps, and Valves

Arrangements

Physical arrangement of pipe, pumps, and valves provides personnel access to weld locations. Working platforms are

provided at areas to facilitate servicing of pumps and valves. Platforms and ladders are provided to gain access to piping welds including the pipe-to-vessel nozzle welds. Removable thermal insulation is provided on welds and components that require access for ISI.

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

Examination techniques and procedures, including any special techniques and procedures, will be written in accordance with the requirements of Table IWB-2500-1 of ASME Section XI.

5.2.4.3.1 Equipment for In-service Inspection

Manual ultrasonic examination is planned for the preservice inspection and the subsequent in-service examination of the welds in the RPV top and bottom heads including the flange-to-vessel weld. Remote ultrasonic scanning will be used to examine the circumferential, longitudinal and nozzle-to-vessel welds on the balance of the vessel. The ISI program will delineate the equipment and methods used for ISI.

5.2.4.3.2 Coordination of Inspection Equipment with Access Provisions

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

5.2.4.3.3 Recording and Comparing Data

Manual data recording will be performed where manual examinations are performed. Electronic data recording and comparison analysis will be employed with automated examination equipment. Each ultrasonic transducer will be fed into an individual channel from which the key parameter of the reflectors will be recorded. The data to be recorded for both manual and automated methods will be in accordance with mandatory Appendix III of ASME Section XI.

5.2.4.4 Inspection Intervals

ISI intervals will be in accordance with Subarticle IWA-2400 in Section XI of the ASME Boiler and Pressure Vessel Code. Each interval is divided into inspection periods in accordance with Table IWB-2412-1 (Inspection Program B) in Section XI. The inservice schedule and inspections to be performed during each period and interval are defined in the ISI program plan.

5.2.4.5 Inservice Inspection Program Categories and Requirements

Examination categories and requirements are defined in the ISI program plan, and closely follow the categories and requirements specified in Table IWB-2500-1 of ASME Section XI.

5.2.4.6 Evaluation of Examination Results

Results of the preoperational examinations and subsequent inservice examinations will be evaluated in accordance with Article IWB-3000 of ASME Section XI.

5.2.4.7 System Leakage and Hydrostatic Pressure Tests

System leakage tests and hydrostatic tests will be performed as required in accordance with IWA-5000 and IWB-5000 of ASME Section XI, and the Inservice Pressure Testing (ISPT) program plan. The scheduling, conduct, and acceptance criteria for these tests are described in the ISPT program plan and its implementing documents. Visual examinations for evidence of leakage will be performed during these tests. Insulation and components need not be removed during the tests.

5.2.4.8 Inservice Inspection Commitment

All safety class components will be examined once prior to startup in accordance with the requirements in Technical Specifications. This preservice inspection examination satisfied the requirements of ASME XI for the RPV and associated piping, pumps, and valves. The preservice examinations and tests were based on the ASME Boiler and Pressure Vessel Code, Section XI, 1980 Edition through the Winter 1980 Addenda, except for the extent of examination of ASME Class 2 piping of the RHR system, ECCS, and containment heat removal system. The extent of examination for these systems was based on ASME Section XI, 1974 Edition through Summer 1975 Addenda.

Portions of ANSI B31.1 upgraded piping ≥2 1/2 in in the main steam system, extending from the outboard MSIVs up to and including the turbine stop valves, including branch connection lines ≥2 1/2 in up to the first isolation valve in the branch lines, were examined in accordance with ASME Section XI, 1980 Edition through Winter 1980 Addenda.

(See Section 3.9A.6 for the IST program for pumps and valves.) Subsequent ISIs will be performed in accordance with the requirements of 10CFR50.55a(g) as described in the ISI program.

Additionally, the ISI program for piping identified in Nuclear Regulatory Commission (NRC) Generic Letter (GL) 88-01 will be performed in accordance with the NRC staff positions on schedule, methods, personnel, and sample expansion included in this generic

letter. GL 88-01 applies to all BWR piping made of austenitic stainless steel that is 4 in or larger in nominal diameter and contains reactor coolant at a temperature above 200°F during power operation, regardless of ASME Code classification. Other NRC approved positions may be used in lieu of related positions in GL 88-01; e.g., BWRVIP-75 guidance, as documented in the ISI program.

5.2.5 RCPB and ECCS Leakage Detection System

5.2.5.1 Leakage Detection Methods

The nuclear boiler leak detection system (LDS) consists of temperature, pressure, level, flow, airborne gaseous and particulate fission product sensors, and process radiation sensors with associated instrumentation used to indicate and alarm leakage from the RCPB. The LDS in certain cases is used to initiate signals used for automatic closure of isolation valves to shut off leakage external to the containment. The system is assessed to be in conformance with RG 1.45. Those portions of the system which affect automatic isolation of leakage are designed to IEEE-279-1971 (refer to Table 3.2-1).

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

1. Main steam lines.
2. RWCU system.
3. RHR system.
4. RCIC system.
5. Feedwater system.
6. HPCS.
7. Coolant systems within the containment.
8. LPCS.
9. RPV.
10. Miscellaneous systems.

Leak detection methods used to obtain conformance with RG 1.45 for plant areas inside the primary containment differ from those for areas located outside the primary containment. These areas are considered separately in the following sections.

5.2.5.1.1 Detection of Leakage Within the Primary Containment

The primary detection methods for small unidentified leaks within the primary containment include continuous monitoring of drywell floor drain tank fill rate and airborne gaseous and particulate radioactivity increases. (The sensitivities of these primary detection methods for unidentified leakage within the primary containment are listed in Table 5.2-8.) These variables are continuously indicated and/or recorded in the control room. If the unidentified leakage increases to a total of 5 gpm, the detecting instrumentation channel(s) will trip and activate an alarm in the main control room. This does not result in a containment isolation signal.

The secondary detection methods (i.e., the monitoring of pressure and temperature of the primary containment atmosphere) are used to detect gross unidentified leakage. High primary containment pressure will alarm and trip the isolation logic which results in closure of the containment isolation valves.

The detection of small identified leakage within the primary containment is accomplished by continuous drywell equipment drain tank fill rate monitoring. An alarm will be activated in the main control room when the leak rate reaches 25 gpm averaged over a 24-hr period.

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

Excessive leakage inside the primary containment (e.g., process line break or LOCA within primary containment) is detected by high primary containment pressure, low reactor water level or high steam line flow (for breaks downstream of the flow elements). The instrumentation channels for these variables trip when the monitored variable exceeds a predetermined limit to activate an alarm and trip the isolation logic which closes appropriate isolation valves (Table 5.2-9). The alarms and indication and isolation trip functions initiated by the LDSs are summarized in Tables 5.2-9 and 5.2-10.

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

The detection of leakage within the reactor building (outside the primary containment) is accomplished by detection of increases in

reactor building floor drain sump and reactor building equipment drain tank fillup time and pumpout time (Section 5.2.5.2.2). The reactor building floor drain sump monitors will detect unidentified leakage increases and activate an alarm in the main control room. The reactor building equipment drain tank monitors will detect identified leakage increases and activate an alarm in the main control room when leakage increases above normal background levels. See Section 5.2.5.2.2 for a discussion of the fuel pool liner leakage detection method.

5.2.5.1.3 Detection of Leakage External to the Primary Containment

Areas outside the primary containment that are monitored for primary coolant leakage are: equipment areas in the auxiliary bays, the main steam tunnel, and the turbine building, i.e., main steam line tunnel lead enclosure (MSLTLE). 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.

These areas are monitored by dual-element thermocouples for sensing high ambient temperature in all these areas and high differential temperature between the inlet and outlet ventilation ducts in the main steam tunnel. The temperature elements are located or shielded so that they are sensitive to air temperature only and not to radiated heat from hot piping or equipment. Increases in ambient and/or differential temperature indicate leakage of reactor coolant into the area. The monitors located in the main steam tunnel and turbine building (i.e., MSLTLE) have sensitivities suitable for detection of increases in ambient air temperature which are equivalent to reactor coolant leakage into the monitored areas of 25 gpm for main steal tunnel and 45 gpm for MSLTLE or less. The temperature trip setpoints are a function of room size and the type of ventilation provided. These monitors provide alarm, indication, and recording in the main control room, and trip the isolation logic to close selected isolation valves (e.g., the main steam tunnel monitors close the MSIV and MSL drain isolation valves and other valves [Table 5.2-9]).

The turbine building (i.e., MSLTLE) temperature monitors alarm and indicate in the main control room and trip the isolation logic to close the main steam isolation and MSL drain isolation valves when leakage exceeds 45 gpm. These sensors monitor ambient temperature in the enclosed space between the steam tunnel outlet and the inlet to the high pressure turbine.

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 temperature in the piping or equipment areas, high differential flow, and low main condenser vacuum. These monitors provide alarm and indication in the main control room and trip the isolation logic to cause

closure of appropriate system isolation valves on indication of excess leakage (Table 5.2-9). Setpoints for the high ambient temperature monitors in the piping and equipment areas of the reactor building and auxiliary bays are based on limiting the maximum environmental conditions of these areas to within the environmental qualification capabilities of the applicable equipment.

5.2.5.1.4 Intersystem Leakage Monitoring

Leakage from the HPCS, LPCS, RCIC, and RHR systems outside containment is detected by a combination of methods, including high area temperature, high area radiation, high sump level, and RPV condition (see Section 5.2.5.1.3).

Radiation monitors are used to detect reactor coolant leakage into cooling water systems supplying the RHR heat exchangers and the RWCU nonregenerative heat exchanger. These monitoring channels are part of the process radiation monitoring system (RMS). Process radiation monitoring channels monitor for leakage into each common cooling water header downstream of the RHR heat exchangers and the RWCU nonregenerative heat exchanger. Channels will alarm on high radiation conditions indicating process leakage into the cooling water. No isolation trip functions are performed by these monitors.

A radiation monitor is also used to detect reactor coolant leakage into that portion of the reactor building closed loop cooling water (RBCLCW) system which supplies cooling water for the RHR and RWCU pump seal coolers. The return water from these coolers is combined with the return from the spent fuel heat exchangers and is monitored with a radiation monitor (see Figure 9.2-3e).

5.2.5.2 Leak Detection Instrumentation and Monitoring

5.2.5.2.1 Leak Detection Instrumentation and Monitoring Inside Primary Containment

Drywell Floor Drain Tank Measurement

The drywell leakage collected in the floor drain tank includes unidentified leakage from the CRDs, valve flanges, component cooling water, service water, air cooler drains, and any leakage not connected to the equipment drain sump. Abnormal leakage rates are detected and alarmed in the main control room. Leakage into the drywell floor drain system flows through a piping header that penetrates the containment wall and is then directed to the drywell floor drain tank located in the reactor building.

As shown on Figure 9.3-13, tank level and pump outflow rate measurements are processed to determine the unidentified leakage rate to the drywell floor drain tank.

The leakage rate is determined and updated on a chart recorder every 6 min. Leakage in excess of Technical Specification limits is alarmed.

The instrumentation and equipment associated with the fill rate monitoring of the drywell floor drain tank was procured with sufficient accuracy, sensitivity, and response time to ensure that a sensitivity of at least 1 gpm in 1 hr, as identified in Table 5.2-8, can be attained.

Drywell Equipment Drain Tank Measurement

The equipment drain tank collects only identified leakage. This tank receives piped drainage from pump seal leakoff and reactor vessel head flange vent drain.

Leakage into the drywell equipment drain system flows through a piping header, separate from the floor drain header which penetrates the primary containment wall, and is then directed to the drywell equipment drain cooler and drain tank in the reactor building.

As shown on Figure 9.3-13, tank level and pump outflow rate measurements are processed to determine the identified leakage rate to the drywell equipment drain tank.

The leakage rate is determined and updated on a chart recorder every 6 min. In addition, the average leakage rate for the previous 24 hr is also determined every 6 min and displayed on an indicator. Average 24-hr leakage in excess of Technical Specification limits is alarmed.

The instrumentation and equipment associated with the fill rate monitoring of the drywell equipment drain tank was procured with sufficient accuracy, sensitivity, and response time to ensure that a sensitivity of at least 1 gpm in 1 hr, as identified in Table 5.2-8, can be attained.

Temperature Measurement

The ambient temperature within the primary containment is monitored by four dual-element thermocouples located equally spaced in the vertical direction. An abnormal increase in primary containment temperature could indicate a leak within the primary containment. Ambient temperatures within the primary containment are recorded and alarmed on the leakage detection and isolation system panel in the control room.

Fission Product Monitoring

The primary containment monitoring system is used along with the temperature and pressure monitors described above to detect leaks in the nuclear system process barrier. The system continuously monitors the primary containment atmosphere for airborne

radioactivity (iodine, noble gases, and particulates). The sample is drawn from the primary containment. A sudden increase of activity, which may be attributed to steam or reactor water leakage, is annunciated in the main control room (Section 7.6).

Containment Pressure Measurement

The primary containment is at a slightly positive pressure during reactor operation and is monitored by pressure sensors. The pressure fluctuates slightly as a result of barometric pressure changes and leakage from containment. A pressure rise above the normally-indicated values indicates a possible leak within the primary containment. Pressure exceeding the preset values is annunciated in the main control room and safety action is automatically initiated.

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. A branch line penetrates the primary containment and terminates at a pressure transmitter. Leakage through the first seal is detected by the pressure transmitter and annunciated in the main control room. When pressure between the seals increases, the second seal then operates to contain the vessel pressure.

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 main control room. The leakage is piped to the equipment drain sump.

Safety/Relief Valves

Temperature sensors connected to a multipoint recorder are provided to detect SRV leakage during reactor operation. SRV temperature elements are mounted, using a thermowell, in the SRV discharge piping several feet downstream from the valve body. Temperature rise above the alarm setpoint is annunciated in the main control room. See the main steam system P&ID (Figure 10.1-3). Refer to Section 1.10, Item II.D.3, for discussion of acoustic monitors.

High Flow in Main Steam Lines (for Leaks Downstream of Flow Elements)

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

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 RCPB, is detected by using the same nuclear boiler system low reactor water level signals that alarm and isolate selected primary system isolation valves (Chapter 7).

RCIC/RHR Steam Line Flow (for Leaks Downstream of Flow Elements)

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

High Differential Pressure Between ECCS Injection Lines (for Leakage Internal to Reactor Vessel Only)

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

5.2.5.2.2 Leak Detection Instrumentation and Monitoring External to Primary Containment

Reactor Building Drain Flow Measurement

Instrumentation monitors and indicates the amount of unidentified leakage into the reactor building floor drainage system outside the primary containment. Background leakage is identified during startup tests. Abnormal leakage is alarmed in the main control room. Identified leakage within the reactor building outside the primary containment includes spent fuel pool, reactor cavity and internal storage pool liner leakage, refueling canal and cask storage area canal gate leakage, and inner and outer refueling seal leakage. Leakage from the liners is piped to sight gauge glasses to provide visual indication. A seal-welded plate has been installed to permanently seal the leak detection drain for the gates between the spent fuel pool and the cask storage pool. The leak detection piping remains routed to the level switch to permanently monitor this plate for leakage. Leakage from the gate drains and inner and outer refueling seals is collected in standpipes with level switches which provide alarms in the main control room. High water level in a leak detection line generates an alarm in the main control room. An alarm will also be generated if, after the high water level is reached, the pump operates for longer than a predetermined time span.

Visual and Audible Inspection

Accessible areas are inspected periodically and the flow indicators previously discussed are monitored regularly. Any instrument indication of abnormal leakage is investigated.

Differential Flow Measurement (RWCU System Only)

Because of its arrangement, the RWCU 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 main control room and a RWCU system isolation signal are initiated when high differential flow occurs between flow into the system and flow from the system, which indicates that a leak equal to the established leak rate limit may exist. Flow elements are installed on the RWCU system inlet, and on the outlets to the feedwater system and the main condenser/liquid radwaste connection.

Main Steam Line Area Temperature Monitors and RCIC Piping Routing Area Temperature Monitors

High temperatures in the MSL tunnel including lead enclosure area and RCIC pipe routing areas are detected by dual-element thermocouples. Some of the dual-element thermocouples are used for measuring 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. 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 pipe routing or tunnel temperature. High ambient or high differential temperature (main steam tunnel only) causes alarms in the main control room and provides signals to close the main steam and drain line isolation valves. High ambient temperature in the RCIC pipe routing areas will alarm in the main control room and provide signals to close the RCIC steam line isolation valves. A high main steam tunnel temperature or differential temperature alarm may also indicate leakage in the reactor feedwater line which passes through the main steam tunnel.

Twelve monitors in the space between the steam tunnel outlet and the high-pressure turbine inlet measure ambient temperature and trip the MSIVs on a high temperature signal.

Temperature Monitors in Equipment Areas

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

control room and provides trip signals for closure of isolation valves of the respective system in the monitored area.

Reactor Building Temperature Monitors

High temperature in the pipe chase areas is detected by dual-element thermocouples. The thermocouples are used for measuring ambient temperature in the vicinity of the RCIC, RWCU, and RHR lines. All temperature elements are located or shielded so as to be sensitive to air temperature and not to radiated heat. High ambient temperature will alarm in the main control room and provide signals to isolate the RCIC steam line, RWCU, and the RHR shutdown cooling path.

High temperature in the reactor building general areas is detected by temperature elements located on the building elevations where the RHR shutdown cooling piping is routed. High ambient temperature in the reactor building will alarm in the main control room and provide signals to isolate the RCIC steam line and the RHR shutdown cooling path.

The isolation trip setpoints for the areas monitored are above the maximum anticipated temperatures for these areas and are set above the maximum anticipated temperatures by the following amounts:

Reactor building general area	=	23.5°F
Reactor building pipe chase	=	11.5°F
RHR equipment area	=	11.5°F
RCIC piping area	=	11.5°F
RCIC pump room	=	11.5°F

The isolation trip setpoints are sufficiently low to limit the reactor building and auxiliary bay temperatures during a high-energy line break (HELB) or MELC to within the established environmental qualification temperature profiles for these areas while providing sufficient margin above expected ambient temperatures to prevent inadvertent isolation. In addition, since this leak detection design does not employ differential temperature monitoring, it is less susceptible to spurious isolations due to sharp drops in outside supply air temperature.

Intersystem Leakage Monitoring

In addition to the intersystem leakage instrumentation and monitoring discussed in this section and Section 5.2.5.2.1, refer to Section 11.5 for a discussion of the process radiation monitors used to detect leakage into the secondary sides of the RHR heat exchangers and the RWCU nonregenerative heat exchanger.

Monitoring Large Leaks External to the Primary Containment

The main steam high flow, RCIC/RHR steam high flow, and reactor vessel low water level monitoring discussed in Section 5.2.5.2.1

can also indicate large leaks from the reactor coolant piping external to the primary containment.

5.2.5.2.3 Summary

Tables 5.2-9 and 5.2-10 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 that are capable of detecting small leaks initiate an alarm in the main control room. The Operator can manually isolate a leaking system or take other appropriate action. A time delay is provided for the RWCU system differential flow to prevent normal system surges from isolating the system.

The LDS is a multidimensional system that is redundantly designed so that failure of any single element does not interfere with a required detection of leakage or isolation. In the four-division portion of the leak detection and isolation system, applied where inadvertent isolation could impair plant performance (e.g., MSIVs), 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 and two channels are required to trip for closure of MSIVs. 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 Indication in Main Control Room

Leak detection methods are discussed in Section 5.2.5.1. Instrumentation and controls for the LDS are in Section 7.6.1.3.

5.2.5.4 Limits for Reactor Coolant Leakage

5.2.5.4.1 Leakage Rate Limits

Reactor coolant leakage consists of identified and unidentified leakage that flows to the drywell floor drain and equipment drain tanks. The leakage rate limits specified in the Technical Specifications are well within the makeup capability of the RCIC system.

The equipment drain tank and the drywell floor drain sumps that collect all leakage are each pumped to the radwaste system by two 50-gpm pumps. Each pump normally alternates between lead and backup.

5.2.5.4.2 Identified Leakage Inside the Primary Containment

The recirculation pump seals and other seals (e.g., reactor head) in systems that are part of the RCPB and from which normal design-identified source leakage can be expected are provided with leakoff drains. Recirculation pumps are equipped with

double seals. Leakage from the primary recirculation pump seals is monitored for flow in the drain line and piped to the equipment drain tank (Section 5.4.1.3). Leakage from the MSL SRVs discharging to the suppression pool is monitored by temperature sensors that transmit signals to the main control room. Any temperature increase above the ambient temperature detected by these sensors indicates valve leakage.

5.2.5.5 Unidentified Leakage Inside the Primary Containment

5.2.5.5.1 Unidentified Leakage Rate

The unidentified leakage rate is the portion of the total leakage rate received in the primary containment sumps that is not identified as previously described. No significant compromise to the nuclear system process barrier exists if the barrier contains a crack that is less than the critical crack length. Even so, the unidentified leakage rate limit is kept 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 total unidentified leakage rate limit is established at 5 gpm to allow time for corrective action before the process barrier could be significantly compromised. This 5-gpm unidentified leakage rate is a small fraction of the calculated flow from a critical crack in a primary system pipe (Figure 5.2-8). Safety limits and safety limit settings are discussed in Technical Specifications.

5.2.5.5.2 This section has been deleted.

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[3]. This analysis relates to axially-oriented through-wall cracks.

<u>Critical Crack Length</u>

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:

$$L_c = \frac{15{,}000\ D}{\sigma_h} \tag{5.2-1}$$

Where:

L_c = Critical crack length, in

D = Mean pipe diameter, in

σ_h = Nominal hoop stress, psi

See data correlation on Figure 5.2-9.

Crack Opening Displacement

The theory of elasticity predicts a crack opening displacement of:

$$\omega = \frac{2L\sigma}{E} \tag{5.2-2}$$

Where:

L = Crack length, in

σ = Applied nominal stress, psi

E = Young's modulus

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

$$C = \sec\left(\frac{\pi}{2}\ \frac{\sigma}{\sigma_f}\right) \tag{5.2-3}$$

The crack opening area is given by:

$$A = C\,\frac{\pi}{4}\ \omega L = \frac{\pi L^2 \sigma}{2E}\ \sec\left(\frac{\pi}{2}\ \frac{\sigma}{\sigma_f}\right) \tag{5.2-4}$$

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

Leakage Flow Rate

The maximum flow rate for blowdown of saturated water at 1,000 psi is 55 lb/sec-sq in, and for saturated steam the rate is 14.6 lb/sec-sq in[4]. 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 sq in (saturated water)

A = 0.0475 sq in (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)	Ration 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 that precede unstable rupture. Judging from observed crack behavior in the GE and BMI experimental programs involving both circumferential and axial cracks, it is estimated that leak rates of hundreds of gpm precede crack instability. Measured crack opening displacements for the BMI experiments were in the range of 0.1 to 0.2 in at the time of incipient rupture, corresponding to leaks of the order of 1 sq in in size for plain carbon steel piping. For austenitic stainless steel piping, even larger leaks are expected to precede crack

instability, although there are insufficient data to permit quantitative prediction.

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

Figure 5.2-8 shows general relationships between crack length, leak rate, stress, and line size, using the mathematical model described previously. The asterisks denote conditions for which the crack opening displacement is 0.1 in, at which time instability is imminent as noted previously under Leakage Flow Rate. This provides a realistic estimate of the leak rate to be expected from a crack of critical size. In every case, the leak rate from a crack of critical size is significantly greater than the 5-gpm criterion. If either the total or unidentified leak rate limits are exceeded, an orderly shutdown can be initiated and the reactor can be placed in a cold shutdown condition within 24 hr.

5.2.5.5.4 Margins of Safety

The margins of safety for a detectable flaw to reach critical size are discussed in Section 5.2.5.5.3. Figure 5.2-8 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 and reactor building bays, as shown in Tables 5.2-9 and 5.2-10. The instrumentation is designed so it can be set to provide alarms at established leakage rate limits and isolate the affected system, if necessary. The alarm points are determined analytically or based on measurements of appropriate parameters made during startup and preoperational tests.

The unidentified leakage rate limit is based, with an adequate margin for contingencies, on the crack size large enough to propagate rapidly. The established limit is sufficiently low 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 LDS can 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 LDS. The ability of the LDS to differentiate between identified and unidentified leakage is discussed in Sections 5.2.5.4 and 5.2.5.5.

5.2.5.7 Safety Interfaces

The balance-of-plant to GE nuclear steam supply system (NSSS) safety interfaces for the LDS are the signals from the monitored balance-of-plant equipment and systems that are part of the nuclear system process barrier, and associated wiring and cable lying outside the NSSS equipment.

5.2.5.8 Testing and Calibration

Provisions for preoperational testing of the LDS are covered in Chapter 14. Calibration is discussed in Technical Specifications.

5.2.5.9 Regulatory Guide Compliance

The detection of leakage through the RCPB, described in the preceding sections, is assessed to be in compliance with RG 1.45. Details of compliance are discussed in the following paragraphs. Leakage is separated into identified and unidentified categories and each is independently monitored, thus meeting position c.1 of RG 1.45.

Leakage from unidentified sources inside the primary containment is collected into the floor drain sump and monitored with an accuracy better than 1 gpm, thus meeting position c.2.

By monitoring (1) floor and equipment drain sump fillup and pumpout rates, (2) airborne particulates, and (3) airborne gaseous radiation rate, position c.3 is satisfied. The containment atmosphere temperature and pressure monitors are secondary methods used to detect gross leakage.

Radiation monitoring of cooling water from the RHR heat exchangers and RWCU nonregenerative heat exchangers satisfies position c.4. For system detail see Section 11.5.

The floor drain sump monitoring, air particulates monitoring, and gaseous radiation monitoring are designed to detect leakage rates given in Table 5.2-8, thus meeting position c.5.

All leakage detection systems are designed to be capable of performing their functions following seismic events that do not

require plant shutdown (OBE). Seismic qualification (SSE) is performed only for safety-related portions and for the primary containment RMS. Thus, position c.6 is met. It must be noted, however, that administrative procedures can be utilized to verify operability following a seismic event if required.

Leak detection indicators and alarms are provided in the main control room as detailed in Tables 5.2-9 and 5.2-10. This satisfies position c.7.

Procedures and graphs are provided to plant Operators for converting the various indicators to a common leakage equivalent to satisfy the remainder of position c.7.

Leak detection complies with IEEE-338. All active components associated with isolation signals can be tested during plant operation. Indication is provided in the main control room that a logic channel is tripped.

The leakage detection systems are equipped with provisions to permit testing for operability and calibration during the 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 (i.e., area temperature monitoring).

3. Operability checked by comparing one method versus another (i.e., tank fillup versus pumpout and particulate monitoring).

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

These satisfy position c.8.

Plant Technical Specifications comply with position c.9 by specifying limiting conditions for identified and unidentified leakage and by addressing the availability of various types of instruments to assure adequate coverage.

<u>Regulatory Guide 1.22 Assessment</u>

The proper operation of the LDS sensors and logic is verified during the preoperational tests and during plant operation. Each temperature switch (both ambient and differential types) that provides isolation signals is connected to one element of a dual thermocouple. A light illuminates when the temperature exceeds the setpoint. Verification of the thermocouple input is accomplished by comparing the reading from the trip channel with the recorder channel which is connected to the other element of the dual thermocouple. The trip logics are tested by applying a simulated trip signal from an external source to the LDS channel.

Keylock test switches are used to prevent the isolation signal from performing its isolating function.

5.2.6 References

1. General Electric Standard Application for Reactor Fuel - United States Supplement, NEDE-24011-P-A-US, (latest approved revision).

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

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

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

5. Crane Technical Paper No. 410, 1974 Edition.

6. Pressure Relieving Device Certifications, National Board of Boiler and Pressure Vessel Inspectors, 1979 Edition.

7. General Electric Design Report 22A7122, Overpressure Protection Report, Revision 2.

8. Licensing Topical Report, Power Uprate Licensing Evaluation for Nine Mile Point Nuclear Power Station, Unit 2, NEDC-31994P, Revision 1, May 1993.

9. Nine Mile Point Unit 2 Power Uprate Vessel Overpressure Transient Analysis, GE-NE-187-05-0291, May 1991.

TABLE 5.2-1
(Sheet 1 of 11)
APPLICABLE CODE CASES[1]

Code Case Number/ Revision	Applicable Equipment	Title
Class 1 Applications		
1141-01	RPV	Foreign Produced Steel
1332-6	RPV and primary containment penetration forgings	Requirements for Steel Forgings, ASME Sections III and VIII, Division 2
1335-10	Recirculating water system flow control valves	Requirements for Bolting Materials, Section III
1361-2	CRD	Socket Welds
	Code Case 1361-2 is used for four weld joints in the CRD. RG 1.84 indicates that this code case is acceptable if used in conjunction with Section III, paragraph NB-3356, Fillet Welds. The stress criteria of one-half the stress limits for primary and secondary stresses in the welds are required by paragraph NB-3356. The CRD stress analysis shows that these criteria are satisfied.	
1388-2	Primary containment liner	Requirements for Stainless Steel Hardening, Section III
1516	ASME III valves	Welding of Non-Integral Seals in Valves for Section III Applications
1539	ASME III valves	Metal Bellows and Metal Diaphragm Stem-Sealed Valves, Section III, Class 1, 2, and 3
1557-1	RPV	Steel Product Refined by Secondary Remelting
1562	RPV	Qualification of Forming and Bending Processes for Class 1, 2, and 3 Components, Section III
1567 (N-38)	HPCS valves and primary containment liner, field welding of ASME III components/structures/pipe supports; special check valves; emergency diesel generator (Div. I and II) starting air skid, special service control valves	Testing Lots of Carbon and Low-Alloy Steel Covered Electrodes, Section III
1568	Primary containment liner	Testing of Flux Cored and Fabricated Carbon and Low Alloy Steel Welding Electrodes, Section III
1572	RPV	Fracture Toughness, Section III, Class 1 Components

TABLE 5.2-1 (Cont'd.)
(Sheet 2 of 11)

Code Case Number/ Revision	Applicable Equipment	Title
1588	Feedwater nozzle safe ends	Electro-Etching of Section III Code Symbols
1590	Primary containment liner	Chemical Analysis Variations, Section III Construction
1620	RPV	Stress Category for Partial Penetration Welded Penetrations, Section III, Class I Construction
1621-1 1621-2 (N-62)	ASME III valves	Internal and External Valve Items Section III, Class 1, 2, and 3 Line Valves
1634-2	Emergency diesel generator (Division I and II) starting air skid	Use of SB-359 for Section III, Division I, Class 3 Construction
1637	HPCS valves	Compliance with NA 3700
1644-2	ASME III Material Class 1, 2, 3 pipe supports	Additional Material for Component Supports, Section III, Subsection NF, Class 1, 2, 3 and MC Construction
	Materials with an ultimate tensile strength above 170 ksi were not used. This complies with the requirements of RG 1.85	
1644-3	ASME III Material Class 1, 2, 3 pipe supports	Additional Material for Component Supports, Section III, Subsection NF, Class 1, 2, 3 and MC Construction
	Materials with an ultimate tensile strength above 170 ksi were not used. This complies with the requirements of RG 1.85.	
1644-5	ASME III Material Class 1, 2, 3 pipe supports	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
	Materials with an ultimate tensile strength above 170 ksi were not used. This complies with the requirements of RG 1.85	
1644-6	Recirculation pump restraints and Category I unit cooler supports, Class 1, 2, 3 pipe supports	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
	Materials with an ultimate tensile strength above 170 ksi were not used. This complies with the requirements of RG 1.85	
1644-7 (N-71-7)	ASME III Material Class 1, 2, and 3 pipe supports, and unit cooler supports	Additional Materials for Component Supports, Section III, Division 1, Subsection NF, Class 1, 2, 3, and MC Construction
	Materials with an ultimate tensile strength above 170 ksi were not used. This complies with the requirements of RG 1.85	

TABLE 5.2-1 (Cont'd.)
(Sheet 3 of 11)

Code Case Number/ Revision	Applicable Equipment	Title
1644-9 (N-71-9)	ASME III additional material for Subsection NF Class 1, 2, 3 and MC component supports fabricated by welding	Additional Materials for Component Supports Fabricated by Welding, Section III, Division 1, Subsection NF, Class 1, 2, 3, and MC Component Supports
	The required exposure times for low hydrogen electrodes were adhered to except for the exposure times were increased slightly for some site welding. For electrode E70XX, the exposure time was increased from 4 to 5 hr, and for electrode E80XX, from 2 to 4 hr. In these cases, all covered electrodes continued to be issued in heated portable rod ovens. The increased exposure times are based on industry usage data and test data to meet the intent of RG 1.85 and Code Case 1644-9 requirements.	
N-71-16	Class 1, 2, 3 and MC component supports	Additional Material for Subsection NF, Classes 1, 2, 3, and MC Component Supports Fabricated by Welding, Section III, Division 1
	Code Case N-71-16 is subject to the conditions set forth in RG 1.85 Revision 31.	
1651 (N-74)	Safety Class 1, 2, and 3 pipe supports and unit cooler supports	Interim Requirements for Certification of Component Supports, Section III, Subsection NF
1672	ASME III valves	Nuclear Valves for Section III, Division 1, Class 1, 2, and 3 Construction
1677	ASME III valves	Clarification of Flange Design Loads, Section III, Class 1, 2, and 3
1682-1	Class 1, 2, and 3 piping, pipe supports and mechanical equipment supports	Alternate Rules for Material Manufacturers and Suppliers, Section III, Subarticle NA-3700
1683-1	Class 1, 2, and 3 pipe supports and mechanical equipment	Bolt Holes for Section III, Division 1, Class 1, 2, 3 and MC Component Supports
1685 (N-85)	Class 1, 2, and 3 pipe supports	Furnace Brazing, Section III, Class 1, 2, 3 and MC Construction
1686 (N-86)	Class 1, 2, and 3 pipe supports	Furnace Brazing, Section III, Subsection NF Component Supports
1690 (N-88)	Class 1, 2, and 3 pipe supports	Stock Materials for Section III Construction, Section III, Division 1

TABLE 5.2-1 (Cont'd.)
(Sheet 4 of 11)

Code Case Number/ Revision	Applicable Equipment	Title
1698	Primary containment liner	Waiver of the Ultrasonic Transfer Method, Sections III, V, and VIII, Division 1
	The use of Code Case 1698 is subject to the following conditions in addition to those identified in the code case: The material from which the basic calibration blocks are fabricated is of the same product form, alloy, and heat treatment as the material being fabricated.	
1702-1	ASME III valves	Flanged Valves Larger than 24 in for Section III, Division 1, Class 1, 2, and 3 Construction
1711 (N-100)	ASME III safety and relief valves	Pressure Relief Valve Design Rules, Section III, Division 1, Class 1, 2 and 3
	Code Case 1711 was acceptable subject to the following conditions, in addition to those conditions specified in the code case: 1. If stress limits are used in excess of those specified for the upset operating condition, it should be demonstrated how the pressure relief function is ensured. Refer to paragraph 3.1, Section I, of the case for Class 1 and paragraph 3.2, Section II, of the case for Class 2 and 3 pressure relief valves. 2. If Case 1660 is to be used in conjunction with this case, it should be stated that the stress limits of Code Case 1660 supersede those of paragraph 3.2(b), Section I, of Code Case 1711. Functional assurance of Item 1 above is required in all situations. The above two conditions have been evaluated and are not applicable to the subject relief valves.	
1724 (N-108)	Class 1, 2, and 3 pipe supports	Deviation from Specified Silicon Ranges in ASME Materials Specifications, Section III, Division 1, and VIII, Divisions 1 and 2
1729 (N-111)	Class 1, 2 and 3 pipe supports and unit cooler supports	Minimum Edge Distance Bolting for Section III, Division 1, Class 1, 2, 3 and MC Construction of Component Supports
1734 (N-116)	Class 1, 2, and 3 pipe supports (spring hanger spring encapsulation cans by ITT Grinnell)	Weld Design for Use for Section III, Division 1, Class 1, 2, and 3 MC Construction of Component Supports
1739-4 (N-119-4)	ASME III pumps	Pump Internal Items, Section III, Division 1, Class 1, 2, and 3
1745	Class 1 pipe supports	Stress Indices for Integral Structural Attachments, Class 1, Section III, Division 1

TABLE 5.2-1 (Cont'd.)
(Sheet 5 of 11)

Code Case Number/ Revision	Applicable Equipment	Title
1773 (N-141)	ASME III valves	Use of Other Product Forms of Materials for Valves, Section III, Division 1
1810	Spent fuel cooling heat exchangers; emergency diesel generator (Division II) starting air skid	Testing Lots of Carbon Steel Solid, Bare Welding Electrode or Wire, Section III, Division 1, Class 1, 2, 3, and MC Component Supports
1820	Feedwater nozzle safe ends	Alternative Ultrasonic Examination Technique, Section III, Division 1
N-180	Class 1 pipe supports	Examination of Springs for Class 1 Component Standard Supports, Section III, Division 1
N-181	ASME III valves	Steel Castings Refined by the Argon Decarburization Process, Section III, Division 1, Construction
N-225	Class 1, 2, and 3 pipe supports	Certification and Identification of Material for Component Supports, Section III, Division 1
N-242	Service water, RHR, and LPCI piping	Material Certification, Section III, Division 1, Class 1, 2, 3, MC and CS Construction
	All paragraphs of the code case were applied to the above-listed piping materials.	
	Emergency diesel generator (Divisions 1 and 2) jacket water standpipe	
	Paragraphs 1 through 4 of the code case were applied to the above-listed standpipe material.	
N-242-1	Shop-fabricated piping, field-purchased pipe and fittings, pipe support material and instrument air system vessels, and safety relief valves	Material Certification Section III, Division 1, Class 1, 2, and 3, MC and CS Construction
	All paragraphs of the code case are applied to the above-listed shop-fabricated piping and valves. For pipe support material and field-purchased pipe and fittings, paragraphs 5.0 through 6.0 are applicable for material procured after the effective date of the code case, and paragraphs 2.0 through 4.0 apply to material procured prior to that date.	

TABLE 5.2-1 (Cont'd.)
(Sheet 6 of 11)

Code Case Number/ Revision	Applicable Equipment	Title
N-242-1 (cont'd.)	Flexible ball joints for main steam safety valve, vent, and drain system; and instrument air system accumulator tanks	
	Paragraphs 5 and 6 of the code case were applied to the above-listed flexible ball joints and air accumulator tanks.	
	Accumulator tanks	
	Paragraphs 1 through 4 of the code case were applied to the above-listed tanks.	
N-247	Class 1, 2, and 3 component standard supports	Certified Design Report Summary for Component Standard Supports
N-249-11	Class 1, 2, 3 and MC component supports	Additional Material for Subsection NF, Class 1, 2, 3, and MC Component Supports Fabricated Without Welding, Section III, Division 1
	Code Case N-249-11 is subject to the conditions set forth in RG 1.85 Revision 30.	
N-249-13	Class 1, 2, 3 and MC component supports	Additional Material for Subsection NF, Classes 1, 2, 3, and MC Component Supports Fabricated Without Welding, Section III, Division 1
	Code Case N-249-13 is subject to the conditions set forth in RG 1.85 Revision 31.	
N-272	Class 1, 2, and 3 piping systems	Compiling Data Report Forms, Section III, Division I

TABLE 5.2-1 (Cont'd.)
(Sheet 7 of 11)

Code Case Number/ Revision	Applicable Equipment	Title
N-275	ASME III welds	Repair of Welds for NB/NC/ND/NE/NF/NG Section III, Division 1 Construction
	Code Case N-275 is subject to the following condition in addition to those conditions specified in the code case: Use of the code case is applicable only when the removal of an indication requires that the full weld thickness be removed and, in addition, the backside of the weld assembly joint is not accessible for the removal of examination material. If an indication is removed and weld-metal layers still remain, it is not acceptable to gouge through the wall in order to qualify for use of the code case. Instead, examination of the cavity is required when such an indication has been removed.	
N-316	Class 1, 2, and 3 socket-welded fittings in piping	Alternate Rules for Fillet Weld Dimensions for Socket Welded Fittings
N-341	ASME III items	Certification of Level III NDE Examiner, Section III, Division 1
N-377	ASME III component supports	Effective Throat Thickness of Partial Penetration Groove Welds Section III, Division 1, Classes 1, 2, and 3
N-411	Class 1, 2, and 3 pipe	Alternative Damping Values for Seismic Analysis of Piping Section, Division 1, Class 1, 2, and 3
N-413	Pipe supports	Minimum Size of Fillet Welds for Linear-Type Supports, Section III, Division 1, Subsection NF
	In accordance with Code Case N-413, allowable weld stress of 21 ksi for base metals with a tensile strength range of 58 ksi to 70 ksi will be used.	
Non-Class 1 Application		
1516-2	Class 2 and 3 control valves	Welding of Seats or Minor Interval Permanent Attachments in Valves for Section III Applications
1541-3	Suppression pool liner drain lines	Hydrostatic Testing of Embedded Class 2 and 3 Piping for Section III, Division 1 Construction
1590	Class 2, primary containment piping penetration forging Z19	Chemical Analysis Variations, Section III Construction
1606-1	Class 2 and 3 piping	Stress Criteria, Section III, Class 2 and 3 Piping Subject to Upset, Emergency, and Faulted Operating Conditions

TABLE 5.2-1 (Cont'd.)
(Sheet 8 of 11)

Code Case Number/ Revision	Applicable Equipment	Title
1644-9 (N-71-9)	CRD piping supports, structural tubing	Material for Component Supports, Section III, Subsection NF, Class 1, 2, 3, and MC Component Supports
	The required exposure times for low-hydrogen electrodes were adhered to except for some site welding. For electrode E70XX, the exposure time was increased from 4 to 5 hr, and for electrode E80XX, from 2 to 4 hr. In these cases, all covered electrodes continued to be issued in heated portable rod ovens. The increased exposure times are based on industry usage data and test data to meet the intent of RG 1.85 and Code Case 1644-9 requirements.	
1677	Class 2 pumps	Clarification of Flange Design Loads, Section III, Class 1, 2, 3
1695-1	Cooling coil for containment building ac units	Brazing, Section III, Division 1, Class 3
1728	Class 2 and 3 pipe supports	Steel Structural Shapes and Small Material Products for Component Supports, Section III, Division 1 Construction
1774 1774-1	Class 2 and 3 valves	Minimum Wall Thickness for Class 2 and 3 Valves, Section III, Division 1
1812 (N-174)	Instrument tubing	Size of Fillet Welds for Socket Welding of Pipe, Section III, Division 1
N-192	Class 2 and 3 piping, and instrument tubing to process piping connections	Use of Flexible Hose for Section III, Division 1, Class 1, 2, and 3 Construction
	Code Case N-192 was acceptable subject to the following conditions in addition to those conditions specified in the code case. The applicant should indicate system application, design and operating pressure, and pressure-temperature rating of the flexible hose. Data to demonstrate compliance of the flexible hose with NC/ND-3649, particularly NC/ND-3649.4(c), are required to be furnished with the application. Technical data which demonstrates compliance with the requirements will be submitted under separate cover.	
N-192-2	Class 2 and 3 piping, and instrument tubing to process piping connections	Use of Braided Flexible Connectors, Section III, Division 1, Class 2 and 3
	Code Case N-192-2 is acceptable subject to the following conditions in addition to those conditions specified in the code case. The applicant should indicate system application, design and operating pressure, and pressure-temperature rating of the flexible hose. Data to demonstrate compliance of the flex hose with NC/ND-649, particularly NC/ND-3649.4(e), are required to be furnished with the application. Technical data which demonstrates compliance with the requirements will be submitted under separate cover.	
N-224	Class 2 and 3 pipe supports	Use of ASTM A500 Grade B and ASTM A501 Structural Tubing for Welded Attachments for Section III, Class 2 and 3 Construction
N-224-1	Class 2 and 3 pipe supports, duct supports, and conduit supports	Use of ASTM A500 Grade B and ASTM A501 Structural Tubing for Welded Attachments for Section III, Class 2, 3, and MC Construction

TABLE 5.2-1 (Cont'd.)
(Sheet 9 of 11)

<table>
<tr><th>Code Case Number/ Revision</th><th>Applicable Equipment</th><th>Title</th></tr>
<tr><td>N-237</td><td>Class 2 and 3 piping</td><td>Hydrostatic Testing of Internal Piping, Section III, Revision 1</td></tr>
<tr><td>N-240</td><td>Class 2 and 3 piping</td><td>Hydrostatic Testing of Open-Ended Piping, Section III, Division 1</td></tr>
<tr><td>N-241</td><td>Class 2 and 3 piping</td><td>Hydrostatic Testing of Piping, Section III, Division 1</td></tr>
<tr><td rowspan="2">N-242-1</td><td>Shop-fabricated piping purchased pipe and fittings, pipe support material and instrument air system vessels</td><td>Material Certification, Section III, Division 1, Class 1, 2, 3, MC, and CS Construction</td></tr>
<tr><td colspan="2">All paragraphs of the code case are applied to the above-listed shop-fabricated piping. For pipe support material and field purchase pipe and fittings, paragraphs 5.0 through 6.0 are applicable for material procured after the effective date of the code case, and paragraphs 2.0 through 4.0 apply to material procured prior to that date.</td></tr>
<tr><td rowspan="2">N-249-11</td><td>Class 1, 2, 3 and MC component supports</td><td>Additional Material for Subsection NF, Class 1, 2, 3, and MC Component Supports Fabricated Without Welding, Section III, Division 1</td></tr>
<tr><td colspan="2">Code Case N-249-11 is subject to the conditions set forth in RG 1.85 Revision 30.</td></tr>
<tr><td>N-282</td><td>ASME III valves</td><td>Nameplates for Valves, Section III, Division I, Class 1, 2, and 3 Construction</td></tr>
<tr><td rowspan="2">N-318</td><td>Class 2 and 3 pipe supports</td><td>Procedure for Evaluation of the Design of Rectangular Cross Section Attachments, Class 2 or 3 Piping, Section III, Division 1</td></tr>
<tr><td colspan="2">In accordance with RG 1.84, the following information is provided.
<table>
<tr><th>No.</th><th>System Identification Line No.</th><th>Location in System Support No.</th><th>Attachment Method (Weld Type)</th></tr>
<tr><td>1</td><td>Service Water
2SWP-018-15-3</td><td>2-SWP-PSSH230A3</td><td>Fillet - All around</td></tr>
<tr><td>2</td><td>Service Water
2SWP-006-74-3</td><td>2-SWP-PSR999A3</td><td>Fillet - Three sides</td></tr>
<tr><td>3</td><td>Service Water
2SWP-006-78-3</td><td>2-SWP-PSR1000A3</td><td>Fillet - Three sides</td></tr>
<tr><td>4</td><td>Service Water
2SWP-006-98-3</td><td>2-SWP-PSR989A3</td><td>Full penetration</td></tr>
</table></td></tr>
</table>

TABLE 5.2-1 (Cont'd.)
(Sheet 10 of 11)

<table>
<tr><th>Code Case Number/ Revision</th><th>Applicable Equipment</th><th>Title</th></tr>
<tr><td>N-318
(cont'd.)</td><td colspan="2">
<table>
<tr><th>No.</th><th>System Identification Line No.</th><th>Location in System Support No.</th><th>Attachment Method (Weld Type)</th></tr>
<tr><td>5</td><td>Service Water
2SWP-006-95-3</td><td>2-SWP-PSR990A3</td><td>Full penetration</td></tr>
<tr><td>6</td><td>Service Water
2SWP-006-107-3</td><td>2-SWP-PSR493A3</td><td>Fillet - Three sides</td></tr>
<tr><td>7</td><td>Service Water
2SWP-006-107-3</td><td>2-SWP-PSR478A3</td><td>Fillet - Three sides</td></tr>
<tr><td>8</td><td>Service Water
2SWP-006-104-3</td><td>2-SWP-PSR477A3</td><td>Fillet - Three sides</td></tr>
<tr><td>9</td><td>Service Water
2SWP-030-422-3</td><td>2-SWP-PSR762A3</td><td>Fillet - Three sides</td></tr>
<tr><td>10</td><td>Service Water
2SWP-030-423-3</td><td>2-SWP-PSR719A3</td><td>Fillet - Three sides</td></tr>
</table>
</td></tr>
<tr><td rowspan="2">N-318-1</td><td>Class 2 and 3 pipe supports</td><td>Procedure for the Evaluation of the Design of Rectangular Attachments on Class 2 or 3 Piping, Section III, Division 1</td></tr>
<tr><td colspan="2">In accordance with RG 1.84, the following information is provided:
<table>
<tr><th>No.</th><th>System Identification Line No.</th><th>Location in System Support No.</th><th>Attachment Method (Weld Type)</th></tr>
<tr><td>1</td><td>Service Water
2SWP-006-104-3</td><td>2-SWP-PSR491A3</td><td>Fillet - Two sides</td></tr>
<tr><td>2</td><td>Service Water
2SWP-003-901-3</td><td>2-SWP-PSR1101A3</td><td>Fillet - Two sides</td></tr>
</table>
</td></tr>
<tr><td>N-339</td><td>CRD penetration sleeve guides, conduit supports attached to suppression chamber liner</td><td>Examination of Ends of Fillet Welds, Section III, Division 1, Classes 1, 2, and MC</td></tr>
<tr><td>N-369</td><td>Class 2 and 3, and MC construction of bellows</td><td>Resistance Welding of Bellows, Section III, Division 1</td></tr>
<tr><td>N-377</td><td>ASME III component supports</td><td>Effective Throat Thickness of Partial Penetration Groove Welds, Section III, Division 1, Classes 1, 2, and 3</td></tr>
</table>

TABLE 5.2-1 (Cont'd.)
(Sheet 11 of 11)

Code Case Number/ Revision	Applicable Equipment	Title
N-392	Class 2 and 3 pipe supports	Procedure for Evaluation of the Design of Hollow Circular Cross Section Welded Attachments on Classes 2 and 3 Piping, Section III, Division 1
N-413	Pipe supports	Minimum Size of Fillet Welds for Linear Type Supports, Section III, Division 1, Subsection NF
	In accordance with Code Case N-413, allowable weld stress of 21 ksi for base metals with a tensile strength range of 58 ksi to 70 ksi will be used.	
N-62-6	ASME III valves	Internal and External Valve Items, Section III, Division 1, Class 1, 2, and 3
N-207	BWR/6 CRD	Use of Modified SA-479 Type XM-19 for Section III, Division 1, Class 1, 2, 3 or CS Construction

(1) For NSSS-supplied equipment only, code cases applicable to reactor coolant pressure boundary components (but not their supports) are identified.

TABLE 5.2-2
(Sheet 1 of 1)
NUCLEAR SYSTEM SAFETY/RELIEF SETPOINTS*

No. of Valves	Spring Set Pressure (psig)	ASME Rated Capacity @ 103% Spring Set Pressure (lb/hr each)	Pressure Setpoint for Power-Actuated Mode (psig)
2	1,165	895,000	1,103
4	1,175	902,000	1,113
4	1,185	910,000	1,123
4	1,195	917,000	1,133
4	1,205	925,000	1,143

NOTE: Seven of the SRVs are used for the automatic depressurization function.

* Cycle-specific values are covered in Appendix A, Table A.5.2-1.

TABLE 5.2-3
(Sheet 1 of 1)
SYSTEMS THAT MAY INITIATE DURING OVERPRESSURE EVENT

Cycle-specific information is covered in Appendix A Table A.5.2-2

System	Initiating/Trip Signal(s)
RPS	Reactor trips with high flux
RCIC	ON with reactor low water level
	OFF with reactor high water level
HPCS	ON with reactor low water level
	ON with high drywell pressure
	OFF with reactor high water level
Recirculation	OFF with reactor low water level
	OFF with reactor high pressure
RWCU	OFF with reactor low water level

TABLE 5.2-4
(Sheet 1 of 1)
SEQUENCE OF EVENTS FOR FIGURE 5.2-1*

(Cycle-specific values are covered in Appendix A, Section A.5.2.2.2.2)

Time (sec)	Event
0	Closure of all MSIVs was initiated.
0.3	MSIVs reached 85 percent open. Failure of direct position scram was assumed.
1.8	Neutron flux reached APRM flux setpoint and initiated reactor scram.
2.2	Sensed reactor dome pressure reached setpoint of RPT.
2.4	Recirculation pump/motor initiated to coastdown.
3.1	Steam line pressure reached Group 1 SRVs pressure setpoint (spring-action safety mode), while power-actuated relief mode was ignored. (See Section 5.2.2.2.2.)
3.4	SRVs all opened due to high pressure.
3.8	Vessel bottom pressure reached its peak value.

* Based on a representative equilibrium fuel cycle.

TABLE 5.2-5
(Sheet 1 of 6)
REACTOR COOLANT PRESSURE BOUNDARY MATERIALS

Component	Form	Material	Specification (ASTM/ASME)
Reactor Pressure Vessel			
Reactor vessel heads, shells	Rolled plate or forgings Welds	Low-alloy steel Low-alloy steel	SA-533 Gr. B Cl. 1 or SA-508 Cl. 2 SFA5.5
Closure flange	Forged ring Welds	Low-alloy steel Low-alloy steel	SA-508 Cl. 2 SFA5.5
Nozzles	Forged shapes Welds	Low-alloy steel Low-alloy steel	SA-508 Cl. 2 SFA5.5
Nozzle safe ends	Forgings or plates	Stainless steel	SA-182, F304, or F316 SA-336, F8 or F8M SA-240, 304 or 316
	Welds	Stainless steel	SFA5.9 Tp. 308L or 316L SFA5.4 Tp. 308L or 316L
Nozzle safe ends	Forgings Welds	Ni-Cr-Fe Ni-Cr-Fe	SB-166 or SB-167 SFA5.14 Tp. ER NiCr-3 or SFA5.11 Tp. ENi CrFe-3
Nozzle safe ends	Forgings Welds	Carbon steel Carbon steel	SA-508 Cl. 1 SFA5.1, SFA5.18 GPA or SFA5.17 F70
Nozzle to safe end weld	Weld overlay repair	Alloy 52	Code Case 2142/2143 UNS N06052/UNS W86152
Cladding	Weld overlay	Austenitic stainless steel	N/A
Main Steam Safety Relief Valve			
Body	Cast	Carbon steel	SA-352 LCB
Seat	Forging	Carbon steel	SA-350 LF2
Disc	Cast	Stainless steel	SA-351 CF3A

TABLE 5.2-5
(Sheet 2 of 6)

Component	Form	Material	Specification (ASTM/ASME)
Main Steam Flow Element			
Instrument nozzle	Forging	Carbon steel	SA-105
Upstream casting	Cast	Stainless steel	SA-351 Tp. CF8
Downstream casting	Cast	Stainless steel	SA-216 Gr. WCB

TABLE 5.2-5
(Sheet 3 of 6)

Component	Form	Material	Specification (ASTM/ASME)
Main Steam Piping			
Pipe	Seamless Seamless	Carbon steel Chrome-moly	SA-106 Gr. B, Gr. C SA-335 P22
Pipe (penetration)	Seamless	Carbon steel	SA-106 Gr. B
Contour nozzle (26" x 6" I.D. - 1,500 lb)	Forged	Carbon steel	SA-105
Tongue flange (4" - 900 lb)	Forged	Carbon steel	SA-105
Elbow	Forged fittings Welded or seamless fittings	Chrome-moly Carbon steel	SA-182 Gr. F22 SA-234 Gr. WPC or Gr. WPCW SA-234 Gr. WPB
Sockolet (1" - 3,000 lb, 2", 3/4" - 6,000 lb)	Forged	Carbon steel	SA-105
Head fitting - groove	Forged	Carbon steel	SA-508 Cl. 1
Concentric reducer	Welded fitting	Carbon steel	SA-234, Gr. WPB
Flange (8" - 1,500 lb)	Forged	Carbon steel	SA-105
Sweepolet (26" x 10")	Forged	Carbon steel	SA-105
Main Steam Isolation Valve			
Valve body	Cast	Carbon steel	SA-216 Gr. WCC
Bonnet	Forged	Carbon steel	SA-350 Gr. LF2
Stem	Forged	Stainless steel	A-182 Gr. F6A Cl. 3 or SA-564, Type 630
Disk	Forged	Carbon steel	SA-350 Gr. LF2
Stem disk	Forged	Carbon steel	SA-350 Gr. LF2
Bonnet studs	Forged	Alloy steel	SA-540 Gr. B23 Cl. 4
Hex nuts	Forged	Alloy steel	SA-194 Gr. 7

TABLE 5.2-5 (Cont'd.)
(Sheet 4 of 6)

Component	Form	Material	Specification (ASTM/ASME)
Main Steam Isolation Valve (cont'd.)			
Leakoff plug	Forged	Carbon steel	SA-105
Recirculation Gate Valve			
Body	Cast	Stainless steel	SA-351 CF8M
Bonnet	Cast	Stainless steel	SA-351 CF8M
Disc	Cast	Stainless steel	SA-351 CF3A
Stem	Rod	Stainless steel	SA-564 Tp. 630 Cond. 1150
Studs	Bolting	Alloy steel	SA-193 Gr. B7
Nuts	Bolting	Alloy steel	SA-194 Gr. 7
Recirculation Flow Control Valve			
Body	Cast	Stainless steel	SA-351 Gr. CF8M, 316
Housing	Cast	Stainless steel	SA-351 Gr. CF8M, 316
Bonnet	Cast	Stainless steel	SA-351 Gr. CF8M, 316
Covers	Cast	Stainless steel	SA-351 CF8M, 316
Recirculation Pump			
Pump case assembly	Cast	Stainless steel	SA-351 CF8M
Stud case to stuffing box	Bolting	Alloy steel	SA-540 Gr. B23 Cl. 5
Stud nut (3 1/4-8N)	Bolting	Alloy steel	SA-194 Gr. B7
Stuffing box	Cast	Stainless steel	SA-351 CF8M
Nozzle - 3/4"	Forging	Stainless steel	SA-182 Tp. F304/F316
Flange nozzle - 1"	Forging	Stainless steel	SA-182 Tp. F304/F316

TABLE 5.2-5
(Sheet 5 of 6)

Component	Form	Material	Specification (ASTM/ASME)
Recirculation Pump (cont'd.)			
Thrust ring	Forging	Stainless steel	SA-182 Tp. F316 with flash chrome plated
Pump flange	Forging	Carbon steel	SA-350 Gr. LF2
Seal holder	Forging	Stainless steel	SA-351 Gr. CF8M
Elbow	Plate	Stainless steel	SA-240 Tp. 304/316
Clip - outer	Bar	Stainless steel	A276 Tp. 304/316 Cond A
Clip - inner	Bar	Stainless steel	A276 Tp. 304/316 Cond A
Clip - tube end	Bar	Stainless steel	A276 Tp. 304/316 Cond A
Upper seal gland	Forging	Stainless steel	SA-351 Gr. CF8M
Grayloc clamp	Cast	Stainless steel	SA-351 Gr. CF8M
Pipe - 1" Sch 80	Plate	Stainless steel	SA-312 Gr. Tp. 304/316
Grayloc hub - 1"	Forging	Stainless steel	SA-182 Tp. F316
Tee - 1" pipe 3000#	Forging	Stainless steel	SA-182 Tp. F316
Thermowell	Forging	Stainless steel	SA-182 Tp. F304/F316
Elbow - 1"	Fitting	Stainless steel	SA-403 Tp. WP304/316
Pipe - 1"	Pipe	Stainless steel	SA-312 Gr. Tp. 304/316
Flange - 1"	Forging	Stainless steel	SA-182 Tp. F304/F316
Grayloc hub	Forging	Stainless steel	SA-182 Tp. F316
Tee	Forging	Stainless steel	SA-182 Tp. F316
Pipe plug	Forging	Stainless steel	SA-182 Tp. F304/F316
Pipe 3/4	Pipe	Stainless steel	SA-312 Gr. Tp. 304/316
Tee	Forging	Stainless steel	SA-182 Tp. F316

TABLE 5.2-5
(Sheet 6 of 6)

Component	Form	Material	Specification (ASTM/ASME)
Recirculation Pump (cont'd.)			
Flange 3/4	Forging	Stainless steel	SA-182 Tp. F304/F316
Grayloc hub	Forging	Stainless steel	SA-182 Tp. F316
Valve body	Plate	Stainless steel	SA-240 Tp. 304/316
Valve bonnet	Plate	Stainless steel	SA-240 Tp. 304/316
Coil - inner	Tubing	Stainless steel	SA-213 Gr. Tp. 316
Pipe cap	Forging	Stainless steel	SA-182 Tp. F316
Recirculation Piping			
Pipe	Rolled and welded	Stainless steel	SA-358 Tp. 316K
Cross, tee, cap, contour nozzle, elbow, concentric reducer	Fittings	Stainless steel	SA-403 Tp. 316K
Control Rod Drive			
Flanges, plugs, head	Forging	Stainless steel	SA-182 F304
Nut	Bar	Stainless steel	SA-193 Gr. B8
Indicator tube	Pipe	Stainless steel	SA-312 Tp. 316
Drive housing	Pipe	Stainless steel	SA-312 Tp. 304
	Welds	Stainless steel	SFA5.9 Tp. 308L SFA5.4 Tp. 308L
	Forging	Stainless steel	SA-182 Tp. F304
In-core housing	Tube	Stainless steel	SA-213 Tp. 304
	Welds	Stainless steel	SFA5.9 Tp. 308L SFA5.4 Tp. 308L
	Forging	Stainless steel	SA-182 Tp. F304

TABLE 5.2-6
(Sheet 1 of 1)
BWR WATER CHEMISTRY**

	Concentration (ppb)				Conductivity	
	Iron	Copper	Chloride	Oxygen	μmho/cm (25°c)	pH (25°c)
Condensate (1)	15-30	3-5	≤20	30-200	0.1	7*
Condensate treatment effluent (2)	≤10	<2	<2	30-200	<0.1	6.5-7.5
Feedwater (3)	5-15	<2	<2	30-200	0.1	6.5-7.5
Reactor water (4)						
Normal operation	10-50	<20	<200	100-300	<1	5.6-8.6
Shutdown	-	-	500		<1	4-10
Hot standby	-	-	<20	See Section 5.2.3.2.2	<1	7*
Depressurized	-	-	500	8,000	<10	5.3-8.6
Steam (5)	0	0	0	10,000-30,000	0.1*	
Control rod drive cooling water (6)	50-500	-	<20	≤200	≤0.1	7*

NOTE: Numerals in parentheses refer to locations delineated on Figure 5.2-6.

* Approximately.
** Chemistry of the RCS is maintained within the limits specified in TRM Section 3.4.1.

TABLE 5.2-7
(Sheet 1 of 1)

THIS TABLE HAS BEEN DELETED

TABLE 5.2-8
(Sheet 1 of 1)
LEAK DETECTION METHODS, ACCURACY, AND SENSITIVITY

Leak Detection Method	Accuracy	Sensitivity
Drywell floor drain tank level	1 gpm	1 gpm/hr
Drywell equipment drain tank level	1 gpm	1 gpm/hr
Airborne particulates	-	6.6-12 uCi/cc[(1)(2)]
Gaseous radioactivity	-	1.9-07 uCi/cc[(1)(2)]

NOTE: 6.6-12 = $6.6x10^{-12}$

(1) Minimum detectable concentration based on 10-min Minimum Detectable Level (MDL) for I-131 (particulate) and Xe-133 (gaseous).

(2) The airborne particulate and gaseous radioactivity readings are compared with identified and unidentified leak rates to determine the alarm setpoints necessary to detect the Technical Specification limit of 5 gpm unidentified leakage.

TABLE 5.2-9
(Sheet 1 of 1)
SUMMARY OF SYSTEM ISOLATION/ALARMS OF SYSTEMS MONITORED
AND THE LEAK DETECTION METHODS USED

(SUMMARY OF ISOLATION SIGNALS AND ALARMS
SYSTEM ISOLATION VS. VARIABLE MONITORED)(3)

System Isolated(1)	Variable Monitored															
	A (2,4)	B (2)	C	D	E	F	G (2)	H	J	K (2)	L	M	N	O	P	Q
Main Steam	1		I	I	I	I										
Recirculation (Sample Line)	2															
RHR	3	I						I			I	I			I	I
RCIC								I	I	I	I	I			I	I
RWCU	2												I	I	I	
Containment Isolation	2						I									

KEY:

A = Reactor Vessel Water Level
B = Reactor Pressure
C = Turbine Building Leak Detection (Ambient Temperature, High)
D = Main Steam Tunnel (including lead enclosure) Ambient Temperature, High
E = Main Steam Tunnel Differential Temperature, High
F = Main Steam Line Flow Rate, High
G = Drywell Pressure, High
H = RHR Equipment Area Ambient Temperature, High
J = RCIC Equipment Area Ambient Temperature, High
K = RCIC Exhaust Diaphragm Pressure, High
L = RHR/RCIC Steam Supply Differential Pressure (High Flow)
M = RHR/RCIC Steam Supply Differential Pressure (Instrument Line Break)
N = RWCU Process Piping Differential Flow, High
O = RWCU Equipment Area Ambient Temperature, High
P = Reactor Building Pipe Chase Area Ambient Temperature, High
Q = Reactor Building General Area Ambient Temperature, High
I = Isolate Alarm, and Indicate or Record

(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 column correspond to reactor water levels as shown on Figure 5.2-4 and are levels at which isolation valves of the related system are closed.

TABLE 5.2-10
(Sheet 1 of 2)
SUMMARY OF ISOLATION ALARMS OF SYSTEM MONITORED
AND LEAK DETECTION METHODS USED

(SUMMARY OF VARIABLE TRIP ALARMS LEAKAGE
SOURCE VS. GENERATED VARIABLES)

Source of Leakage	Affected Variable Monitored																					
	A	B	C*	D*	E	F	G*	H	J	K	L	M	N	O	P	Q	R	S	T	U	V	W
Main Steam Line	X		I	I	I		I	I	I				I									
		X		I						I			I		I							
RCIC/RHR Steam Line	X		I	I	I		I	I					I									
		X		I									I	I		I					I	I
RCIC Steam Line																						
		X		I									I	I		I					I	I
RWCU Water	X		I	I	I		I	I														
		X		I										I		I	I				I	
HPCS Water	X				I																	
		X												I								
LPCS Water	X				I															I		
		X												I								
Recirculation Pump Seal	X		I			I	I	I			I											
Feedwater	X		I	I	I		I	I														
		X										I										
RHR Water	X		I	I	I			I					I							I		
		X		I										I		I			I			I
Reactor Vessel Head Seal	X											I										
Refueling Pool																						
		X												I				I				

TABLE 5.2-10
(Sheet 2 of 2)

Source of Leakage	Affected Variable Monitored																					
	A	B	C*	D*	E	F	G*	H	J	K	L	M	N	O	P	Q	R	S	T	U	V	W
Miscellaneous Leaks	X				I																	
		X												I								
RCIC Water	X				I																	
		X												I								

KEY:

A = Located Inside Primary Containment
B = Located Outside Primary Containment
C = Drywell Pressure, High
D = Reactor Water Level, Low
E = Drywell Floor Drain Tank Fill Rate, High
F = Drywell Equipment Drain Tank Fill Rate, High
G = Fission Product Radiation, High
H = Drywell Temperature, High
J = Safety/Relief Valve Discharge Pipe Temperature, High
K = Main Steam Line Tunnel Radiation, High
L = Pump Seal Flow, High
M = Seal Pressure, High
N = Flow Rate, High
O = Floor Drain Sump, High Fillup/Pumpout (Reactor Building)
P = Main Steam Line Tunnel Ambient and Differential Temperature, High (Reactor Building)
Q = Equipment Area Ambient Temperature, High (Reactor Building)
R = RWCU Differential Flow, High
S = Seal Drain Flow, High
T = Intersystem Leakage (Radiation), High
U = ECCS Injection Line Leakage (Internal to Reactor Vessel) Differential Pressure
V = Reactor Building Pipe Chase Area Ambient Temperature, High
W = Reactor Building General Area Ambient Temperature, High

I = Alarm and Indicate or Record Only
X = Location of leakage source
* = These leak detection signals are provided by other systems

5.3 REACTOR VESSEL

5.3.1 Reactor Vessel Materials

5.3.1.1 Materials Specifications

The materials used in the RPV and appurtenances are listed in Table 5.2-5 together with the applicable specifications.

5.3.1.2 Special Processes Used for Manufacturing and Fabrication

The RPV is primarily constructed from low-alloy, high-strength steel plate and forgings. Plates are ordered to ASME SA-533 Grade B, Safety Class 1, and forgings to ASME SA-508, Safety Class 2. These materials are melted to fine grain practice and are supplied in the quenched and tempered condition. Further requirements include vacuum degassing to lower the hydrogen level and improve the cleanliness of the low-alloy steels.

Studs, nuts, and washers for the main closure flange are ordered to ASME SA-540, Grade B23 or B24. Welding electrodes are low-hydrogen type ordered to ASME SFA-5.5, SFA-5.1, SFA-5.4, SFA-5.9, SFA-5.11, SFA-5.14, SFA-5.17, and SFA-5.18.

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 Section III standards. Fracture toughness properties are also measured and controlled in accordance with ASME Section III requirements. Refer to Appendix 5A, Section 5A.1, for discussion.

The RPV was fabricated in accordance with GE-approved drawings, fabrication procedures, and test procedures. The shells and vessel heads are made from formed plates, and the flanges and nozzles from forgings. Welding performed to join these vessel components is in accordance with procedures qualified in accordance with ASME 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 (HAZ), and weld metal.

Submerged arc and manual stick electrode welding processes are employed. Electroslag welding is not permitted. Preheat and interpass temperatures employed for welding of low-alloy steel meet or exceed the requirements of ASME Section III. Postweld heat treatment at 1,100°F minimum is applied to all carbon and low-alloy steel welds.

Radiographic examination is performed on all pressure-containing welds in accordance with requirements of ASME Section III, Subsubarticle 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 RPVs meet or exceed requirements of ASME Section III, Safety Class 1 vessels.

5.3.1.3 Special Methods for Nondestructive Examination

The materials and welds on the RPV were examined in accordance with methods prescribed and meet the acceptance requirements specified by ASME Section III. In addition, the pressure-retaining welds were ultrasonically examined using manual techniques. The ultrasonic examination method, including calibration, instrumentation, scanning sensitivity, and coverage, was based on the requirements imposed by ASME Section XI, 1980 Edition through the Winter 1980 Addenda, Appendix I. Acceptance standards were equivalent to or more restrictive than those required by ASME Section XI, 1980 Edition through the Winter 1980 Addenda. Nozzle weld overlays for FWS are examined per the guidance of Code Case N504-1 and EPRI guidelines.

5.3.1.4 Special Controls for Ferritic and Austenitic Stainless Steels

5.3.1.4.1 Compliance With Regulatory Guides

Regulatory Guide 1.31 Controls on stainless steel welding are discussed in Section 5.2.3.4.2.

Regulatory Guide 1.34 Electroslag welding was not employed for the RPV fabrication.

Regulatory Guide 1.43 RPV specifications require that all low-alloy steel be produced to fine grain practice. The requirements of this regulatory guide are not applied to BWR vessels.

Regulatory Guide 1.44 Controls to avoid severe sensitization are discussed in Section 5.2.3.4.1.

Regulatory Guide 1.50 Preheat controls are discussed in Section 5.2.3.3.2.

Regulatory Guide 1.71 Qualification for areas of limited accessibility is discussed in Section 5.2.3.4.2.

Regulatory Guide 1.99 Predictions for changes in transition temperature and upper shelf energy were assessed to be in accordance with the requirements of RG 1.99.

5.3.1.5 Fracture Toughness

5.3.1.5.1 Compliance with 10CFR50 Appendix G

The interpretation of and compliance to Appendix G of 10CFR50 for Safety Class 1 RCPB components is as discussed in Section 5.3.2 and Appendix 5A with the following exceptions:

1. The specific temperature limits for operation when the core is critical are based on 10CFR50 Appendix G, May 1983.

2. A minimum boltup and pressurization temperature of 70°F is required, which is at least 60°F above the flange region nil ductility transition reference temperature (RT_{NDT}) of 10°F. This exceeds the minimum RT_{NDT} temperature required by ASME 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 the presence of stresses due to boltup is most limiting.

Method of Compliance

The following items are the interpretations and methods used to comply with Appendix G of 10CFR50. The fracture toughness test results and the background information used as the basis to show compliance with 10CFR50 Appendix G are reported in this section and in Appendix 5A. The RPV materials are in compliance with ASME Section III, 1971, including the 1972 Winter Addenda.

Records and Procedures for Impact Testing

Appendix G allows the component manufacturer to assign to qualified subcontractors, such as material suppliers, the actual preparation of written impact testing procedures. Personnel involved in impact testing are qualified to written impact testing procedures. The Unit 2 RPV had been ordered prior to August 16, 1973, and records may not be sufficient to document full compliance to Appendix G; however, sufficient records are available to ensure that the technical requirements are met.

Specimen Orientation for Original Qualification Versus Surveillance

The special beltline longitudinally-oriented Charpy specimens, required by the general reference Subarticle NB-2300 and, specifically, Subparagraph NB-2322.2(a)(6), are not to be included in the surveillance program. ASTM E185-73 also does not require longitudinal specimens as part of the surveillance program.

Charpy V-Curves for the RPV Beltline

The orientation of impact test specimens for the Appendix G requirements complies with the requirements of Subparagraph NB-2322.2(a)4 (transverse specimen) for plate material as opposed to Subparagraph NB-2322.2(a)(6) (longitudinal specimen). The general reference to Paragraph NB-2322 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 allows this curve to comply with ASTM E185-73.

The number, type, and locations of specimens necessary for the full curves are those required to comply with Paragraphs 4.3 and 4.4 of ASTM E185-73. This interpretation is considered necessary to assure that the adjusted reference temperature of irradiated base metal, HAZ and weld metal called for in Appendix H can be based on directly comparable data for the unirradiated reference temperature.

The beltline plate material with the highest predicted end-of-life (EOL) RT_{NDT} value, and the weld material with the highest predicted shift in RT_{NDT}, were used for surveillance specimen base material and weld material to provide a conservative adjusted reference temperature for the beltline material. The weld test plate for the surveillance program specimens had the principal working direction transverse to the weld seam to assure that HAZ specimens were parallel to the principal working direction to represent the longitudinal weld seams in Unit 2 beltline.

Upper Shelf Energy for Beltline

Upper shelf energy test results for beltline materials is presented in Table 5.3-1. It can be seen that minimum upper shelf values for plate heat numbers C3121-2 and C3147-1 are below the 75 ft-lb value presently specified in 10CFR50 Appendix G. NRC Branch Technical Position (BTP) MTEB 5-2 permits minimum upper shelf values of 70 ft-lb when the EOL fluence is less than 1×10^{19} n/sq cm. Since the peak EOL fluence for this vessel is 1.1×10^{18} n/sq cm (Ref. 3) or 1.62×10^{18} n/sq cm (extended power uprate [EPU/MELLLA+]), the 71 and 70 ft-lb values measured for plates C3121-2 and C3147-1 are acceptable.

Bolting Materials

Bolting for Unit 2 meets the requirement of Appendix G.

Alternative Procedures for the Calculation of Stress Intensity Factor

Stress intensity factors were calculated by the methods of ASME Section III Appendix G. Discontinuity regions were evaluated, as well as shell and head areas. 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. It has been determined that an additional restriction on operating limits would be required for outside surface flaw size greater than 0.24 in at the outside surface of the flange-to-shell joint (based on additional analyses made for comparable BWR 6 reactor vessels). It has been demonstrated, using a test mockup of these areas, that smaller defects can be detected by the ultrasonic in-service examination 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.

Fracture Toughness Margins in the Control of Reactivity

Appendix G of ASME Section III was used in determining pressure-temperature (P-T) limitations for all phases of plant operation. The additional requirements of 10CFR50 Appendix G, May 1983, are included in the P-T limitations.

Results of Chemical Analysis and RT_{NDT} Evaluations

See Tables 5.3-1, 5.3-2a, and 5.3-2b.

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.

The original reactor vessel materials surveillance specimens were provided in accordance with requirements of ASTM E185-73 and 10CFR50 Appendix H, except for material selection indicated in Section 5.3.1.5.1. Materials for the program were selected to represent materials used in the reactor beltline region. Specimens were manufactured from a plate actually used in the beltline region, and a weld typical of those in the beltline region, and thus represent base metal, weld material, and the weld HAZ material. The plate and weld were heat treated in a manner that simulates the actual heat treatment performed on the core region shell plates of the completed vessel.

Each in-reactor surveillance capsule contains 36 Charpy V-notch specimens. The capsule loading consists of 12 specimens each of base metal, weld metal, and HAZ material. A set of out-of-reactor baseline Charpy V-notch specimens and archive material are provided with the surveillance test specimens.

In Reference 7, the NRC approved Unit 2 participation in the BWR Vessel and Internals Project (BWRVIP) Integrated Surveillance Program (ISP), as described in BWRVIP-78 (Reference 4) and BWRVIP-86-A (Reference 5). The NRC approved the ISP for the industry in their safety evaluation dated February 1, 2002 (Reference 6). The ISP meets the requirements of 10CFR50, Appendix H. Participation in the ISP replaces the Unit 2 plant-specific vessel material surveillance program.

The current surveillance capsule withdrawal schedule for Unit 2 representative materials is based on the latest NRC-approved

version of BWRVIP-86 (Reference 5). No capsules from the Unit 2 vessel are included in the ISP. Capsules from other plants will be removed and specimens will be tested in accordance with the ISP implementation plan. The results from these tests will provide the necessary data to monitor embrittlement of the Unit 2 vessel. However, the Unit 2 surveillance capsules will remain in place and will continue to be available for removal and testing, in accordance with the NRC-approved ISP.

5.3.1.6.2 Neutron Flux and Fluence Calculations

A description of the methods of analysis is contained in Sections 4.1.4.5 and 4.3.2.8. The peak fluence of 1/4 T depth of the vessel beltline material is 1.1 x 10^{18} n/sq cm (Ref. 3) or 1.12 x 10^{18} n/sq cm (EPU/MELLLA+)(Reference 11). All predictions of radiation damage to the reactor vessel beltline material were made using peak fluence values.

5.3.1.6.3 Predicted Irradiation Effects on Vessel Beltline Materials

Estimated maximum changes in RT (initial reference temperature) and upper shelf fracture energy, as a function of the EOL fluence at the 1/4 T depth of the vessel beltline materials, are listed in Tables 5.3-2a and 5.3-2b. The predicted peak EOL fluence at the 1/4 T depth of the vessel beltline is 1.1 x 10^{18} n/sq cm (Ref. 3) or 1.12 x 10^{18} n/sq cm (EPU/MELLLA+) (Reference 11) after 40 yr or 60 yr of service, respectively. Transition temperature changes and changes in upper shelf energy were calculated in accordance with the guidance of RG 1.99. Reference temperatures were established in accordance with 10CFR50 Appendix G, and Subsubarticle NB-2330 of ASME Section III.

5.3.1.6.4 Positioning of Surveillance Capsules and Methods of Attachment

Surveillance specimen capsules are located at three azimuths at a common elevation in the core beltline region. The sealed capsules are not attached to the vessel but are mechanically retained by capsule brackets welded to the vessel cladding as shown on Figure 5.3-1. Since reactor vessel specifications require that all low-alloy steel pressure vessel boundary material be produced to fine grain practice, underclad cracking is of no concern. The capsule holder brackets allow the removal and reinsertion of capsule holders. These brackets are designed, fabricated, and analyzed to the requirements of ASME Section III. A positive spring-loaded locking device is provided to retain the capsules in position throughout any anticipated event during the lifetime of the vessel.

In areas where brackets, such as the surveillance specimen holder brackets, are located, additional nondestructive 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 surrounding band 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 Section III standards. Cladding thickness is required to be at least 1/8 in. The above requirements have been successfully applied to a variety of bracket designs that are attached to weld-deposited stainless steel cladding or weld buildups in many operating BWR RPVs.

ISI examinations of core beltline pressure-retaining welds are performed from the outside surface of the RPV. 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 ISI ultrasonic examinations performed from the outside surface of the vessel.

NOTE: Surveillance specimen capsule at 3° azimuth location was removed for testing in accordance with the original plant-specific material surveillance program.

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. The design and analysis of this area of the vessel is in full compliance with all ASME Section III, Safety Class 1, code requirements. The material for studs, nuts, and washers is SA-540 Grade B23 or B24 at the 130,000 psi-specified minimum yield strengths level.

Hardness tests are performed on all main closure bolting to demonstrate that heat treatment has been properly performed. A minimum of 45 ft-lb Charpy V-notch (C_v) energy and 25 mils lateral expansion is required at 70°F. The maximum reported ultimate tensile strength is below the 170,000-psi maximum specified in RG 1.65. Also, the Charpy impact test requirements of 10CFR50 Appendix G are satisfied, since the lowest reported C_v energy is 46 ft-lb at +10°F, compared to the requirement of 45 ft-lb at 70°F, and the lowest reported C_v expansion was 26 mils, compared to the 25 mils required. Studs, nuts, and washers are ultrasonically examined in accordance with ASME Section III, Paragraph NB-2585, and the following additional requirements:

1. Examination was performed after heat treatment and prior to machining threads.

2. Straight beam examination was performed on 100 percent of each stud. Reference standard for the radial scan is a 1/2-in diameter flat-bottom hole having a depth equal to 10 percent of the material thickness. For the end scan the standard of Paragraph NB-2585 is used.

3. Nuts and washers were examined by angle beam from the outside circumference in accordance with ASME SA-388 in both the axial and circumferential directions.

The surface examinations required by Paragraph NB-2583 are applied after heat treatment and threading.

There are no metal platings applied to closure studs, nuts, or washers. A manganese-phosphate coating is applied to threaded areas of studs and nuts and bearing areas of nuts and washers to assist in retaining lubricant on these surfaces. Subsequent to fabrication, the studs are lubricated with a graphite/alcohol or nickel powder base lubricant.

RG 1.65 defines acceptable materials and testing procedures with regard to reactor vessel closure stud bolting for light-water-cooled reactors.

The RPV closure studs are SA-540 Grade B23 or B24 (AISI-4340) and have a maximum ultimate tensile strength of 170 ksi. Additionally, the bolting material was specified to have Charpy V-notch impact properties of 45 ft-lb minimum with 25 mils lateral expansion. Nondestructive examination before and after threading is specified to be in accordance with ASME Section III, Subsubarticle NB-2580, which complies with RG 1.65 (C.2).

In accordance with RG 1.65 (C.2.b), the bolting materials were ultrasonically examined after final heat treatment and prior to threading. As required for compliance, the examination was done in accordance with SA-388. The procedures approved for use in practice were judged to ensure comparable material quality and, moreover, were considered adequate on the basis of compliance with the applicable requirements of ASME Section III, Paragraph NB-2583. Additionally, straight beam examination was performed on 100 percent of cylindrical surfaces, and from both ends of each stud using a 3/4-in maximum diameter transducer. In addition to the code-required notch, the reference standard for the radial scan contained a 1/2-in diameter flat-bottom hole with a depth equal to 10 percent of the thickness. The end scan standard contained a 1/4-in diameter flat-bottom hole 1/2-in deep. Angle beam ultrasonic examination was performed on the outer cylindrical surface in both a flat and circumferential direction. Surface examinations were performed on the studs and nuts after final heat treatment and threading, as specified by RG 1.65, in accordance with Paragraph NB-2583 of the applicable ASME Code.

Radial scan calibration is based on a 1/2-in (12.7-mm) diameter flat-bottom hole of a depth equal to 10 percent of the material

thickness. Angle beam examination is performed on the outer cylindrical surface of nuts and washers in accordance with ASME SA-388 in both axial and circumferential directions. No indication greater than the indication from the applicable calibration feature was acceptable. A distance-amplitude correction curve in accordance with Paragraph NB-2858 is used for the longitudinal wave examination.

In relationship to RG 1.65 (C.3), stud bolting surfaces are allowed to be exposed 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 fracture toughness requirements for the pressure vessel for testing and operational conditions are specified in Section IV of 10CFR50 Appendix G (May 1983). This appendix requires implementation of the acceptance and performance criteria of Appendix G to Section III of the ASME Code. The basis for the technical requirements of the ASME Code is discussed in Welding Research Council Bulletin 175. Appendix G to 10CFR50 requires that the effects of neutron irradiation on the RT_{NDT} of the beltline materials must be included in the P-T curve calculations. The latest revision (Revision 2) to RG 1.99 is used for this purpose. Calculated adjusted nil ductility transition reference temperature (ART_{NDT}) values and temperature limits are given in this section for limiting locations in the reactor vessel. The P-T limit curves are presented in Figures 1, 2 and 3 of Reference 12. The P-T limit curves reported for up to a peak vessel wall wetted surface fast fluence (E > 1 Mev) of 9.60×10^{17} n/sq cm (Reference 12) were developed for a period of up to 32 EFPY.

All 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 the RT_{NDT} of the flange and adjacent shell region (10°F) plus 60°F. The maximum through-wall temperature gradient from continuous heating or cooling at 100°F/hr was used. The safety factors applied were as specified in ASME Code Section III, Appendix G.

P-T curve calculations are performed on the beltline material which has the highest ART_{NDT} over the period for which the P-T curves are valid. Therefore, ART_{NDT} calculations were performed using RG 1.99 Revision 2, and the results are presented in Tables 5.3-2a and 5.3-2b. The limiting material is plate C3147.

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 can be fully tensioned at 70°F to seal the closure flange O-rings for the purpose of raising the reactor water level above the closure flanges 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 required bolt preload. The fully preloaded boltup limits are shown on Figures 1 through 3 of Reference 12.

The RT_{NDT} is no greater than +10°F for both the vessel and the head flange material, as well as the plate material that is connected to the closure flanges.

5.3.2.1.2 Temperature Limits for Preoperational System Hydrostatic Tests and Inservice Pressure Tests

Based on 10CFR50 Appendix G, if there is no fuel in the reactor, the preoperational system hydrostatic test at 1,563 psig may be performed at a minimum temperature of 100°F.

The fracture toughness analysis for inservice system pressure test with fuel in the vessel resulted in the P-T limits shown on Figure 1 of Reference 12. The curves are based on an initial RT_{NDT} of 0°F.

The calculated adjustment to the RT_{NDT} shown on Figure 5.3-3 (based on Revision 2 to RG 1.99) is used in the analysis to account for the effect of fast neutrons.

The predicted shift in the RT_{NDT} from Figure 5.3-3 (based on the neutron fluence at 1/4 T of the vessel wall thickness) is added to the beltline curve to account for the effect of fast neutrons.

Performance of system leakage or hydrostatic testing is conducted in accordance with the requirements outlined in Technical Specifications.

5.3.2.1.3 Operating Limits During Heatup, Cooldown, and Core Operation

The fracture toughness analysis was done for the assumed heatup or cooldown rate of 100°F/hr. 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 noncritical heatup and cooldown, and critical operations (heatup and cooldown), as shown on Figures 2 and 3 of Reference 12. Figure 3 of Reference 12 applies whenever the core is critical.

RG 1.99 Revision 2 requires an estimation of the standard deviation (σ_1) of the initial RT_{NDT}. The RT_{NDT} for plate C3147 was determined from Charpy data since 50 ft-lbs was not achieved within 60°F of the NDT. Therefore, σ_I was determined by fitting the Charpy data using a nonlinear least squares regression

analysis and using the confidence band data to estimate σ_I. Using this approach, σ_I was taken to be 14.5°F.

For EPU, NRC-approved methods (Reference 8) were used for determining the initial RT_{NDT}. Because this method operates on the lowest Charpy energy value and provides a conservative adjustment to the 50 ft-lb level, σ_I was taken to be 0°F.

In order to assess the ART_{NDT} at the 1/4 T and 3/4 T positions, RG 1.99 Revision 2 requires an assessment of the peak fast (E>1Mev) neutron flux at the ID surface of the pressure vessel. RG 1.190 compliant fluence methods were used to derive the P-T curves. The 1/4 T fluence was calculated to be 3.77 x 10^{17} n/cm^2 (Reference 3) or 1.12 x 10^{18} n/sq cm (EPU/MELLLA+) Reference 11, and the 3/4 T fluence was calculated to be 1.79 x 10^{17} n/cm^2 (Reference 3).

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 of the adjusted reference temperature does not exceed 200°F.

5.3.2.1.5 Predicted Shift in RT_{NDT}

The allowable internal vessel pressure for a specific coolant temperature is a function of several key variables including the ART_{NDT}. The ART_{NDT} for the vessel beltline region enters the P-T calculations directly via the reference stress intensity factor relation (K_{IR}). Therefore, it is necessary to provide reasonable and conservative estimates of the shift in ΔRT_{NDT} for the period of time for which the P-T calculations will be used. The ART_{NDT} was calculated using Revision 2 to RG 1.99 (Figure 5.3-3).

5.3.2.2 Operating Procedures

By comparison of the pressure versus 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 are not exceeded during any foreseeable upset condition. Reactor operating procedures have been established in such a manner that actual transients are less 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 heads and/or shell areas has 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 fluid temperature, so the applicable operating limits are given on Figures 1, 2 and 3 of Reference 12. For a temperature of 250°F, the maximum allowable pressure exceeds 1,600 psig for the intended margin against nonductile failure. The maximum transient pressure of 1,180 psig is within the specified allowable limits.

5.3.3 Reactor Vessel Integrity

The reactor vessel was fabricated for GE's Nuclear Energy Division by CBI Nuclear Company, and was subject to the requirements of GE's QA program. The CBI Nuclear Company has extensive experience in fabricating 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 agreement between Chicago Bridge and Iron Company and GE. Prior experience by the Chicago Bridge and Iron Company with GE reactor vessels dates back to 1966.

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

Inspection surveillance of the reactor vessel fabricator's in-process manufacturing, fabrication, and testing operations is in accordance with GE's QA program and approved inspection procedures. The reactor vessel fabricator is responsible for the first-level inspection of his manufacturing, fabrication, and testing activities. GE is responsible for the first level of audit and surveillance inspection. Adequate documentary evidence that the reactor vessel material, manufacture, testing, and inspection conform to the specified QA requirements contained in the procurement specification is available at the fabricator plant site.

RG 1.2 states that a suitable program be followed to ensure that the RPV will behave in a nonbrittle manner under LOCA conditions. Should it be considered that the margin of safety against RPV brittle fracture due to ECCS operation at any time during vessel life is unacceptable, the regulatory guide states that an engineering solution, such as annealing, could be applied to ensure adequate recovery of the fracture toughness properties of the vessel material.

An analysis of the structural integrity of boiling water RPVs during a design basis accident (DBA) has been performed. While the analysis specifically addressed the BWR 6 vessels only, the analysis was determined applicable to the Unit 2 vessel.

The analysis included:

1. Description of the LOCA event.

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

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

4. Consideration of radiation effect on material toughness (nil ductility transition temperature [NDTT] shift and changes in toughness).

5. 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 LOCA.

The criteria of 10CFR50 Appendix G are interpreted as the established requirements for annealing. Appendix G requires the vessels to be designed for annealing of the beltline only where the predicted value of ART_{NDT} exceeds 200°F, as defined in Paragraph NB-2331 of ASME Section III. This predicted value is not exceeded; therefore, design for annealing is not required.

For further discussion of fracture toughness of the RPV, refer to Section 5.3.1.5.

5.3.3.1 Design

5.3.3.1.1 Description

<u>Reactor Vessel</u>

The reactor vessel (Figure 5.3-4) is a vertical, cylindrical pressure vessel of welded construction. The vessel was designed, fabricated, tested, inspected, and stamped in accordance with ASME Section III, Safety Class 1, 1971 Edition through 1972 Winter Addenda, based on the date of order placement. Design of the reactor vessel and its support system meets Category I equipment requirements. The materials used in the RPV are listed 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 nozzle and nozzle weld zones.

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

The vessel top head is secured to the reactor vessel by studs and nuts. These nuts are tightened with a stud tensioner. The vessel flanges are sealed with two concentric metal seal rings designed to permit no detectable leakage through the inner or

outer seal at any operating condition, including heating to operating pressure and temperature at a maximum rate of 100°F/hr. To detect seal failure, a vent tap is located between the two seal rings. A pressure monitor line and transmitter are attached to the tap to provide an indication of leakage from the inner seal ring seal.

Shroud Support

The shroud support consists of a horizontal support plate that fits around a vertical cylinder that is supported by vertical stilts from the bottom head. These parts are welded together and the stilts and support plate are welded to the RPV. This shroud support is designed to carry the weight of the shroud, peripheral fuel elements, neutron sources, core plate, top guide, the steam separators, and 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 dynamic loadings. The shroud support design is specified to meet ASME Code Section III, Subsection NG, stress limits.

Protection of Closure Studs

The BWR does not use borated water for reactivity control during normal operation or refueling. Therefore, this section is not applicable.

5.3.3.1.2 Safety Design Basis

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

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

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

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

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

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

5.3.3.1.3 Power Generation Design Basis

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

1. The reactor vessel has been designed for a useful life of 40 yr and has been evaluated for 60 yr.

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

3. Design of the reactor vessel and appurtenances 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.

Vessel Support

The concrete and steel vessel support pedestal is constructed as an integral part of the building foundation. Steel anchor bolts, set in the concrete, extend through the bearing plate and secure the flange of the reactor vessel support skirt to the bearing plate, and thus to the support pedestal.

Control Rod Drive Housings

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

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).

Reactor Vessel Insulation

The reactor vessel insulation has an average maximum heat transfer rate of approximately 44.0 Btu/hr/sq ft of insulation surface at the operating conditions of 550°F for the vessel and 135°F for the drywell air. The bottom head insulation heat transfer rate is 80 Btu/hr/sq ft. The insulation panels for the cylindrical shell of the vessel are held in place by insulation supports located on the biological shield. The bottom head insulation is supported by the skirt flange. The insulation is designed to be removable over those portions of the vessel where inspection is required by the ISI program plan.

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 nozzle has a flange with large groove facing. The drain nozzle is of the full penetration weld design. The recirculation inlet nozzles (located as shown on Figure 5.3-4), feedwater inlet nozzles, core spray inlet nozzles, and the LPCI nozzles all have thermal sleeves. Nozzles connecting to stainless steel piping have safe ends or extensions made of stainless steel (316) with Inconel weld. These safe ends or extensions are welded to the nozzles after the pressure vessel is heat treated to avoid furnace sensitization of the stainless steel. The material used is compatible with the material of the mating pipe, as shown in Table 5.2-5.

The HPCS nozzle used for standby liquid control (SLC) is designed to minimize thermal shock effects on the reactor vessel in the event that use of the standby liquid control system (SLCS) is required.

The solution to the feedwater nozzle cracking problem involved several elements, including nozzle clad removal and thermal sleeve redesign. A description of the design changes incorporated for Unit 2 and appropriate analysis to support such changes are presented in Reference 2.

The LPCI and water level instrumentation nozzles contained in shell #2 (lower-intermediate shell ring) fall within the region in the beltline that attains irradiation in excess of 1×10^{17} n/sq cm. This level of fluence is defined in 10CFR50 Appendix H as the threshold for requiring fracture toughness considerations. These nozzles are evaluated in accordance with RG 1.99 as discussed in Section 5.3.1.5.

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.

Reactor Vessel Schematic (BWR)

The reactor vessel schematic is contained on Figure 5.3-4. Trip system water levels are indicated as shown on Figure 5.3-5.

5.3.3.2 Materials of Construction

All materials used in the construction of the RPV conform to the requirements of ASME Section II materials. The vessel heads, shells, flanges, and nozzles are fabricated from low-alloy steel plate and forgings purchased in accordance with ASME Specifications SA-533 Grade B, Safety Class 1, and SA-508, Safety 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

All fabrication of the RPV 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 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 met or exceeded the requirements of ASME Section III, Subsection NA. Postweld heat treatment of 1,100°F minimum is applied to all carbon and low-alloy steel welds.

All previous BWR pressure vessels have employed similar fabrication methods. These vessels have operated for periods up to 20 yr and their service history is excellent. The vessel fabricator, CBI Nuclear Company, has had extensive experience with BWR reactor vessels and has been a primary supplier for domestic BWR vessels and some foreign vessels.

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 Section III

requirements. Welds on the RPV were examined in accordance with methods prescribed and meet the acceptance requirements specified by ASME Section III. In addition, the pressure-retaining welds were ultrasonically examined using acceptance standards that were required by ASME Section XI, 1980 Edition through the Winter 1980 Addenda.

5.3.3.5 Shipment and Installation

The completed reactor vessel was thoroughly cleaned and examined prior to shipment. The vessel was tightly sealed for shipment to prevent entry of dirt or moisture. Preparations for shipment were 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 ensure that vessel integrity was maintained; for example, access controls were applied to personnel entering the vessel, weather protection was provided, and periodic cleanings were performed.

5.3.3.6 Operating Conditions

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

1. The average rate of change of reactor coolant temperature during normal heatup and cooldown will not exceed 100°F during any 1-hr period.

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

3. The pump in an idle reactor recirculation loop will not be started unless the coolant temperature in that loop is within 50°F of saturated reactor coolant temperature corresponding to the steam dome pressure.

The limit regarding the normal rate of heatup and cooldown (Item 1) assures that the vessel closure, closure studs, vessel support skirt, and CRD housing and stub tube stresses and usage remain within acceptable limits. The limit regarding a vessel temperature limit on recirculating pump operation and power level increase restriction (Item 2) augments the Item 1 limit. This limit ensures that the vessel bottom head region is not warmed at an excessive rate, caused by rapid sweepout 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 in-leakage and/or low recirculation flow rate during startup or hot standby). The Item 3 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.

These 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. 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 most severe anticipated transients have been included in the design conditions. Therefore, it is concluded that vessel integrity will be maintained during the most severe postulated transients, since all such transients are evaluated in the design of the reactor vessel. The postulated transient for which the vessel has been designed is discussed in Section 5.2.2.

5.3.3.7 Inservice Surveillance

ISI of the RPV is discussed in Section 5.2.4.

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. Operating procedures will be modified as necessary in accordance with program results to assure adequate brittle fracture control.

The ISI program is in accordance with applicable ASME Code requirements, and provides assurance that brittle fracture control and pressure vessel integrity are maintained throughout the service lifetime of the RPV.

5.3.4 References

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7. NRC Letter to NMPNS dated November 8, 2004, "Nine Mile Point Nuclear Station Unit Nos. 1 and 2 - Issuance of Amendments RE: Implementation of the Reactor Pressure Vessel Integrated Surveillance Program (TAC Nos. MC1758 and MC1759)."

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11. GEH Report NEDC-33414P, Revision 1, "Pressure-Temperature Curves for Constellation Generation Group Nine Mile Point Nuclear Station Unit 2," October 2012.

12. Nine Mile Point Unit 2 Pressure and Temperature Limits Report (PTLR), PTLR-2, Revision 0.

TABLE 5.3-1
(Sheet 1 of 4)
UNIT 2 REACTOR VESSEL CHARPY TEST RESULTS
VESSEL BELTLINE CHEMICAL COMPOSITION

I. Vessel Beltline Material Identification

A. No. 2 shell ring

Plates	Pc.	Heat
	Pc. 22-1-1	Heat C3065-1
	Pc. 22-1-2	Heat C3121-2
	Pc. 22-1-3	Heat C3147-1

Welds in No. 2 shell ring. Vertical seams BD, BE, BF
RACO-1NMM, Heat 5P5657, Lot 0931

B. No. 1 shell ring

Plates	Pc.	Heat
	Pc. 21-1-1	Heat C3147-2
	Pc. 21-1-2	Heat C3066-2
	Pc. 21-1-3	Heat C3065-2

Welds in No. 1 shell ring. Vertical seams BA, BB, BC
RACO-1NMM, Heat 5P6214B, Lot 0331

C. Girth weld between No. 1 and No. 2 shell rings. Seam AB
RACO-1NMM, Heat 4P7465, Lot 0751
RACO-1NMM, Heat 4P7216, Lot 0751

D. Nozzles
N6 (LPCI Nozzle) Forging 854A Heat Q2QL3W
N12 (Water Level Instrumentation Nozzle) Heat 717456

II. Chemical Analyses For Beltline Materials (wt %)

A. Plates	C	Mn	P	S	Cu	Si	Ni	Mo	V	Al
Pc. 22-1-1, C3065-1	0.21	1.36	0.010	0.015	0.06	0.25	0.63	0.58	0	0.021
Pc. 22-1-2, C3121-2	0.20	1.25	0.012	0.015	0.09	0.26	0.65	0.56	0	0.030
Pc. 22-1-3, C3147-1	0.19	1.28	0.012	0.015	0.11	0.24	0.63	0.56	0	0.020
Pc. 21-1-1, C3147-2	0.19	1.28	0.012	0.015	0.11	0.24	0.63	0.56	0	0.020
Pc. 21-1-2, C3066-2	0.19	1.25	0.012	0.015	0.07	0.21	0.64	0.56	0	0.018
Pc. 21-1-3, C3065-2	0.21	1.36	0.010	0.015	0.06	0.25	0.63	0.58	0	0.021

TABLE 5.3-1
(Sheet 2 of 4)

II. Chemical Analyses For Beltline Materials (wt %)(cont'd.)

B. Welds	C	Mn	P	S	Cu	Si	Ni	Mo	V	Al
Ht. 5P5657[1]	0.075	1.47	0.015	0.021	0.07	0.42	0.71	0.42	0.007	0.024
Lot 0931[2]	0.078	1.45	0.016	0.020	0.04	0.44	0.89	0.50	0.006	0.016
Ht. 5P6214B[1]	0.051	1.39	0.013	0.017	0.02	0.53	0.82	0.52	0.004	0.007
Lot 0331[2]	0.085	1.24	0.011	0.014	0.014	0.51	0.70	0.44	0.003	0.007
Ht. 4P7465[1]	0.050	1.63	0.010	0.013	0.02	0.39	0.82	0.45	0.006	0.010
Lot 0751[2]	0.061	1.50	0.012	0.014	0.02	0.33	0.80	0.42	0.006	0.006
Ht. 4P7216[1]	0.07	1.40	0.011	0.011	0.045	0.45	0.80	0.43	0.07	0.01
Lot 0751[2]	0.087	1.45	0.011	0.012	0.035	0.33	0.82	0.41	0.05	0.01
C. Forgings										
N6 LPCI Nozzle, Ht. Q2QL3W[3]					0.07		0.86			
N12 Water Level Instrumentation Nozzle, Ht, 717456[4]					0.11		0.65			

TABLE 5.3-1
(Sheet 3 of 4)

III. Unirradiated Fracture Toughness Properties						
Plates Ht. No.	Drop Wt. NDT, °F	Transverse Charpy V-Notch ft-lb	MLE	Temperature (°F)	Reference Temperature (°F)	Upper Shelf (ft-lb)
C3065-1						
Top	-30	55, 60, 63	48, 46, 48	+40		
Bottom	-30	70, 50, 50	42, 52, 41	+50	-10	94 min
C3121-2						
Top	-30	50, 51, 50	46, 41, 44	+40		71 min
Bottom	-50	50, 53, 50	46, 46, 45	+60	0	75 ave
C3147-1						
Top	-20	50, 51, 50	45, 44, 41	+60		70 min
Bottom	-30	50, 50, 52	46, 44, 42	+60	0	74 ave
C3147-2						
Top	-20	52, 50, 50	48, 44, 44	+60	0	86 min
Bottom	-30	51, 56, 51	48, 51, 48	+30		
C3066-2						
Top	-30	58, 72, 58	58, 48, 48	+40		
Bottom	-40	55, 52, 52	45, 42, 45	+40	-20	86 min
C3065-2						
Top	-10	56, 56, 60	50, 48, 46	+70		
Bottom	-40	51, 53, 51	46, 45, 43	+70	+10	83 min
Weld Metal						
5P5657[1]	-60	51, 55, 68	50, 50, 63	0	-60	88 min
Lot 0931[2]	-80	51, 57, 55	50, 54, 40	0	-60	88 min
5P6214B[1]	-50	56, 50, 54	45, 41, 46	+10	-50	88 min
Lot 0331[2]	-40	50, 61, 64	46, 50, 52	+10	-40	96 min
4P7465[1]	-70	63, 57, 68	54, 45, 63	0	-60	102 min
Lot 0751[2]	-60	79, 83, 74	66, 60, 54	0	-60	110 min
4P7216[1]	-60	64, 60, 72, 66, 60	48, 41, 60, 53, 41	+10	-50	89 min
Lot 0751[2]	-80	62, 73, 84	40, 51, 56	-20	-80	98 min

TABLE 5.3-1
(Sheet 4 of 4)

III. Unirradiated Fracture Toughness Properties (cont'd.)

Plates Ht. No.	Drop Wt. NDT, °F	Transverse Charpy V-Notch ft-lb	MLE	Temperature (°F)	Reference Temperature (°F)	Upper Shelf (ft-lb)
Forgings						
N6 LPCI Nozzle Ht. Q2QL3W						
Forging 854A-1	-20	53, 101, 84	48	40	-20	85
Forging 854A-3	-20	64, 74, 70	53	40	-20	85
Forging 854A-2	-20	65, 70, 60	54	40	-20	85
N12 Water Level Instrumentation Nozzle Ht. 717456[5]						

(1) Single Wire.
(2) Tandem Wire.
(3) The %Cu and %Ni values are the average of all ladle and check values provided in the Certified Material Test Report.
(4) The N12 nozzle is classified as a partial penetration in Shell Ring #2. Because the forging is <2.5" thick, fracture toughness evaluation per ASME Code Appendix G, Section G2223(c) is not required. Therefore, evaluation is performed using the chemical properties of the plate material in Shell #2, where this nozzle is located.
(5) The N12 nozzle is classified as a partial penetration in Shell Ring #2. Because the forging is <2.5" thick, fracture toughness evaluation per ASME Code Appendix G, Section G2223(c) is not required. Therefore, evaluation is performed using the fracture toughness properties of the plate material in Shell #2, where this nozzle is located.

TABLE 5.3-2a
(Sheet 1 of 1)
ADJUSTED RT_{NDT} FOR NINE MILE POINT UNIT 2 BELTLINE MATERIALS USING RG 1.99 REV. 2

Plates and Nozzles - Beltline

Heat No.	Wt. % Cu	Wt. % Ni	ASME NB-2300 Start RT_{NDT} (°F)	REF. 3 PT CURVES[4]			54 EFPY[3]		
				ΔRT_{NDT} (°F)	Margin (°F)	ART_{NDT} (°F)	ΔRT_{NDT} (°F)	Margin (°F)	ART_{NDT} (°F)
C3065-1	0.06	0.63	-10	11.6	31.2	32.9	16.3	16.3	23
C3121-2	0.09	0.65	0	18.2	34.3	52.5	25.5	25.5	51
C3147-1	0.11	0.63	0	23.4	37.3	60.7[2]	32.7	32.7	66[2]
C3147-2[1]	0.11	0.63	0	23.4	37.3	60.7[2]	32.4	32.4	65[2]
C3066-2	0.07	0.64	-20	13.8	32.1	26.0	19.1	19.1	18
C3065-2	0.06	0.63	+10	11.6	31.2	52.9	16.1	16.1	42
Nozzles									
N6 LPCI Q2QL3W	0.07	0.86	-20				10.9	10.9	2
N12 Instrumenta-tion[5] 717456	0.11	0.65	0				14.8	14.8	30

(1) These materials are also in the reactor vessel surveillance capsules.
(2) Limiting plate. See Section 5.3.1.6 for a description of the reactor vessel material surveillance program.
(3) Calculations performed using the 54 EFPY EPU fluence.
(4) Calculations performed based on wetted surface fluence of 5.71 x 10^{17} (n/cm^2) derived based on RG 1.190 compliant methods.
(5) Calculated using material properties for Shell 2.

TABLE 5.3-2b
(Sheet 1 of 1)
ADJUSTED RT_{NDT} FOR NINE MILE POINT UNIT 2 BELTLINE MATERIALS USING RG 1.99 REV. 2

Welds - Beltline

Weld Seam	Heat/Lot No.	Wt. % Cu	Wt. % Ni	ASME NB-2300 Start RT_{NDT} (°F)	REF. 3 PT CURVES[6]			54 EFPY[5]		
					ΔRT_{NDT} (°F)	Margin (°F)	ART_{NDT} (°F)	RT_{NDT} (°F)	Margin (°F)	ART_{NDT} (°F)
BD,BB,BF	5P5657/0931[1,2]	0.07	0.71	-60	29.9	41.6	11.5	41.8	41.8	24
	5P5657/0931[1,3]	0.04	0.89	-60	17.0	33.6	-9.4	23.7	23.7	-12
BA,BB,BC	5P6214B/0331[2]	0.02	0.82	-50	8.5	30.2	-11.3	11.7	11.7	-26
	5P6214B/0331[3]	0.014	0.70	-40	7.2	29.9	-3.0	9.9	9.9	-20
AB	4P7465/0751[2]	0.02	0.82	-60	8.5	30.2	-21.3	11.7	11.7	-36
	4P7465/0751[3]	0.02	0.80	-60	8.5	30.2	-21.3	11.7	11.7	-36
	4P7216/0751[2]	0.045	0.80	-50	17.0	33.6	0.6	26.5	26.5	3
	4P7216/0751[3]	0.035	0.82	-80	14.9	32.6	-32.5	20.6	20.6	-38

(1) These materials are also in the reactor vessel surveillance capsules.
(2) Single wire submerged arc process.
(3) Tandem wire submerged arc process.
(4) Limiting weld. See Section 5.3.1.6 for a description of the reactor vessel material surveillance program.
(5) Calculations performed using the 54 EFPY EPU fluence.
(6) Calculations performed based on wetted surface fluence of 5.71×10^{17} (n/cm^2) derived based on RG 1.190 compliant methods.

5.4 COMPONENT AND SUBSYSTEM DESIGN

5.4.1 Reactor Recirculation System

5.4.1.1 Safety Design Bases

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

1. An adequate fuel barrier thermal margin will be assured during postulated transients.

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

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

5.4.1.2 Power Generation Design Bases

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

1. The system will provide sufficient flow to remove heat from the fuel.

2. System design will minimize maintenance situations that would require fuel removal.

5.4.1.3 Description

The reactor recirculation system consists of the two recirculation pump loops external to the reactor vessel. These loops provide the piping path for the driving flow of water to the reactor vessel jet pumps (Figures 5.4-1 and 5.4-2). Each external loop contains one high-capacity motor-driven recirculation pump, a flow control valve (FCV), 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 RCPB and are located inside the drywell structure. The jet pumps are considered to be reactor vessel internals. Their location and mechanical design are discussed in Section 3.9B.5. However, certain operational characteristics of the jet pumps are 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. The head, net positive suction head (NPSH), flow, and efficiency curves are shown on Figure 5.4-3. Instrumentation and controls for the recirculation flow control system are described in Section 7.7.1.2.

The recirculated coolant consists of saturated water from the steam separators and dryers that has been subcooled by incoming feedwater. This water passes down the annulus between the reactor vessel wall and the core shroud. A portion of the coolant flows from the vessel, through the two external recirculation loops, and becomes the driving flow for the jet pumps. Each of the two external recirculation loops discharges high pressure flow into an external manifold from which individual recirculation inlet lines are routed to the jet pump risers within the reactor vessel. The remaining portion of the coolant mixture in the annulus becomes the suction 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 suction, are mixed in the jet pump throat section and result in partial pressure recovery. The balance of recovery is obtained in the jet pump diffuser (Figure 5.4-4). The adequacy of the total flow to the core is discussed in Section 4.4.

The allowable heatup rate for the recirculation pump casing is 100°F/hr, which 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; with the FCV in the minimum position, this permits the reactor pressure plus the active jet pump head to cause reverse flow in the idle loop.

The objective of the recirculation gate valve trim design is to minimize the need for maintenance of the valve internals. The valves have high-quality backseats that permit renewal or replacement of stem packing while the system is full of water.

The pump/motor operates at two speeds, 25 and 100 percent. When operation is at 25 percent speed, the power comes from the low frequency motor generator (LFMG) set which operates at 15 Hz. Power for 100-percent speed operation comes from a 60-Hz source.

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 (FCV and jet pumps). When the pump is operating at 100-percent speed, feedwater flow provides additional NPSH margin by providing subcooling. The subcooling temperature is measured by detectors that are provided in the recirculation lines, and by the steam dome delta pressure that is converted to temperature. The difference between these two readings is a direct measurement of the subcooling. If the subcooling falls below approximately 11°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 FCV.

The 100-percent speed trip to 25-percent speed is a step as far as the power supplies are concerned but linear as far as motor is concerned. The pump is tripped from 100-percent speed; it coasts

down to 25-percent speed when the LFMG comes on line and the pump is run at 25 percent.

When preparing for hydrostatic tests, the nuclear system temperature must be raised above the vessel NDTT limit.

The vessel is heated by core decay heat and/or by operating the recirculation pumps at 100-percent speed and FCVs at minimum position.

Each recirculation pump 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 pump shaft passes through a breakdown bushing in the pump casing to reduce leakage in the event of a gross failure of both shaft seals. The cavity temperatures and pressures of each individual are continuously monitored.

Each recirculation pump motor is a dual-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 FCV can throttle the discharge flow of the pump proportionally to an instrument signal. The FCV has 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 equipment that does not require removal from the system for rework or overhaul. Pump casing and valve bodies are designed for a 40-yr life and are welded to the pipe.

The pump drive motor, impeller, and wear rings and FCV internals are designed for a long operational life. Pump mechanical seal parts and the valve packing are expected to have a life expectancy that affords convenient replacement during the refueling outages.

The recirculation system piping is of all-welded construction. The effective ASME III Code for the recirculation piping system, based on the piping contract award and the requirements of NCA1140, is the 1977 Edition, Summer 1977 Addenda. No one contractor has overall responsibility for the complete scope, including design, procurement, material supply, and installation. Therefore, on the N-5 Data Report (modified), GE certifies that the design, procurement, and material supply activities performed by GE are in accordance with the ASME III Code; and RCI, the

system installer (NA Certificate Holder), certifies that the piping system was installed in accordance with the design specification and ASME III Code. SWEC, as the N Certificate Holder assuming overall responsibility for the balance of the plant, performed the pressure testing activities. The piping system designer (GE) and installer (RCI and its AIA, Hartford Steam Boiler) witnessed the pressure testing of those items for which they were responsible.

The reactor recirculation system pressure boundary equipment is designed as Category I equipment. As such, it is designed to resist sufficiently the response motion for the SSE 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 dynamic 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 have 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 protection against the dynamic effects associated with a postulated pipe rupture are contained in Section 3.6B. The recirculation system piping, valves, and pump casings are covered with thermal insulation having a total maximum heat transfer rate of 65 Btu/hr-sq ft with the system at rated operating conditions. This heat loss includes losses through joints, laps, pipe supports and restraints, and other openings that may exist in the insulation. The maximum heat transfer is based upon a recirculation system temperature of 550°F and a drywell temperature of 135°F.

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

5.4.1.4 Safety Evaluation

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

The core-flooding capability of a jet pump design plant is discussed in detail in the ECCS document filed with the NRC as a GE topical report[1]. The ability to reflood the BWR core to the top of the jet pumps is shown schematically on Figure 5.4-5 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 shown in Table 3.9B-2. 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.

Purchase specifications require that the recirculation pumps' first critical speed not be less than 130 percent of operating speed. Calculations performed by the vendor to conform to this specification were reviewed and approved by the NSSS vendor. 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 SSE. Calculations performed by the vendor to substantiate this were reviewed and approved by the NSSS vendor.

Pump overspeed occurs during the course of a LOCA due to blowdown through the pump in the broken loop pump. Design studies determined that the overspeed is not sufficient to cause destruction of the motor.

5.4.1.5 Inspection and Testing

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

Before the preoperational test program, the reactor recirculation system is hydrostatically tested at 125-percent reactor vessel design pressure. Preoperational tests on 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 and restraints are adjusted, as necessary, to assure that components are free to move as designed. Nuclear system responses to RPTs at rated temperatures and pressure are evaluated during the startup tests, and plant power response to recirculation flow control is determined.

5.4.2 Steam Generators (PWR)

Section 5.4.2 is not applicable to this Final Safety Analysis Report (FSAR).

5.4.3 Reactor Coolant Piping

The reactor coolant piping is discussed in Sections 3.9B.3.1.4 and 5.4.1. The recirculation loops are shown on Figures 5.4-1 and 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 MSL flow restrictors are designed:

1. 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 MSIVs.

2. To withstand the maximum pressure difference expected across the restrictor, following complete severance of a MSL.

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

4. To provide trip signals for MSIV closure.

5. In accordance with ASME Fluid Meters, Sixth Edition, 1971.

5.4.4.2 Description

A main steam flow restrictor (Figure 5.4-6) is provided for each of the four MSLs. The restrictor is a complete assembly welded into the MSL. It is located in the drywell. The restrictor limits the coolant inventory loss and loss rate from the reactor vessel in the event a MSLB occurs outside the containment. The loss is limited to a maximum (choke) flow of 7.08×10^6 lb/hr at 1,015 psig upstream pressure. The restrictor assembly consists of a venturi-type nozzle insert welded (in accordance with applicable code requirements), into the MSL.

The flow restrictor has no moving parts. Its mechanical structure can withstand the velocities and forces associated with a MSLB. The maximum differential pressure is conservatively assumed (by calculation) to be 1,265 psia.

The ratio of venturi throat diameter to steam line inside diameter of approximately 0.551 results in a maximum pressure

differential (unrecovered pressure) of about 10.10 psi at 100 percent of rated flow. This design limits the steam flow in a severed line to 165 percent of rated flow, yet it results in negligible increase in steam moisture content during normal operation. The restrictor is also used to measure steam flow to initiate closure of the MSIVs when the steam flow exceeds the preselected operational limits.

5.4.4.3 Safety Evaluation

In the event a MSL should break outside the containment, the critical flow phenomenon would restrict the steam flow rate in the venturi throat to 165 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 MSLB results in doses to the public that are a fraction of 10CFR50.67 limits.

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 approximately 176 ft/sec (steam piping ID) to approximately 575 ft/sec (steam restrictor throat). ASTM A351 (Type 304 Grade CF8) cast stainless steel was selected for the steam flow restrictor throat material, because it has excellent resistance to erosion and corrosion in a high-velocity steam atmosphere. The excellent performance of stainless steel in high-velocity steam is due to its resistance to erosion and corrosion. A protective surface film forms on the stainless steel, which prevents any surface attack, and this film is not removed by the steam.

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

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

5.4.4.4 Inspection and Testing

Because the flow restrictor forms a permanent part of the main steam piping and has no moving components, no testing program is planned beyond testing the main steam system piping. Only very slow erosion occurs with time, and such a slight enlargement has no safety significance. Stainless steel resistance to erosion has been substantiated by turbine inspections at the Dresden Unit 1 facility which 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 yr of operation the increase in restrictor choked flow rate would be no more than 5 percent. A 5-percent increase in the radiological dose calculated for the postulated MSLB accident is not significant.

5.4.5 Main Steam Isolation System

5.4.5.1 Safety Design Bases

The MSIVs, individually or collectively, will:

1. Isolate the MSLs within the time established by DBA analysis to limit the release of reactor coolant.

2. Isolate the MSLs slowly enough that simultaneous isolation of all steam lines does not induce transients that exceed the nuclear system design limits.

3. Isolate the MSL when required, despite single failure in either valve or in the associated controls, to provide a high level of reliability for the safety function.

4. Use separate energy sources as the motive force to isolate independently the redundant isolation valves in the individual steam lines.

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

6. Be able to isolate the MSLs, either during or after seismic and/or hydrodynamic loadings, to assure isolation if the nuclear system is breached.

7. Have capability for testing, during normal operating conditions, to demonstrate that the valves are functional.

5.4.5.2 Description

Two isolation valves are welded in a horizontal run of each of the four MSLs. One valve is as close as possible to the inside of primary containment and the other is just outside the containment.

Figure 5.4-7 shows a MSIV. Each is a 26-in, Y-pattern, globe valve. Design steam flow rate through each line is 4.409×10^6 lb/hr. The main disk or poppet is attached to the lower end of the stem. Normal steam flow tends to close the valve, and higher inlet pressure tends to hold the valve closed.

The bottom end of the valve stem closes a small pressure-balancing hole in the poppet. When the hole is open, it acts as a pilot valve to relieve differential pressure forces on the poppet. Valve stem travel is sufficient to give flow areas past the wide-open poppet approximately equal to the seat port area. The poppet travels approximately 90 percent of the valve stem travel; approximately the last 10 percent of travel closes the pilot hole. The air cylinder can open the poppet with a maximum differential pressure of 200 psi across the isolation valve in a direction that tends to hold the valve closed.

A 45° 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 rated flow is 10 psi maximum. The valve stem penetrates the valve bonnet through a stuffing box that has replaceable packing. To help prevent leakage through the stem packing, the poppet backseats when the valve is fully open.

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

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. (Spring closure of the valve due to loss of air supply pressure is assisted by air from an air tank accumulator directed to the top of the actuator cylinder). The motion of the spring seat member actuates a scram switch in the 85-percent open valve position and indicator light switches in the 90-percent open and 10-percent open valve positions.

The valve is operated by pneumatic pressure and by the action of compressed springs. The control unit is attached to the air cylinder. This unit contains air pilot valves and solenoid-operated valves (SOVs). The SOVs control opening and closing of the air valves and provide exercising capability at slow speed. Remote manual switches in the main control room enable the Operator to operate the valves.

Operating air is supplied to the outboard valves from the instrument air system, and to the inboard valves from the instrument nitrogen system (Section 9.3.1). An air accumulator between the control valve and a check valve provides backup operating air. The leak-tightness of the check valve is tested periodically to assure sufficient air pressure in the accumulator to assist in closing the valve on demand.

Each valve is designed to accommodate saturated steam at plant operating conditions. The valves are furnished in conformance with a design pressure and temperature rating in excess of plant operating conditions to accommodate plant accident and transient overpressure conditions.

In the worst case, if the MSL should rupture, steam flow would quickly increase to 165 percent of rated flow. Further increase is prevented by the venturi flow restrictor in the MSL inside 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-yr service at the specified operating conditions. The estimated operating cycles are estimated to be 50 to 400 cycles/yr (full open to full close and return).

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

Design specification ambient conditions inside the primary containment for normal plant operation are specified as 135°F average and 150°F maximum temperature and 90-percent maximum humidity. Design normal gamma plus neutron radiation dose over a 5-yr maintenance period is 7.4 x 10^6 rads.

The MSIVs (inside and outside the primary containment) are designed to close under accident environmental conditions of 340°F for 1 hr at -5 to +45 psig drywell pressure. In addition, they will not fail open under the following post-accident environmental conditions:

1. 340°F for an additional 2 hr at drywell pressure of -5 to +45 psig.

2. 320°F for an additional 3 hr at -5 to +45 psig.

3. 250°F for an additional 18 hr at -5 to +45 psig.

4. 200°F during the next 99 days at -5 to +45 psig.

5. Relative humidity: 20 to 100 percent.

6. Radiation: 1.5×10^8 rads gamma (100-day integrated accident dose).

The above parameters address post-accident environmental conditions under which the MSIVs are required to stay closed. An evaluation performed to address minimum requirements to ensure closure of MSIVs under accident conditions determined that a minimum accumulator pressure of 74 psig (in conjunction with spring action) is required to ensure closure of the MSIVs. The derivation of the 74 psig (minimum) accumulator pressure assumed 32 psig drywell pressure (from Figures 6.2-2 through 6.2-5), based on MSIV isolation for a DBA event initiated within 8 sec and MSIVs full closed within 5 sec. This ensures that the MSIVs will close under DBA containment pressure conditions and stay closed for the conditions identified above.

The MSIV systems are Category I equipment and the valves are designed, fabricated, inspected, and tested in accordance with ASME Section III, Safety Class 1. The valves are designed to be operable when subjected to various combinations of the following loads:

1. Operating base earthquake (OBE) and SSE.

2. A double-ended guillotine pipe break outside containment during plant operation at full power.

3. Worst-case loading imposed by the attached piping.

4. Suppression pool dynamic loads resulting from the discharge of SRVs and LOCA.

Operability, as used above, is defined as the ability of the valve system, when subjected to the described loadings, to close and remain closed.

The valve body and associated internal pressure boundary components are modeled using finite element techniques. Application of the described loadings results in stresses within the limits of ASME Section III, Subarticle NB-3500. Deformations were evaluated for worst-case conditions, for all regions where critical clearances or alignments might conceivably be compromised so as to jeopardize functional capability. Acceptable margins were determined for all such regions. A flaw was identified in the valve-to-pipe weld of MSIV 7A during the baseline examination of the replacement activities for this valve under ASME Section XI. This flaw was evaluated and meets the acceptance criteria of ASME Section XI, 1980 Edition through the Winter 1980 Addenda, Table IWB-3514-1.

All Class 1E electrical equipment of the MSIV system is qualified in accordance with IEEE-323-1974 and RG 1.89. Assurance of operability of the MSIV actuators and their control logic cabinets is demonstrated by a comprehensive dynamic testing program, in accordance with IEEE-344-1975 and RG 1.100. The ability of the operator to perform its safety function before, during, and after the tests, is demonstrated.

The qualification of all Class 1E electrical equipment of the MSIV system meets or exceeds the requirements for Category II qualification in accordance with NUREG-0588.

5.4.5.3 Safety Evaluation

In a direct-cycle nuclear power plant, the reactor steam goes to the turbine and to other equipment outside the containment.

Radioactive materials in the steam can be released to the environs through process openings (leaks) in the steam system, or escape from pipe breaks. A large break in the steam system can drain the water from the reactor core faster than it is replaced by feedwater. The MSIVs are provided to limit both the release of radioactive material and the drainage of water from the reactor due to a MSLB outside containment.

The analysis of a complete, sudden steam line break outside the containment is described in Chapter 15. The analysis shows that the fuel barrier is protected against loss of cooling if MSIV closure is within specified limits, including instrumentation delay to initiate valve closure after the break. The calculated radiological effects of the radioactive material postulated to be released following a MSLB are presented in Section 15.6.4.

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

The ability of this 45-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-in valve was tested in a range of steam-water blowdown conditions simulating postulated accident conditions.[3]

The following specified hydrostatic, leakage, and stroking tests, as a minimum, are performed by the valve manufacturer in shop tests:

1. To verify its capability to close between 3 and 10 sec, each valve is tested at 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.

2. Leakage is measured with the valve seated and backseated. The specified maximum seat leakage, using cold water at design pressure, is 2 cm^3/hr/in of nominal valve size. In addition, an air seat leakage test is conducted using 50 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 full open to full closed and return to open position, and the packing leakage still must be zero by visual examination.

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

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

After the valves are installed in the MSLs, 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. The inside valve, the outside valve, and their respective control systems are separated physically. The design of the isolation valve system has been analyzed for earthquake and suppression pool dynamic loading. The stress caused by the specified dynamic loading on the actuator does not exceed material allowables or prevent the valve from closing as required.

Electrical equipment that is associated with the isolation valves and is required to operate in an accident environment is limited to the wiring, terminal blocks, wire lugs, relays, solenoid valves, trip solenoids, and position switches.

The expected pressure and temperature transients following an accident are discussed in Chapter 15. All these components are tested for their specified performance and qualified under the specified environmental conditions.

5.4.5.4 Inspection and Testing

The MSIVs will be functionally tested for operability during plant operation and refueling outages. The test provisions are as follows:

1. During a refueling outage, the MSIVs can be functionally tested, leak tested, and visually inspected.

2. During plant operation the MSIVs can be tested and exercised individually to a partially closed position without a significant effect on plant operation.

3. The MSIVs can also be tested and exercised individually to the fully closed position in accordance with the requirements of the Technical Specifications.

Leakage from the valve stem packings can be detected visually during shutdown. The leak rate through the valve seats can be measured accurately during shutdown.

During prestartup tests following an extensive shutdown, the valves receive the same hydrostatic tests as those imposed on the primary system. Such a test and leakage measurement program ensures that the valves are operating correctly and that a leakage trend is detected.

5.4.6 Reactor Core Isolation Cooling System

5.4.6.1 Design Bases

The RCIC system is a safety system that 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 the reactor fuel from overheating in the event that:

1. The vessel is isolated and maintained in the hot standby condition.

2. The vessel is isolated in conjunction with the loss of coolant flow from the reactor feedwater system.

3. A complete plant shutdown occurs under conditions of loss of normal feedwater system before the reactor is

depressurized to a level where the shutdown cooling system can be placed into operation.

Following a reactor scram, steam generation continues at a reduced rate due to core fission product decay heat. At this time, the turbine bypass system diverts the steam to the main condenser, and the feedwater system supplies the makeup water required to maintain 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 drops due to continued steam generation by decay heat. Upon reaching a predetermined low level, the RCIC system is initiated automatically. The turbine-driven pump supplies demineralized makeup water from the condensate storage tank (CST) to the reactor vessel. An alternate source of water is available from the suppression pool. The RCIC system allows automatic switchover of pump suction from the CST to the suppression pool if the RCIC pump suction pressure falls to a preset low level. Two level transmitters are used to detect low pressure for the RCIC pump suction. If either transmitter senses low pressure (indicating low CST level), pump suction is automatically transferred to the suppression pool. The turbine is driven with a portion of the decay heat steam from the reactor vessel and exhausts to the suppression pool. Suppression pool water is not maintained demineralized and is used only in the event all sources of demineralized water have been exhausted.

If the main feedwater system is not operable, a reactor scram is automatically initiated when reactor water level falls to Level 3. The Operator can then remotely manually initiate the RCIC system from the main control room, or the system is automatically initiated as follows. Reactor water level continues to decrease due to boiloff until Level 2 is reached. At this point, the HPCS and the RCIC systems are automatically initiated to supply makeup water to the RPV. These systems continue automatic injection until the reactor water level reaches Level 8, at which time the HPCS injection valve is closed and the RCIC steam supply valve is closed.

In the nonaccident case, the RCIC system is normally the only makeup system used to furnish subsequent makeup water to the RPV. The Operator remotely manually shuts down the HPCS system from the main control room. When level reaches Level 2 again due to loss of inventory through the main steam relief valves or to the main condenser, the RCIC system automatically restarts as described in Section 1.10.II.K.3.13. This system then maintains the coolant makeup supply. RPV pressure is regulated by the automatic or remote manual operation of the main steam relief valves which discharge to the suppression pool.

To remove decay heat during a planned isolation event, assuming that the main condenser is not available, the SRVs can be used to dump the residual steam to the suppression pool. The suppression pool will then be cooled by remote manual alignment of the RHR system in the suppression pool cooling mode which routes the pool water through the RHR heat exchangers, cools it, and returns it to the suppression pool in a closed cycle. Makeup water to the RPV is still supplied by the RCIC system.

For the accident case with the RPV at high pressure, the HPCS system can also be used to automatically provide the required makeup flow. No manual operations are required. If the HPCS system is postulated to fail at these conditions and the RCIC capacity is insufficient, the ADS will automatically initiate depressurization of the RPV to permit the condensate pumps or the low-pressure ECCS (LPCI and LPCS) to provide makeup coolant.

Whenever the RCIC system is initiated, the large steam turbine generator (LSTG) turbine is tripped to prevent water induction into the turbine, and the control room is alarmed that the RCIC injection valve is open.

Therefore, although manual actions can be taken to mitigate the consequences of a loss of feedwater, there are no short-term manual actions which must be taken. Sufficient systems exist to automatically mitigate these consequences.

During RCIC operation, the suppression pool acts as the heat sink for steam generated by reactor decay heat. This results in a rise in pool water temperature. Heat exchangers in the RHR system are used to maintain the pool water temperature within acceptable limits by cooling the water directly.

The RCIC system is equipped with a discharge line fill pump that operates to maintain the pump discharge line in a filled condition. Keeping the discharge line filled reduces the lag time between pump startup and attainment of full flow to the RPV. Additionally, its operation eliminates the possibility of RCIC pumps discharging into a dry pipe and minimizes water hammer effects. The fill pump is classified as Category I and Safety Class 2. The pump motor is Class 1E and is powered from a Class 1E source. Indication of pump operating status is provided in the main control room. Low discharge line pressure is also indicated in the main control room.

Pump discharge pipe routing and valve locations inside and outside containment ensure that a maximum amount of piping is maintained full of water.

In addition to the fill pump, the RCIC water discharge line is designed to accommodate water hammer loads due to postulated voids in the piping between the reactor and injection valve (MOV126). This section of piping is normally isolated from the fill pump circuit by the isolation valve. To fill this section

of piping between the injection valve and the check valve (V157) near the reactor, a bypass line around the injection valve has been provided to facilitate a manual fill to limit voiding in that section of pipe. Also, appropriate drain and vent lines are provided in the discharge line.

In addition to the physical design features described above, the RCIC pump discharge piping has been analyzed and conservatively designed for the effects of possible water hammer forces using the methods described in Appendix 3A, Section 3A.21.

5.4.6.1.1 Residual Heat Removal and Isolation

Residual Heat Removal

The RCIC system initiates and discharges flow to the reactor vessel (Figure 5.4-10) over a 165 to 1,215 psia pressure range. The temperature of RCIC water discharged into the reactor vessel varies from 40° to 140°F when using water from the CST. The mixture of the cool RCIC water and the hot steam does the following:

1. Quenches the steam.

2. Removes reactor residual heat.

3. Replenishes reactor vessel inventory.

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

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

1. High pressure drop across a flow device in the steam supply line equivalent to 300 percent of the steady state steam flow (with a time delay for TMI modification) at the reactor high-pressure steam condition.

2. High area temperature, utilizing temperature switches as described in the LDS. High area temperature is alarmed in the main control room.

3. Low reactor pressure of 50 psig minimum.

4. High pressure between the RCIC turbine exhaust rupture diaphragms.

These redundant leak detection devices, activated by the redundant power supplies, automatically isolate the steam supply to the RCIC turbine. Other isolation bases are defined in the following section. The HPCS provides redundancy for the RCIC should the RCIC become isolated, hence providing single-failure protection.

Isolation

Isolation valve arrangements include the following:

1. Two RCIC lines penetrate the RCPB. The first is the RCIC steam line which branches off one of the MSLs between the reactor vessel and the MSIV. This line has two automatic motor-operated isolation valves, one located inside and the other outside the primary containment. An automatic motor-operated inboard RCIC isolation bypass valve is used to equalize the line pressure across the inboard isolation valves and warm up the downstream line. The isolation signals noted earlier close these isolation valves.

2. The second RCIC line that penetrates the RCPB is the RCIC pump discharge line, which has two check valves (one inside the primary containment and the other outside). Additionally, an automatic MOV, in parallel with a manual locked closed valve, is located outside primary containment.

3. The RCIC turbine exhaust line vacuum breaker system line has two automatic MOVs 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 is automatic via a combination of low reactor pressure and high drywell pressure. The automatic isolation signals energize a 70-sec time delay to slow automatic closure of MOVs 2ICS*MOV148 and 2ICS*MOV164. The time delay is provided to ensure sufficient time to fully condense the exhaust steam in the RCIC turbine exhaust line. The vacuum breaker valve complex is placed outside the primary containment where there is a more desirable environment. In addition, the valves are readily accessible for maintenance and testing.

4. The RCIC pump suction line, minimum flow pump discharge line, and turbine exhaust line all penetrate the primary containment and are submerged in the suppression pool. The isolation valves for these lines are all outside the primary containment and require remote-manual operation, except for the minimum flow valves that actuate automatically.

5.4.6.1.2 Reliability, Operability, and Manual Operation

Reliability and Operability

The RCIC system (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 baseline for system reliability. To confirm that the system maintains this line, functional and operability testing is performed at predetermined intervals throughout the life of the reactor plant in accordance with Technical Specifications.

A design flow functional test of the RCIC system may be performed during normal plant operation by drawing suction from the CST and discharging through a full flow test return line to the CST. 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, except valves 2ICS*V156 and 2ICS*V157, which are tested during reactor shutdown. System control provides automatic return from test to operating mode if system initiation is required, with the following three exceptions:

1. Auto/manual initiation on the flow controller is required for Operator flexibility during system operation.

2. Closure of either or both of the steam inside/outside isolation valves requires Operator action to properly sequence their opening. An alarm sounds when either of these valves leaves the fully-open position.

3. Other bypassed or otherwise deliberately rendered inoperable parts of the system are automatically indicated in the main control room at the system level.

To demonstrate proper system response, RCIC may be periodically initiated with flow through the discharge valve to the head spray cooling nozzle during reactor operation, with the main turbine shut down and reactor power at approximately 10 to 15 percent.

Manual Operation

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

5.4.6.1.3 Loss of Offsite Power

The RCIC system power is derived from an emergency auxiliary power distribution system that is normally energized from offsite

power sources. Upon loss of offsite power (LOOP), this is automatically energized from standby onsite power sources (diesel generator or battery). All components necessary for initiation of the RCIC system are capable of startup independent of auxiliary ac power, plant service air, and external cooling water systems.

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 turbine and pump are located in a different quadrant of the reactor building and utilize different divisional power (and separate electrical routings) than that of its redundant system HPCS (Sections 5.4.6.1.1 and 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 takes suction from the CSTs during normal modes of operation. The CSTs are located within the condensate storage building which is maintained at a minimum temperature of 65°F, as described in Section 9.4.7.2.5.

All interconnecting piping is located within piping tunnels which are beneath heated structures or below the frost line. To provide a Category I source of cooling water for the RCIC system, automatic transfer circuitry has been provided to transfer suction from the CSTs to the suppression pool, which is inside the reactor building and protected from cold weather. All other RCIC piping is located within the reactor building and is protected from cold weather.

5.4.6.2 System Design

5.4.6.2.1 General

<u>Description</u>

A summary description of the RCIC system is presented in Section 5.4.6.1, which defines in general the system functions and components. The detailed description of the system, its components, and operation is presented in the following sections.

<u>Diagrams</u>

The following diagrams are included for the RCIC systems:

1. A schematic P&ID (Figure 5.4-9) shows all components, piping, points where interface system and subsystems tie together, and instrumentation and controls associated with subsystem and component actuation.

2. A schematic process diagram (Figure 5.4-10) shows temperature, pressures, and flows for RCIC operation and system process data hydraulic requirements.

3. Performance curves showing temperature, pressure, steam flow, brake horsepower, and shaft speed for the RCIC turbine manufactured by the Terry Corporation, are shown on Figures 5.4-10a and 5.4-10b.

<u>Interlocks</u>

The following defines the various electrical interlocks:

1. There are three keylocked switches controlling valves F063, F064, F068 (2ICS*MOV128, 2ICS*MOV121, 2ICS*MOV122), and two keylocked reset switches that reset the isolation signal seal-in feature.

2. The F031 (2ICS*MOV136) limit switch activates when fully open and closes F010 (2ICS*MOV129), F022 (2ICS*FV108), and F059 (2ICS*MOV124).

3. The F068 (2ICS*MOV122) limit switch activates when fully open and clears F045 (2ICS*MOV120) permissive so F045 (2ICS*MOV120) can open.

4. The F045 (2ICS*MOV120) limit switch activates when F045 (2ICS*MOV120) is not fully closed and energizes a 25-sec time delay for alarms for low pump discharge flow, low turbine bearing oil pressure, and low gland seal air pressure, also energizing a 10-sec time delay which closes F019 (2ICS*MOV143), initiates startup ramp functions, and alarms low water leg pump discharge pressure. This ramp resets each time F045 (2ICS*MOV120) is closed.

5. The F045 (2ICS*MOV120) limit switch activates when fully closed; this permits F004 (2ICS*AOV109), F005 (2ICS*AOV110), F025 (2ICS*AOV131), and F026 (2ICS*AOV130) to open and closes F013 (2ICS*MOV126). When the limit switch opens it activates a 10-sec time delay to drop out the coil causing the F004 (2ICS*AOV109), F005 (2ICS*AOV110), F025 (2ICS*MOV131), and F026 (2ICS*AOV130) to close and open F013 (2ICS*MOV126).

6. The turbine trip throttle valve limit switch activates when fully closed and closes F013 (2ICS*MOV126) and F019 (2ICS*MOV143).

7. The steam line isolation valves F063 (2ICS*MOV128) and F064 (2ICS*MOV121) are closed by RCIC Division I and II

isolation signals. These divisions are different from the ECCS divisions.

8. High turbine exhaust pressure, low pump suction pressure, or an isolation signal actuate and close the turbine trip throttle valve. When the signal is cleared, the trip throttle valve must be reset from the main control room.

9. Overspeed of 116.2 percent trips the mechanical trip at the turbine which closes the trip throttle valve. The mechanical trip is reset at the turbine.

10. An isolation signal closes F063 (2ICS*MOV128), F064 (2ICS*MOV121), F076 (2ICS*MOV170), and other valves as noted in Items 6 and 8.

11. An initiation signal opens F045 (2ICS*MOV120), F010 (2ICS*MOV129), if closed, and F013 (2ICS*MOV126), and closes F059 (2ICS*MOV124) if open. The initiation signal causes F022 (2ICS*FV108) to receive a close signal; however, this is not sealed in. Valve F059 (2ICS*MOV124) closes and seals in. An initiation signal also trips the LSTG turbine.

12. High and low inlet RCIC steam line drain pot levels, respectively, and open and close F054 (LV132).

13. The combined signal of low flow plus high pump discharge pressure opens and with increased flow closes F019 (2ICS*MOV143) (Items 4 and 6).

14. The F013 (2ICS*MOV126) limit switch activates when F013 (2ICS*MOV126) is not fully closed and energizes a control room relay to alarm the control room that valve F013 (2ICS*MOV126) is not fully closed.

15. A LOCA signal prevents turbine trip and throttling valve motor operation.

5.4.6.2.2 Equipment and Component Description

<u>Design Conditions</u>

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

1. One 100-percent capacity turbine and accessories.

2. One 100-percent capacity pump assembly and accessories.

3. Piping, valves, and instrumentation for:

a. Steam supply to the turbine.

b. Turbine exhaust to the suppression pool.

c. Supply from the CST to the pump suction.

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

e. Pump discharge to the head cooling spray nozzle, including a test line to the CST, a minimum flow bypass line to the suppression pool, and a coolant water supply to the turbine lube oil cooler.

4. System pressure pump to maintain injection lines full of water to the outside containment isolation valves.

The basis for the design conditions is ASME Section III.

Design Parameters

Design parameters for the RCIC system components are listed as follows (see Figure 5.4-9 for a cross-reference of component numbers):

1. RCIC Pump Operation (C001)

Flow rate	Injection flow - 600 gpm Cooling water flow - 60 gpm Total pump discharge - 660 gpm (includes no margin for pump wear)
Water temperature range	40° to 140°F
NPSH	23 ft minimum
Developed head (Required)	≤3,080 ft @ 1,215 psia reactor pressure 610 ft @ 165 psia reactor pressure
BHP, not to exceed	750 hp @ 3,080 ft developed head 130 hp @ 610 ft developed head
Design pressure	1,525 psig
Design ambient temperature	60° to 122°F

2. RCIC Turbine Operation (C002)

	H.P. Condition (psia)	L.P. Condition (psia)

Reactor pressure (saturated temperature)	1,215	165
Steam inlet pressure	1,190, min	150, min
Turbine exhaust pressure	65, max	65, max
Turbine design inlet pressure	1,250 psig at saturated temperature	
Turbine exhaust casing design pressure	165 psig at saturated temperature	

3. RCIC Orifice Sizing

Coolant loop orifice (D012)	Sized to maintain a minimum of 16 gpm to a maximum of approximately 40 gpm to the lube oil cooler based upon pump suction line pressure varying from 50 psig to minimum NPSH (and the PCV operating within its normal band).
Minimum flow orifice (D005)	Sized with piping arrangement to ensure minimum flow of 75 gpm with MO-F019 (MOV143) fully open.
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.
Leakoff orifices (D008, D010)	Sized for 1/8-in diameter minimum, 3/16-in diameter maximum.

4. Valve Design Requirements

The following are the design differential pressure requirements. However, the actuator sizing/setting is based on maximum operating differential pressure.

Steam supply valve (F045) (MOV120)	Open and/or close against full differential pressure of 1,200 psi within 15 sec.
Pump discharge valve (F013) (MOV126)	Open and/or close against full differential pressure of 1,450 psi within 20 sec.

Pump minimum flow bypass valve (F019) (MOV143)	Open and/or close against full differential pressure of 1,450 psi within 10 sec.
RHR steam supply isolation valves (F063 & F064) (MOV128 and MOV121)	Open and/or close against full differential pressure of 1,200 psi within 25 sec.
Cooling water pressure control valve (F015) (PCV115)	Air-operated valve capable of maintaining constant downstream pressure of 125 psia.
Pump suction relief valve (F017) (RV114)	150 psig relief setting; 10 gpm at 10 percent accumulation.
Cooling water relief valve (F018) (RV112)	Sized to prevent overpressurizing piping, valves, and equipment in the coolant loop in event of failure of pressure control valve F015.
Pump test return valve (F022) (FV108)	Capable of throttling against 1,000 psi differential pressure and closure against differential pressures of 1,450 psi.
Pump suction valve, suppression pool (F031) (MOV136)	Located outside as close as practical to the primary containment.
Check valves (F065-F066) (V156+V157)	System test mode bypasses this valve, and its functional capability is demonstrated separately. Therefore, valve test provisions are provided, including limit switches for V156 to indicate disc movement. The valve and valve-associated equipment are capable of proper functional operation during maximum ambient conditions.
Warmup line isolation valve (F076) (MOV170)	Opens and/or closes against differential pressure of 1,200 psi within 9 sec.
Vacuum breaker valves (F080, F086) (MOV164,MOV148)	Opens and/or closes against differential pressure of 160 psi at a minimum rate of 4 in/min.
5. Rupture Disc Assemblies	Utilized for turbine casing protection; includes a mated

	(D001 & D002)	vacuum support to prevent rupture disc reversing under vacuum conditions.
	Rupture pressure flow capacity	150 ±10 psig. 60,000 lb/hr at 165 psig.
6.	Instrumentation	For instruments and control definition refer to Chapter 7.
7.	Condensate Storage Requirements	Total required reserve storage for RCIC and HPCS systems is 135,000 gal.
8.	Piping RCIC Water Temperature	The maximum water temperature range for continuous system operation will not exceed 140°F. However, due to potential short-term operation at higher temperatures, piping expansion calculations were based on 170°F.
9.	Turbine Exhaust Vertical Reaction Force	The turbine exhaust sparger is capable of withstanding a vertical pressure unbalance of 20 psi. Pressure unbalance is due to turbine steam discharge below the suppression pool water level.

10. Ambient Conditions

	Temperature (°F)	Relative Humidity (%)
Normal plant operation	60-100	95
Isolation conditions (Isolation of the primary system requiring RCIC operation)	150	100

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 Section III, Safety Class 1. The RCIC system component classifications and those for the condensate storage system are given in Table 3.2-1.

5.4.6.2.4 System Reliability Considerations

To assure that the RCIC operates when necessary and in time to prevent inadequate core cooling (ICC), 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.

The most limiting operating condition for the RCIC pump occurs when the pump takes suction from the suppression pool and discharges at its rated flow of 660 gpm. This represents the limiting operating condition because of the minimum static suction head (17.4 ft) and the maximum temperature/vapor pressure (170°F/6.0 psia) of the water that might exist during RCIC system operation. The NPSH margin during this condition is 5.46 ft (NPSH available = 28.46; NPSH required = 23 ft). The RCIC system meets the requirements of RG 1.1, since the calculation of NPSH available takes no credit for increased containment atmospheric pressure accompanied by a LOCA, and is computed using the maximum anticipated water temperature of 170°F.

<u>Physical Independence</u>

The HPCS and RCIC systems are located in separate areas of the secondary containment. Piping runs are separated, and the water delivered from each system enters the reactor vessel via different nozzles.

<u>Prime Mover Diversity and Independence</u>

Prime mover independence is achieved by using a steam turbine to drive the RCIC pump and an electric motor-driven pump for the HPCS system. The HPCS motor is supplied from emergency ac power or a separate diesel generator.

<u>Control Independence</u>

Control independence of HPCS and RCIC is provided by using different battery systems to provide control power to each unit. Separate detection initiation logics are also used for each system.

Portions of HPCS and RCIC within the RCPB are designed to meet Safety Class 1 requirements. Environment in the equipment rooms is maintained by separate auxiliary systems.

<u>Periodic Testing</u>

A design flow functional test of the RCIC can be performed during plant operation (Section 5.4.6.1.2). Periodic inspections and maintenance of the turbine-pump unit are conducted in accordance with Station procedures. Valve position indication and instrumentation alarms are displayed in the main control room.

5.4.6.2.5 System Operation

Automatic startup of the RCIC system due to an initiation signal from reactor low water level requires no Operator action. The test operation mode is manually initiated by the Operator. The Operator actions associated with these modes are defined in the operating procedures.

The most limiting single failure with the RCIC system and its HPCS backup system is the failure of the HPCS. With a HPCS failure, if the capacity of the RCIC system is adequate to maintain reactor water level, the Operator follows specific procedures to facilitate the automatic operation. If, however, the RCIC capacity is inadequate, the same procedures still apply, but the Operator may also initiate the ADS (Section 6.3.2).

Operation of the RCIC system following a Station blackout is addressed in Section 8.3.1.5.

5.4.6.3 Performance Evaluation

The analytical methods and assumptions in evaluating the RCIC system are presented in Chapter 15 and Appendix 15A. The RCIC system provides the flows required from the analysis (Figure 5.4-10) based upon considerations noted in Section 5.4.6.2.4. If the piping downstream of the injection valve is void, the water injection into the reactor vessel will be delayed. However, the water level in the reactor vessel will be above the top of active fuel (TAF) and will avoid a reactor vessel Level 1 (L1) trip.

5.4.6.4 Preoperational Testing

The preoperational and initial startup test program for the RCIC system is presented in Chapter 14.

5.4.7 Residual Heat Removal System

5.4.7.1 Design Bases

The RHR system is composed of three independent loops, each containing a 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 either the reactor vessel via a separate nozzle, or back to the suppression pool via a full-flow test line. The A and B loops have heat exchangers that are cooled by service water. Loops A and B can also take suction from the reactor recirculation system suction and can discharge into the reactor recirculation discharge or to the suppression pool and drywell spray spargers. In addition, Loops A and B take suction from the fuel pool and discharge to the fuel pool cooling discharge.

5.4.7.1.1 Functional Design Basis

The RHR system has four subsystems, each of which has its own functional requirements. Each subsystem is discussed separately as follows.

Residual Heat Removal Mode (Shutdown Cooling Mode)

The functional design basis of the shutdown cooling mode is to have the capability to remove decay and sensible heat from the reactor primary system so that the reactor outlet temperature is reduced to 125°F, in approximately 20 hr after the control rods have been inserted, to permit refueling when the service water temperature is 67°F, the core is "mature," and the RHR heat exchanger tubes are assumed to be 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 maximum cooldown rate of 100°F/hr with both loops in service. However, the flushing operation associated with normal initiation of the shutdown cooling mode prevents attaining 212°F coolant temperature at the minimum time.

If flushing is performed in 2 hr, the minimum time required to reduce vessel coolant temperature to 212°F is depicted on Figure 5.4-11.

The design basis for the most limiting single failure for the RHR system (shutdown cooling mode) is that the shutdown line can be made usable by manual action (Section 15.2.9) and the plant is then shut down using the capacity of a single RHR heat exchanger and related service water capability. Figure 5.4-12 shows the time required to reduce vessel coolant temperature to 212°F using one RHR heat exchanger and allowing 2 hr for flushing.

In the event that the RHR shutdown cooling suction line is not available because of single failure, the alternate shutdown cooling method may be used to accomplish the shutdown cooling function as discussed in Section 15.2.9. This alternate shutdown cooling path uses the RHR and ADS/SRV systems. The RHR pump flow is directed to the RPV from the suppression pool through the RHR heat exchanger via the LPCI lines. A sufficient number of SRVs are powered open to establish a liquid flow path back to the suppression pool.

Alternatively, when the Operator is using EOPs to control RPV parameters and shutdown cooling is not available, the Operator is permitted to continue cooldown using the systems previously used for depressurization. These systems include SRVs, MSL drains, RWCU, and RCIC.

Further operational description of the alternate shutdown cooling method is discussed in Section 15.2.9. The adequacy of the SRVs for liquid flow in this mode of operation is discussed in Section 1.12.

Design calculations demonstrate that, at a flow of 982 lbm/sec, the RHR pumps have sufficient head to satisfy the requirement of the alternate shutdown cooling mode of operation.

Calculations also demonstrate that sufficient head exists to ensure the return of the water from the RPV to the suppression pool with four SRVs open, even with no credit taken for any pressure head which may exist within the RPV.

Low-Pressure Coolant Injection Mode

The functional design basis for the LPCI mode is to provide cooling to the reactor core only when the reactor vessel pressure is low, as is the case for large LOCA break sizes. The head flow characteristics assumed in the LOCA analyses for the LPCI pumps are shown on Figure 6.3-5a. The ECCS initiation signals are shown in Table 6.3-1.

Suppression Pool Cooling Mode

The functional design basis for the suppression pool cooling mode (SPCM) is that it will have the capacity to ensure that the suppression pool temperature, upon manual initiation after a blowdown or isolation event, does not exceed design limits.

Containment Spray Cooling Mode

The functional design basis for the containment spray cooling mode is that there should be two redundant means to spray into the drywell and suppression pool vapor space to reduce internal pressure to below design limits.

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 (Section 5.4.7.1.3). In addition, automatic isolation may occur as a result of low reactor water level which is unrelated to line pressure rating (see Section 5.2.5 for an explanation of the LDS and the isolation signals).

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

Possible Operator errors during plant startup and cooldown when the RHR system is not isolated from the RCS have been minimized through the implementation of the following design bases:

1. The low-pressure suction piping is protected from inadvertent opening of valves F008 (2RHS*MOV113) and F009 (2RHS*MOV112) by pressure interlocks on these valves.

2. The probability of draining some reactor water to the suppression pool is reduced by the existence of an interlock on valve F006 (2RHS*MOV2) which prevents its opening unless valves F004 (2RHS*MOV1), F024 (2RHS*FV38), and F027 (2RHS*MOV33) are closed.

3. The pump cannot be started unless a suction path is open.

5.4.7.1.3 Design Basis for Pressure Relief Capacity

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

1. Thermal relief only.

2. Valve bypass leakage only.

3. Control valve failure and the subsequent uncontrolled flow which results.

Transients are treated by bases 1 and 3; basis 2 results from an excessive leak past isolation valves. E12-F005 (2RHS*RV110), F025 (2RHS*RV20), F030 (2RHS*RV139), F088 (2RHS*RV61), F231 (2RHS*RV152), 2RHS*RV56, 2RHS*RV42, and F236 (2RHS*RV117) are set at the design pressure specified in the process data drawing, with allowance for piping elevation.

Redundant interlocks prevent opening valves to the low-pressure suction piping when the reactor pressure is above the shutdown range. These same interlocks initiate valve closure on increasing reactor pressure. A pressure interlock prevents connecting the discharge piping to the primary system whenever the pressure difference across the discharge valve is greater than the design differential. In addition, a high-pressure check valve closes to prevent reverse flow if the reactor pressure should increase. Relief valves in the discharge piping are sized to account for leakage past the check valves and motor-operated injection valves.

The relief and safety valve capacities and settings are shown in Table 5.4-2. All relief and safety valves are purchased, maintained, and installed to Safety Class 1, 2, or 3 requirements to match the requirements of the piping and equipment to which they are installed. All valves discharge to the suppression pool with the exception of E12-F231 (2RHS*RV152), located on the shutdown cooling suction line; 2RHS*RV42A and B, located on the RHR heat exchanger tubeside; and E12-F236 (2RHS*RV117), located

on the RHR pressure pump discharge piping. These valves discharge to the reactor building equipment drain system.

With the exception of the abandoned steam SRVs (E12-F055 and E12-F230) discharge piping, all pressure relief valve discharge lines in the RHR system are designed to accommodate the dynamic loading resulting from relief valve actuation. For a discussion of the analyses used to verify the design adequacy of these discharge lines, see Section 3.9A.1.5.2.

5.4.7.1.4 Design Basis for Reliability and Operability

The design basis for the shutdown cooling mode of the RHR system is that this mode is controlled by the Operator from the control room. The only operations performed outside the control room for a normal shutdown are certain manual valve lineups and system flushings.

Two separate shutdown cooling loops are provided. Although both loops are normally used for shutdown, the reactor coolant can be brought to 212°F in less than 20 hr after control rod insertion with only one loop in operation. A single RHR suction line can supply either or both shutdown cooling loops. With the exception of the shutdown cooling suction and shutdown cooling return, the entire RHR system is part of the ECCS and containment spray and suppression pool cooling system. It is designed with the redundancy, flooding protection, piping protection, power separation, and other features required of such systems (see Section 6.3 for an explanation of the design bases for the ECCS). Shutdown cooling suction and discharge valves are provided with both offsite and standby emergency power supplies for purposes of isolation and shutdown. The power supply to the suction supply valves will be de-energized during normal plant operation and will be administratively controlled due to main control room Appendix R fire concerns. In the event either of the two shutdown cooling supply valves fails to operate, an Operator is sent to open the valve manually. If this is not feasible, the shutdown line is isolated using manual valve E12-F020, and repairs are made to the shutdown cooling valves so that they can be opened to supply shutdown cooling suction to the RHR pumps. While repairs are in process, residual heat is absorbed by the main condenser or by the suppression pool, which is cooled by the RHR system. In the event that the RHR shutdown cooling suction line is not available because of single failure, alternate shutdown cooling methods may be used to accomplish the shutdown cooling function (see Section 5.4.7.1.1). Thus, no single failure in either system design or power source will result in the loss of shutdown cooling capability, because the plant may use the normal shutdown cooling through the recirculation loops, or the alternate shutdown cooling using the ADS SRVs and suppression pool cooling.

A loop of RHR may be placed in shutdown cooling on the same loop as a reactor recirculation pump operating in slow speed. Since a

reactor recirculation pump initially starts in fast speed then automatically downshifts to slow speed, a shutdown cooling loop must be secured prior to starting a reactor recirculation pump on the same loop.

The RHR system takes suction from either the recirculation piping or the suppression pool. All piping is within the reactor building and is protected from cold weather.

The RHR heat exchangers dissipate their heat to the service water system. All service water piping and components supplying the RHR heat exchangers are either within heated structures or underground piping tunnels located below the frost line. Design provisions which protect water in the service water intake and discharge system from freezing are described in Section 9.2.5.

5.4.7.1.5 Design Basis for Protection from Physical Damage

Pumps A, B, and C, as well as heat exchangers A and B, are physically separated. Each is housed in a separate cubicle. Pump A and heat exchanger A are in separate cubicles in the reactor building auxiliary bay north. Pumps B and C and heat exchanger B are in separate cubicles in the reactor building auxiliary bay south. The RHR system pressure pump P2, which maintains the loop B and C discharge header full of water and pressurized, is located in the same cubicle as pump C. The same function is provided by LPCS system pressure pump P2 for RHR loop A, located in a separate cubicle with LPCS pump P1.

The design basis for protection from physical damage, such as flooding, internally-generated missiles, pipe break, and seismic effects, is discussed in Sections 3.4, 3.5, 3.6B, and 3.7B, respectively.

5.4.7.2 System Design

5.4.7.2.1 System Diagrams

All components of the RHR system are shown on Figure 5.4-13. A description of the controls and instrumentation is presented in Sections 7.3.1.1.4 and 7.6.1.2.

Figure 5.4-14 is the RHR process diagram and data. All sizing modes of the system are shown in the process data. The flow values were used for original pipe and component sizing and do not represent design basis flow requirements. The functional control diagram (FCD) for the RHR system is provided on Figure 7.3-6.

Interlocks are provided: 1) to prevent draining vessel water to the suppression pool, 2) to prevent opening the RHR shutdown cooling suction valve if vessel pressure is above the RHR suction line design pressure, or the discharge line design pressure with the pump at shutoff head, 3) to prevent inadvertent opening of

drywell spray valves while in shutdown, and 4) to prevent pump start when suction valve(s) are not open.

The RHR system is connected to higher-pressure piping at shutdown suction, shutdown return, LPCI injection, head spray, and heat exchanger steam supply lines. The vulnerability to overpressurization of each location is discussed in the following paragraphs.

Shutdown suction has two gate valves in series, F008 and F009, that have independent pressure interlocks to prevent opening at high inboard pressure for each valve. The pressure interlock setpoint for RHR shutdown suction valves F008 and F009 is 128 psig. Shutdown suction valves F006A and B are not interlocked because valves F008 and F009 provide the required overpressurization protection. No single-active failure nor Operator error will result in overpressurization of the low-pressure piping.

In the event of leakage past F008 and F009, pressure transmitter N057 provides indication and alarm to the Control Room Operator.

The shutdown return line has a swing check valve, F050, to protect it from overpressurization. Additionally, a globe valve, F053, is located in series and has pressure interlock to prevent opening at high inboard pressures. No single-active failure nor Operator error with the shutdown return line will cause overpressurization of the lower-pressure piping.

LPCI injection valves F042A, B, and C, along with LPCI injection line check valves F041A, B, and C, provide overpressurization protection for the low-pressure RHR discharge line piping. The pressure interlock setpoint of LPCI injection valves F042A, B, and C is 130 psid. The differential pressure transmitters N658A, B, and C connected to LPCI injection valves F042A, B, and C are not an overpressurization protection device. In situations where the injection valves F042A, B, and C are open, i.e., surveillance/operability tests, check valves F041A, B, and C, and relief valves F025A, B, and C prevent overpressurization of the low-pressure piping. The routine verification of low leak rates of check valves F041A, B, and C ensures that valves F041A, B, and C and F025A, B, and C can provide adequate overpressurization protection when LPCI injection valves F042A, B, and C are open. This feature improves the reliability of the LPCI system by reducing the differential pressure which permits valve actuation. This decreases the size of the actuators required for the F042A, B, and C injection valves. The smaller valve actuators have correspondingly smaller actuator/valve-stem/disc loads and hence increased reliability. Valve installations are modified to prevent bonnet pressure locking.

The head spray line has a swing check valve, F019, to protect it from overpressurization. Additionally, a gate valve, F023, is located in series and has pressure interlocks to prevent opening

at high inboard pressure. No single-active failure nor Operator error with the head spray line will cause overpressurization of the lower-pressure piping.

5.4.7.2.2 Equipment and Component Description

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 reactor building ventilating system. The pumps are sized on the basis of the LPCI mode (Mode A) and the minimum flow bypass mode (Mode G) of process data on Figure 5.4-14. Design pressure for the pump suction structure is 220 psig with a temperature range from 40° to 360°F. Design pressure for the pump discharge structure is 500 psig. The bases for the design temperature and pressure 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 Section 6.3.2.2 and the pump characteristic curve provided on Figure 5.4-15. Available NPSH is calculated in accordance with RG 1.1 as shown in Section 6.3.2.2.

Pressure Pumps

RHR pressure pump P2 is provided to maintain RHR Loop B and C discharge header full of water and pressurized to avoid water hammer upon system initiation. This function is provided by the LPCS pressure pump P2 for RHR loop A. RHR pump P2 and CSL pump P2 are similar.

The RHR pump P2 is a horizontal-mounted, one-stage centrifugal pump. The pump is rated for 50 gpm with 175 ft TDH.

Heat Exchangers

The RHR heat exchangers are sized on the basis of the duty for shutdown cooling mode (Mode E). All other uses of these exchangers require less cooling surface. Flow rates are 7,450 gpm (rated) on the shellside and 7,400 gpm (rated) on the tubeside (service water side). Rated inlet temperature is 125°F shellside and 67°F tubeside. The overall heat transfer coefficient is 273 Btu/hr sq ft-°F. The exchangers contain 4,240 sq ft of effective surface. Design temperature of the shellside is 40° to 480°F. Design temperature on the tubeside is 32° to 480°F. Design pressure is 500 psig on both sides; fouling factors are 0.0005 shellside and 0.001 tubeside. The construction materials are carbon steel for the pressure vessel with Type 304L stainless steel tubes and stainless steel clad tube sheet.

Valves

All directional valves in the system are conventional gate, globe, butterfly, stop check, and check valves designed for nuclear service. The injection valves, reactor coolant isolation valves, and pump minimum flow valves are high-speed valves, as required for LPCI or vessel isolation. Valve pressure ratings are as necessary to provide the control or isolation function; that is, all vessel isolation valves are rated Safety Class 1 nuclear valves rated at the same pressure as the primary system. All other containment isolation valves are Safety Class 2 nuclear valves rated at the same pressure as the piping in which they are installed.

The RHS service water crosstie valves 2RHS*MOV115 and 2RHS*MOV116 have been modified to meet the requirements of Generic Letter 95-07 for pressure locking.

Valves 2RHS*MOV4A, B and C have been modified to meet the requirements of Generic Letter 95-07 for pressure locking.

ECCS Portions of the RHR System

The ECCS portions of the RHR system include those sections described through Mode A-1 and A-2 of Figure 5.4-14. The flow path includes suppression pool suction strainers, suction piping, RHR pumps, discharge piping, injection valves, and drywell piping into the vessel nozzles and core region of the reactor vessel. Pool cooling components include pool suction strainers, suction piping, pumps, heat exchangers and pool return lines. Containment spray components are the same as suppression pool cooling components except that the spray headers replace the suppression pool return lines.

5.4.7.2.3 Controls and Instrumentation

Controls and instrumentation for the RHR system are described in Sections 7.3.1.1.1 and 7.4.1.3.

5.4.7.2.4 Applicable Standards, Codes, and Classifications

Piping, Pumps, and Valves

Process side	Safety Class 1 and 2
Service water side	Safety Class 3

Heat Exchangers

Process side	Safety Class 2 TEMA Class C
Service water side	Safety Class 3 TEMA Class C

Electrical Portions

BOP	NSSS
IEEE-279-1971	IEEE-279-1971
IEEE-308-1974	IEEE-308-1971
IEEE-384-1974	IEEE-323-1971
IEEE-379-1977	IEEE-384-1974
IEEE-323-1974	

5.4.7.2.5 Reliability Considerations

The RHR system design includes the redundancy requirements of Section 5.4.7.1.5. Two completely redundant loops, each powered from a separate emergency bus, remove residual heat. All mechanical and electrical components, except for the common cooling shutdown suction line, are separate. Either loop is capable of shutting down the reactor within a reasonable length of time. The system design features, which assure that the systems connected to the RHR system do not degrade the reliability of the RHR system, are discussed in Section 6.3.2.5.

5.4.7.2.6 Manual Action

Residual Heat Removal (Shutdown Cooling Mode)

In shutdown cooling operation, when vessel pressure is 128 psig or less, the pool suction valve may be closed for the initial shutdown loop or loops. Flushing valves connecting the RHR condensate and makeup system are manually opened, and the stagnant water flushed to the radwaste. At the end of this flush, service water flow is established through the heat exchanger. This is followed by the gradual warming of the cooling loop by permitting vessel water to warm the shutdown cooling loop (RHR pump not running). The vessel water used in prewarming is directed to radwaste, and a temperature element is used to monitor the effluent temperature. When the prewarming is completed, the radwaste effluent valves are closed, and the RHR heat exchanger bypass is opened. The RHR pump is started and the total flow is regulated through 2RHS*MOV40A/B (E12-F053A/B), shutdown cooling throttle valve. The vessel cooldown rate is then normally controlled by regulating total flow using 2RHS*MOV40A/B (E12-F053A/B) for total flow control, regulating heat exchanger bypass flow using 2RHS*MOV8A/B (E12-F048A/B), and/or regulating flow through the heat exchanger using 2RHS*MOV12A/B (E12-F003A/B).

A loop of RHR may be placed in shutdown cooling on the same loop as a reactor recirculation pump operating in slow speed. Since a reactor recirculation pump initially starts in fast speed then automatically downshifts to slow speed, a shutdown cooling loop must be secured prior to starting a reactor recirculation pump on the same loop.

The manual actions required for the most limiting failure are discussed in Section 15.2.9.

Steam Condensing

There is no steam-condensing mode of RHR. This mode of operation has been abandoned by installation of pipe caps and removal of power from components associated with this mode of operation.

5.4.7.3 Performance Evaluation

Thermal performance of the RHR heat exchangers is based on the residual heat generated at 20 hr after rod insertion, a 125°F heat exchanger inlet temperature, a service water temperature of 67°F, and the flow of two loops in operation. These are nominal design conditions; if the 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 of vessel cooldown temperatures versus time is shown here due to the infinite variety of such curves that may be due to: 1) clean steam systems that may allow the main condenser to be used as the heat sink when nuclear steam pressure is insufficient to maintain steam air ejector performance, 2) the condition of fouling of the exchangers, 3) Operator use of one or two cooling loops, 4) coolant water temperature, and 5) system flushing time. Since the exchangers are designed for the fouled condition with relatively high service water temperature, the units have excess capability to cool when first cut in at high vessel temperature. Total flow and mix temperature must be controlled to avoid exceeding 100°F/hr cooldown rate (see Section 5.4.7.1.1 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 15.2.9. The capability of the heat exchanger for any time period is balanced against reactor 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 (Chapter 14) are used to generate data to verify the operational capabilities of each piece of equipment in the system: instruments, setpoints, logic elements, pumps, heat exchangers, valves, and limit switches. In addition, these programs verify the capabilities of the system to provide the flows, pressures, condensing rates, cooldown rates, and reaction times required to perform all system functions as specified for the system or component in the system data sheets and process data. Logic

elements are tested electrically; valves, pumps, controllers, and relief valves are tested mechanically. Finally, the system is tested for total system performance against the design requirements, as specified above, using both 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 suppression pool cooling.

5.4.8 Reactor Water Cleanup System

The RWCU system is classified as a power generation system (not an engineered safety feature [ESF]), a small part of which is part of the RCPB up to and including the outside isolation valve. The other portions of the system are not part of the RCPB and can be isolated from the reactor. The RWCU system may be operated at any time during planned reactor operations or it may be shut down if water quality is within the Technical Specification limits. The seismic and safety class for this system are provided in Table 3.2-1.

The reactor chemistry limits outlined in RG 1.56 Revision 1, Table 1, will be met. These chemistry limits, as well as corrective action, will be established in the Technical Specifications. The following methods of chemical analyses will be used:

Parameter	Analysis Method*
Chloride	ASTM D512-81C or Ion Chromatographic*
pH	ASTM D1293-84B
Conductivity	ASTM D1125-82B

5.4.8.1 Design Bases

5.4.8.1.1 Safety Design Basis

The RCPB portion and the high-pressure portion beyond the outside containment isolation valves are assessed to meet the requirements of RG 1.26 and 1.29 in order to:

1. Prevent excessive loss of reactor coolant.

* The method used will provide adequate sensitivity to ensure that the limits described in RG 1.56 Rev. 1, Table 1, will be met. Alternative methods are acceptable provided adequate analytical sensitivity is ensured.

2. Prevent the release of radioactive material from the reactor.

3. Isolate the major portion of the RWCU system from the RCPB.

5.4.8.1.2 Power Generation Design Basis

The RWCU system:

1. Removes solid and dissolved impurities from reactor coolant. The RWCU system is assessed to be in compliance with RG 1.56.

2. Discharges excess reactor water during startup, shutdown, and hot standby conditions to the main condenser or liquid radwaste system.

3. Minimizes temperature gradients in the reactor recirculation piping and RPV during periods when the reactor recirculation system pumps are unavailable.

The RWCU system is designed to:

1. Minimize system heat loss.

2. Enable the major portion of the system to be serviced during reactor operation.

3. Prevent 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 takes its suction from the inlet of each reactor recirculation pump and from the RPV bottom head. The process fluid is circulated by the cleanup pumps through the regenerative and nonregenerative heat exchangers 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, the main condenser, or liquid radwaste system (Figures 5.4-16a through 5.4-16f and 5.4-17).

The major components of the RWCU system are located outside the drywell. These components include pumps, regenerative and nonregenerative heat exchangers, filter demineralizers, and associated precoat equipment. Flow rate capacities for the major components are presented in Table 5.4-3.

Because the operating temperature of the filter demineralizer units is constrained by the resin operating temperature limit, the reactor coolant must be cooled before being processed in the filter demineralizer units. The regenerative heat exchanger transfers heat from the influent (tubeside) to the effluent (shellside). The effluent returns to the reactor. The

nonregenerative heat exchanger cools the process influent further by transferring heat to the RBCLCW system.

The filter demineralizer units (Figures 5.4-16d through 5.4-16f and 5.4-19) are pressure precoat-type filters, using a nonregenerable, mixed ion-exchange resin precoat material with or without a fiber filter aid material. Spent resins are not regenerable and are sluiced from the filter demineralizer unit to a phase separator tank in the radwaste system for processing and disposal (the resin backwash transfer system is described in Section 11.2.2.4). Resins are discarded based on filter demineralizer performance, as indicated by monitoring effluent conductivity, differential pressure across the unit, and sample analysis. Initial total capacity is not measured on resin that is finely ground and mixed since separation into anion and cation components is not practical. To prevent resins from entering the reactor coolant system in the event of failure of a filter demineralizer resin support, a strainer is installed in the effluent line of each filter demineralizer. 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 automatically isolates.

The backwash and precoat cycle for each filter demineralizer is semiautomatic; however, permissives and interlocks are installed in the logic program to prevent human operational errors, such as inadvertent opening of valves that would initiate a backwash or contaminate reactor water with precoat material or 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.

The vent line from each filter demineralizer is routed to the phase separator tanks which are vented directly to the reactor building heating, ventilating, and air conditioning (HVAC) system (Section 9.4.2).

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. This ensures that the precoat and resin material are held in place on the septum screens. 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 main control room. The control room alarm setpoints of the conductivity meters at the inlet and outlet lines are 1.0 umho/cm and $\leq$0.15 umho/cm, respectively. 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 RWCU system contains two motor-operated isolation valves, which automatically close in response to signals from the RPV (low water level), the LDS, and actuation of the SLCS. Activation of the SLCS closes the outside isolation valve from Division I logic and the inside isolation valve from Division II logic. Nonregenerative heat exchanger high outlet temperature closes the outside isolation valve only. Section 7.6 describes the LDS requirements which are summarized in Table 5.2-8. This isolation prevents the loss of reactor coolant and release of radioactive material from the reactor. The isolation valves close automatically to prevent removal of liquid boron reactivity control material from the reactor vessel in the event of SLCS activation. In addition, the outside isolation valve closes automatically to prevent damage of the filter demineralizer resins if the outlet temperature of the nonregenerative heat exchanger is high. 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 globe valve on the return line to the reactor provides long-term leakage control. Instantaneous reverse flow isolation is provided by check valves in the RWCU system piping.

The RWCU return line is split into two branches connecting to the two feedwater loops outside the primary containment. Each branch is equipped with a MOV. Under normal plant operation, both MOVs are open and RWCU water is returned to the RPV via two feedwater loops. During plant startup when the reactor power is below 16.67 percent, one of the MOVs may be closed directing all RWCU flow into a single feedwater loop. This will be done to minimize thermal stratification in the feedwater line during startup and shutdown and allow sharing the thermal cycles between the two feedwater loops. Both valves may be closed as necessary to support surveillance testing during plant shutdown.

Operation of the RWCU system is controlled from the main control room. Resin-changing operations, which include backwashing and precoating, are controlled from a local control panel.

A FCD is provided on Figure 7.3-7. Controls for valves 2WCS*MOV404A and B are shown on Figure 10.4-11.

5.4.8.3 System Evaluation

The RWCU system, in conjunction with the CND system, maintains reactor water quality during all reactor operating modes (normal, hot standby, startup, shutdown, and refueling). During refueling mode, the spent fuel pool cooling and cleanup system (SFC) also contributes to this function. This type of pressure precoat cleanup system has been used in all operating BWR plants since 1971. Operating plant experience has shown that the RWCU system, as designed, maintains 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 diverting 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 (flow rates, 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 Figures 5.4-16a through 5.4-16f and 5.4-17.

5.4.9 Main Steam Line and Feedwater Piping

This section describes the design of main steam piping (Sections 3.6A.1, 3.6A.2 and 3.6B.2, and 10.3) and feedwater piping up to and including the piping through the jet impingement wall (Sections 3.6A.1, 3.6A.2, and 3.6B.2).

5.4.9.1 Safety Design Bases

In order to satisfy the safety design bases, the main steam and feedwater lines have been designed:

1. To withstand operational stresses, i.e., internal pressures, thermal expansion, SRV loads, fluid and thermal transients, and SSE loads, without a failure that could lead to the release of radioactivity in excess of the guideline values in published regulations.

2. With suitable access to permit ISI and IST.

3. To withstand without loss of safety function the fluid jets missiles, reaction forces, pressures, and temperatures postulated to result from pipe breaks (Sections 3.6A.1, 3.6A.2, and 3.6B.2).

5.4.9.2 Power Generation Design Bases

In support of reliable power generation, the following design bases have been employed for the main steam and feedwater piping:

1. The MSLs are designed to conduct steam from the reactor vessel throughout the full range of reactor power operation.

2. The feedwater lines are designed to conduct water to the reactor vessel throughout the full range of reactor power operation.

5.4.9.3 Description

The main steam piping is described in Sections 3.6A.1, 3.6A.2, 3.6B.2, and 10.3. The main steam and feedwater piping is shown

on Figures 10.1-3 and 10.1-6. The feedwater piping consists of two 24-in outside diameter lines, each of which penetrates the drywell and branch into three 12-in lines that connect to the reactor vessel. Each line includes three containment isolation valves consisting of one check valve inside the drywell, and one motor-operated gate valve, and one spring-loaded piston-actuated check valve outside the drywell. The design pressure and temperature of the feedwater piping between the reactor and long-term isolation valve are 1,300 psig and 575°F. The design pressure and temperature of the feedwater piping between the reactor feed pumps and 2FWS-MOV47A/B/C and 2FWS-V103A/B are 2,250 psig and 395°F, between 2FWS-MOV47A/B/C and 2FWS-V103A/B and the sixth-point heaters are 2,200 psig and 395°F, and between the sixth-point heaters and the long-term isolation valves are 2,200 psig and 450°F. The Category I design requirements are invoked on the feedwater piping from the reactor through the outboard isolation valve and jet impingement wall. The materials used in the piping are in accordance with the applicable design code and supplementary requirements (Section 3.2). The general requirements of the feedwater system are described in Sections 7.7.1.3 and 10.4.7.5.

5.4.9.4 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 MSLs. All main steam and feedwater lines of the RCPB are designed in accordance with the requirements defined in Sections 3.2, 3.6A.1, 3.6A.2, and 3.6B.2. Design of the piping in accordance with these requirements ensures that the safety design bases are met.

5.4.9.5 Inspection and Testing

Inspection and testing are carried out in accordance with Sections 3.9A.1, 3.9B.1, 5.2.4 and Chapter 14. ISI is considered in the design of the main steam and feedwater piping. This consideration assures adequate working space and access for the inspection of selected components.

5.4.10 Pressurizer

(Not applicable to BWR)

5.4.11 Pressurizer Relief Discharge System

(Not applicable to BWR)

5.4.12 Valves

5.4.12.1 Safety Design Bases

Line valves are located in the fluid systems to perform the mechanical function of controlling or prohibiting flow in one or more directions. They are components of the system pressure boundary and are designed to operate efficiently to maintain the integrity of that boundary. The valves operate under the internal pressure/temperature loading experienced during the various system transient operating conditions and various plant conditions such as seismic events or main steam system relief valve blowdown. The design, loading, and acceptability criteria are as required for Safety Class 1, 2, and 3 valves (Sections 3.9A.3 and 3.9B.3) and ANSI B31.1 for Safety Class 4 valves. Compliances with ASME Codes are discussed in Section 5.2.1. The ADS valves comply with the requirements of NUREG-0737 for reactor coolant system venting, as described in Section 1.10.

5.4.12.2 Description

Line valves are standard manufactured types such as gate, globe, check, ball, and butterfly. They are designed and constructed in accordance with the requirements of Safety Class 1, 2, and 3 valves and ANSI B31.1 for Safety Class 4 valves. All materials, exclusive of seals, packing, and wearing components, are designed to remain functional for a 40-yr life under the environmental and design conditions applicable to the particular system and physical location in the plant when appropriate maintenance is periodically performed.

Valves requiring remote power operation utilize motor, electrohydraulic, hydraulic/mechanical, pneumatic or solenoid-actuated operators, or a combination thereof. The actuators have been sized by the manufacturer to operate successfully under the specified maximum operating conditions at the specified time rate stated in the design specification. Control and operation of these power-operated valves are discussed in the sections covering the systems in which particular valves are located.

5.4.12.3 Safety Evaluation

Line valves have been hydrostatically tested by the manufacturer for their performance as a pressure boundary. Pressure-retaining parts are subject to the testing and examination requirements of Safety Class 1, 2, and 3 valves and ANSI B31.1 for Safety Class 4 valves.

All electrical components of power-actuated valve operators utilized on Safety Class 1, 2, and 3 valves have been designed and qualified as described in Sections 3.10A, 3.10B, and 3.11.

To prevent motor overheating due to frequent cycling of ECCS/RCIC MOVs, operating procedures have been developed to caution plant Operators to be aware of the allowable duty cycles on these valves. The allowable duty cycle of ECCS QA Category I MOVs is five cycles open and closed per hour. The duty rating for RCIC

MOVs is also five cycles per hour except for a two-cycle-per-hour rating for the turbine exhaust valve (E51-F068). These allowable cycles per hour envelope the required duty for ECCS and RCIC valves during normal, transient, and accident modes of operation.

The frequent cycling of the ECCS/RCIC MOVs is ended after the first hour of LOCA or transient event.

To verify their operability, Class 1E MOVs are qualified in accordance with IEEE-382-1980, IEEE-323-1974, and IEEE-344-1975 as part of the Unit 2 Equipment Qualification Program.

5.4.12.4 Inspection and Testing

Inspection and testing of all valves and valve power actuators have been performed in accordance with Safety Class 1, 2, and 3 valves, ANSI B31.1 for Safety Class 4 valves, and the additional requirements of the design specification as applicable.

Hydrostatic shell tests have been performed in accordance with Article NB-6000 for Safety Class 1 valves, Article NC-6000 for Safety Class 2 valves, and Article ND-6000 for Safety Class 3 valves. Seat-tightness tests for Safety Class 1, 2, and 3 valves, and shell and seat-tightness tests for Safety Class 4 valves were performed in accordance with Manufacturer's Standardization Society Standard MSS-SP61 as a minimum. Certain Class 4 air-actuated valves have had seat-tightness testing performed in accordance with ANSI B16.104 in lieu of MSS-SP61.

Nondestructive testing of Safety Class 1, 2, and 3 valves was performed in accordance with the applicable subsection of ASME Section III and the additional requirements of the design specification when required. Personnel performing nondestructive testing on Safety Class 1, 2, and 3 valves were qualified in accordance with Society of Nondestructive Testing Standard SNT-TC-1A. Nondestructive testing of Safety Class 4 valves was performed in accordance with ANSI B31.1 and the additional requirements of the design specification when applicable.

All power-actuated valve operators have 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 at the manufacturer's shop. Valve actuator electric motors have been furnished in accordance with applicable sections of NEMA Standard MG-1. Assembled power-actuated valves have been tested to demonstrate adequate stem thrust (or torque) capability to operate the valve within the specified time at specified differential pressure. Tests verified that no mechanical damage to valve components occurred during full stroking of the valve.

<u>Operational Analysis</u>

Preoperational and operational testing performed on the installed valves consists of total circuit checkout and performance tests.

Valves that function as containment isolation valves will be exercised in accordance with Technical Specifications to assure their operability at the time of an emergency or faulted condition. Other valves, serving as system blocks or throttling valves, will be exercised when appropriate.

5.4.13 Safety and Relief Valves

5.4.13.1 Safety Design Bases

Overpressure protection has been provided at isolatable portions of systems in accordance with the rules set forth in Safety Class 1, 2, and 3 components.

5.4.13.2 Description

Pressure relief valves have been designed and constructed in accordance with the same code class as that of the line valves in the system.

Table 3.2-2 lists the applicable code classes for valves. The design criteria, design loading, and design procedure are described in Section 3.9B.3. Specific data (e.g., capacity, setpoint) are discussed in Section 5.2.2.

5.4.13.3 Safety Evaluation

The use of pressure-relieving devices assures that overpressure does 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.

5.4.13.4 Inspection and Testing

No provisions are made for in-line testing of pressure relief valves. Certified set pressures and relieving capacities are stamped on the body of the valves by the manufacturer, and further examinations would necessitate removal of the component.

5.4.14 Component Supports

Support elements are provided for those components included in the RCPB and the connected systems.

5.4.14.1 Safety Design Bases

The structural integrity of component supports is such that the supports are capable of safely sustaining the maximum combination of design loads.

Design loading combinations, design procedures, and acceptability criteria are as described in Sections 3.9A.3 and 3.9B.3. Flexibility calculations and seismic analysis for Safety Class 1, 2, and 3 components conform to the appropriate requirements of ASME Section III.

Support types and materials used for fabricated support elements conform to Articles NF-2000 and NF-3000 of ASME Section III. Pipe support spacing guidelines of Table NF-3133.1-1 of ASME Section III were followed.

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 and stress analysis. Component support elements are manufacturer's standard items. Direct weldments 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 have been made to protect spring-type supports from the initial deadweight loading due to hydrostatic testing of steam systems to prevent damage to this type of support.

5.4.14.4 Inspection and Testing

After completion of the installation of a support system, all hanger elements are visually examined to assure that they are in correct adjustment to their cold setting positions. Upon hot startup operations, thermal growth will be observed to confirm that spring-type hangers will function properly between their hot and cold setting positions. Final adjustment capability is provided on all spring hanger-type supports.

5.4.15 References

1. Ianni, P. W. Effectiveness of Core Standby Cooling Systems for General Electric Boiling Water Reactors. APED-5458, March 1968.

2. Analysis of Recirculation Pump Under Accident Conditions, Revision 2, General Electric Company, March 30, 1979.

3. Design and Performance of General Electric Boiling Water Reactor Main Steam Line Isolation Valves, General Electric Company Atomic Power Equipment Department, APED-5750, March 1969.

4. Licensing Topical Report, Power Uprate Licensing Evaluation for Nine Mile Point Nuclear Power Station, Unit 2, NEDC-31994P, Revision 1, May 1993.

TABLE 5.4-1
(Sheet 1 of 3)
REACTOR RECIRCULATION SYSTEM DESIGN CHARACTERISTICS

External Loops (2)

Single Loop Piping Description	Quantity	Approx. Length (ft)	Nominal Size (in)
Pump suction line			
Straight pipe	-	30	24
Elbows	3	-	24
Gate valves	1	-	24
Discharge line			
Straight pipe	-	20	24
Elbows	2	-	24
Flow control valves	1	-	24
Gate valves	1	-	24
Discharge manifold			
Pipe	-	40	16
Reducer cross	1	-	24x16
Contour nozzle	4	-	16x12
Caps	2	-	16
Concentric reducer	1	-	24x12
External risers			
Straight pipe	5	5	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			1,650/575
Piping after discharge blocking valve up to vessel			1,550/575
Pump auxiliary cooling water piping			145/150
Vessel bottom drain			1,275/575

TABLE 5.4-1 (Cont'd.)

(Sheet 2 of 3)

Operation at 3988 MWt rated power and 100% rated core flow	
Recirculation pump	
Flow, gpm	43,844
Flow, lb/hr	16.56E6
Total developed head, ft	843.2
Suction pressure (static), psia	1,036.3
Required NPSH, ft	103.98
Water temperature, °F	533.7
Pump brake, hp (60 Hz)	8,104
Flow velocity at pump suction, fps	38.0
Pump Motor	
Voltage rating	13,200
Speed, rpm	1,780
Motor rating, hp	8,900
Phase	3
Rotational inertia, lbm-ft^2	21,500
Jet Pumps	
Number	20
Total flow, lb/hr/jet pump	$5.42x10^6$
Throat ID, in	6.34
Diffuser ID, in	19.0
Nozzle ID (5 each), in	1.3
Diffuser exit velocity, fps	16.2
Jet pump head, ft	88.2
Flow Control Valve	
Type	Ball
Material	Austenitic stainless steel
Type actuation	Hydraulic
Failure mode (on loss of power or control signal)	As is
CV	8,310
Valve size diameter, in	24

TABLE 5.4-1 (Cont'd.)
(Sheet 3 of 3)

Recirculation Block Valve, Discharge

Type	Gate
Actuator	Motor operator
Material	Stainless steel
Valve size diameter, in	24

Recirculation Block Valve, Suction

Type	Gate
Actuator	Motor operator
Material	Stainless steel
Valve size diameter, in	24

LFMG Set Name Plate

Motor hp	400
Voltage	4,000
Generator frequency, Hz	15

TABLE 5.4-2
(Sheet 1 of 1)
RHR RELIEF AND SAFETY VALVE DATA

Valve	Function	Capacity Required/Actual (gpm)	Set Pressure (psig) Maximum
F025 (2RHS*RV20 A,B,C)	Thermal relief	NA/10	470
F088A,B (2RHS*RV61 A,B)	Thermal relief	NA/1	200
F088C (2RHS*RV61C)	Thermal relief	NA/1	105
F005 (2RHS*RV110)	Thermal relief	NA/1	200
F030 (2RHS*RV139)	Thermal relief	NA/1	220
(F231) 2RHS*RV152	Thermal relief	NA/10	1240
2RHS*RV56A,B	Thermal relief	NA/20	500
2RHS*RV42A,B	Thermal relief	NA/20	500
(F236) 2RHS*RV117	Thermal relief	NA/3.6	180

TABLE 5.4-3
(Sheet 1 of 1)
REACTOR WATER CLEANUP SYSTEM EQUIPMENT DESIGN DATA

System Flow Rate, lb/hr	340,000	
Main Cleanup Recirculation Pumps		
Number provided	2	
Capacity, % each	50	
Design temperature, °F	575	
Design pressure, psig	1,410	
Discharge head at shutoff, ft	630	
Minimum required NPSH, ft	13*	
Heat Exchangers		
	Regenerative	Non-Regenerative
Number required	1	1
Shell design pressure, psig	1,410	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 of systems	2	
Number of units/system	2	
Capacity %, each	25	
Flow rate/unit, lb/hr	85,000**	
Design temperature, °F	150	
Design pressure, psig	1,410	

* 23 ft required for 1 pump and 3 filter/demin operation.
** For 1 pump and 3 filter/demin operation, flow rate/unit is 100,000 lbm/hr.

APPENDIX 5A

COMPLIANCE WITH 10CFR50, APPENDIX G AND APPENDIX H

APPENDIX 5A

COMPLIANCE WITH 10CFR50, APPENDIX G AND APPENDIX H

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ATTACHMENTS

APPENDIX 5A

COMPLIANCE WITH 10CFR50, APPENDIX G AND APPENDIX H

5A.1 REACTOR VESSEL BELTLINE PLATE AND WELD INFORMATION

The Unit 2 reactor vessel was procured to meet the requirements of ASME Section III, 1971 Edition with Winter 1972 Addenda. Thus, it is in compliance with the toughness testing requirements of 10CFR50 Appendix G which are specified in the ASME Code (e.g., RT_{NDT}).

Upper shelf Charpy toughness testing was not required by ASME Section III for vessel procurement, but is a requirement of 10CFR50 Appendix G. An upper shelf level of at least 75 ft-lb is required by Appendix G. All Unit 2 vessel beltline weld materials had at least 75 ft-lb upper shelf Charpy toughness, as shown in Table 5A-1. Unit 2 beltline materials were procured to a specified minimum value of 70 ft-lb (transverse), as permitted by NRC BTP MTEB 5-2 when the EOL fluence is less than 1 x 10^{19} n/sq cm. The predicted EOL fluence at the Unit 2 vessel wall 1/4-thickness (1/4 T) location from the ID is 1.1 x 10^{18} n/sq cm (Reference 3) or 1.12 x 10^{18} n/sq cm (EPU/MELLLA+).

All Unit 2 vessel beltline plates have upper shelf transverse Charpy specimen results of at least 75 ft-lb (Table 5A-2) except for Heats C3121-2 and C3147-1. Heat C3121-2 is in compliance with the proposed revision of 10CFR50 Appendix G since the average of three upper shelf tests is 75 ft-lb. Heat C3147-1 has an average result just slightly less than 75 ft-lb (74.3), and is also considered acceptable because of the low copper content of this material and the low Unit 2 neutron fluence. Using RG 1.99, a decrease in upper shelf of approximately 12 percent for the fluence in Reference 3 or 12.5 percent for EPU/MELLLA+ is predicted for C3147-1. This would result in an EOL upper shelf of at least 65 ft-lb for the fluence in Reference 3 or 61 ft-lb for EPU/MELLLA+, which is well in excess of the goal of 50 ft-lb as stated in the proposed revision of 10CFR50 Appendix G.

Tables 5.3-2a and 5.3-2b present percent Cu and percent Ni contents, RT_{NDT} values, and the estimated effect of neutron fluence on beltline RT_{NDT} values for beltline materials.

Figure 5A-1 defines beltline weld seam and plate locations.

5A.2 REACTOR VESSEL NONBELTLINE INFORMATION

As stated for the beltline material, the Unit 2 vessel was procured to the requirements of ASME Section III, 1971 Edition with Winter 1972 Addenda, which are consistent with the toughness testing requirements of 10CFR50 Appendix G.

A review of QA records (documented deviations from the reactor vessel vendor) does not reveal any deviations of fracture toughness specifications from purchase requirement limits:

1. RT_{NDT} no greater than +10°F for the shell course, head, and closure flange.

2. RT_{NDT} no greater than -20°F for nozzle forgings.

3. RT_{NDT} no greater than -20°F for low alloy weld metal used to join base or weld materials requiring impact testing.

4. Vessel main closure studs meet the Charpy requirement of 45 ft-lb and 25 mils at +10°F.

The use of these toughness limits and values to establish vessel operating limits is described in Section 5.3.2.

5A.3 OTHER FERRITIC REACTOR COOLANT PRESSURE BOUNDARY MATERIALS (NSSS)

The subject materials were impact tested, and are considered to be in compliance with 10CFR50 Appendix G. Specific components, applicable code requirements, and impact test temperatures are as follows:

1. SRV (8 x 10 in) - ASME Section III, 1974 and Summer 1976 Addenda, +60°F maximum.

2. HPCS isolation valve - ASME Section III, 1971 and Winter 1973 Addenda, +40°F maximum.

3. MSIV - ASME Section III, 1977 and Summer 1977 Addenda, +60°F maximum.

5A.4 ORIGINAL PLANT-SPECIFIC REACTOR PRESSURE VESSEL SURVEILLANCE SPECIMENS

Surveillance specimen materials are identified, with properties, in Tables 5A-1, 5A-2, 5.3-2a, and 5.3-2b. It can be seen in Tables 5.3-2a and 5.3-2b that all beltline materials are resistant to radiation degradation of toughness. One of the limiting plates (in terms of EOL RT_{NDT}) was used to fabricate surveillance specimens. The weld materials (Table 5.3-2b) used for the surveillance specimen weld had a predicted EOL RT_{NDT} of

11°F to 13°F lower than the limiting weld material in this respect, but had the highest predicted shift in RT_{NDT} (which is the current basis in ASTM E185-79 (Table 5A-2) for surveillance program design). This is not considered significant because of the insensitivity of these materials to neutron radiation damage. HAZ specimens were taken from the HAZ of the weldment fabricated from the materials indicated in Table 5.3-2a.

The weld procedure used to prepare the surveillance specimen weldment is shown as Attachment 5A-1. Although stick electrodes (shielded metal arc weld) were used to seal backup bars, as shown on Figure 5A-2, these materials were removed by backgouging. The weld joint design (Figure 5A-2) shows that the weld specimens are not in the vicinity of the weld root. Therefore, the weld materials in the surveillance specimens are determined to be those indicated in Table 5.3-2b.

Specimen orientations, locations in test plate, and quantities are in compliance with ASTM E185-73. The HAZ specimen axis is parallel to the plate principal rolling direction.

Surveillance capsule contents and locations are shown in Table 5A-4. Further details of the surveillance program are included in Section 5.3.1.6.

TABLE 5A-1
(Sheet 1 of 2)
BELTLINE WELD TOUGHNESS DATA, SUBMERGED ARC WELDS

Post Weld 1,150°F for 50 Hr Typical

Weld Seam	Type	Heat No.	Lot No. or Flux No.	Drop-Weight NDT (°F)	Charpy Toughness			
					Charpy Temp (°F)	Charpy Energy (ft-lb)	Lateral Expansion (mils)	% Shear
No. 2 shell longitudinal seams BD,BE,BF	1NMM (single wire, trade name Raco)	5P5657*	0931 (Linde 124)	-60	-80	39,39,29	27,37,32	5,5,5
					-60	19,20,32	18,22,28	10,10,10
					0	51,55,68	50,50,63	30,30,55
					+10	69,69,66	61,65,59	50,50,40
						62,57	60,63	60,40
					+40	77,76	73,72	70,80
					+212	88,91,85	86,75,83	100,100,100
	1NMM (tandem wire, trade name Raco)	5P5657*	0931 (Linde 124)	-80	-80	14,23,20	15,22,19	5,5,5
					-20	42,45,47	41,43,44	20,15,20
					-10	48,46,39	44,42,40	15,20,20
					0	51,57,55	50,54,40	20,30,20
					+10	58,61,65	58,54,59	55,40,55
						55,63	50,60	45,75
					+40	69,76	64,74	75,80
					+212	88,88,91	75,84,74	100,100,100
No. 1 shell longitudinal seams BA,BB,BC	1NMM (single wire, trade name Raco)	5P6214B	0331 (Linde 124)	-50	-70	22,13,11	17,10,9	2,2,2
					-50	42,13,34	34,11,26	15,5,10
					+10	56,50,54	45,41,46	25,20,30
					+40	76,66	66,52	75,45
					+100	87,89	70,64	95,90
					+120	96,90,88	68,61,71	100,100,100
	1NMM (tandem wire, trade name Raco)	5P6214B	0331 (Linde 124)	-40	-80	17,21	14,17	2,2
					-40	25,37,29	20,29,28	5,5,5
					+10	50,61,64	46,50,52	50,40,35
					+20	75,69,78	53,47,55	55,60,55
					+40	82,80	65,62	75,75
					+120	100,96,97	60,81,57	100,100,100

TABLE 5A-1
(Sheet 2 of 2)

Weld Seam	Type	Heat No.	Lot No. or Flux No.	Drop-Weight NDT (°F)	Charpy Toughness			
					Charpy Temp (°F)	Charpy Energy (ft-lb)	Lateral Expansion (mils)	% Shear
No. 1 to 2 shell girth seam AB	1NMM (single wire, trade name Raco)	4P7465	0751 (Linde 124)	-70	-80 -70 0 +10 +40 +212	27,14 48,43,26 63,57,68 56,58,90 87,55 67,97 118,102,112	21,12 42,36,22 54,45,63 62,62,86 83,42 71,90 88,71,72	5,0 15,15,15 30,25,35 30,25,45 40,30 45,50 100,100,100
	1NMM (tandem wire, trade name Raco)	4P7465	0751 (Linde 124)	-60	-80 -60 0 +10 +40 +212	15,21 62,56,43 79,83,74 71,72,74 76,85 94,106 123,110,111	11,20 47,45,36 66,60,54 69,73,72 77,76 74,88 70,76,77	0,0 30,30,25 30,30,25 30,35,35 40,40 60,75 100,100,100
	1NMM (single wire, trade name Raco)	4P7216	0751 (Linde 124)	-60	-80 -60 0 +10 +40 +212	5,7 19,38,24 41,59,58 64,60,72 66,60 61,63 90,89,94	4,6 22,35,22 32,46,47 48,41,60 53,41 41,44 83,77,68	5,0 15,20,15 20,20,30 70,30,35 45,40 45,40 100,100,100
	1NMM (tandem wire, trade name Raco)	4P7216	0751 (Linde 124)	-80	-100 -80 -20 +10 +40 +212	22,9 31,23,28 62,73,84 84,92,95 89,87 96,97 102,101,98	17,7 15,11,19 40,51,56 57,61,60 51,64 74,68 87,66,66	0,5 5,5,5 30,40,60 60,95,90 95,90 100,100 100,100,100

* Limiting weld. Weld materials 5P5657/0931, both single and tandem wire processes, are also in the reactor vessel surveillance capsules. See Section 5.3.1.6 for a description of the reactor vessel material surveillance program.

TABLE 5A-2
(Sheet 1 of 3)
BELTLINE PLATE TOUGHNESS DATA

(SA-533 Grade B, Class 1 Plate - Lukens Steel Company)

Plate	Heat No./ Slab No.	Drop-Weight NDT (Top/Bottom) (°F)	Charpy V-Notch Toughness (Top/Bottom)				
			Orien-tation (L or T)	Charpy Test Temp. (ft-lb)	Energy (ft-lb)	Lateral Expansion (mils)	% Shear
No. 2 Shell (lower intermediate) 22-1-1	C3065-1	-30/-30	T	+30	44,49,50/54,54,46	46,36,41/36,41,31	40,40,40/30,30,30
			T	+40	55,60,63/41,55,54	48,46,48/40,45,44	50,50,50/40,40,40
			L	+30	63,61,87/74,76,70	41,61,50/56,61/63	60,60,60/70,70,70
			T	+50	70,50,50	42,52,41	50,50,50
			T	+212	100,97,94	76,74,74	99,99,99
				+150	100,106,100	81,84,85	99,99,99
				+70	54,51,54	46,45,47	50,50,50
				-30	40,34,21	16,24,31	30,30,30
				-70	7,10,10	4,6,6	1,1,1
				-100	7,8,6	4,3,2	1,1,1
22-1-2	C3121-2	-30/-50	T	+30	40,60,56	50,54,36	50,50,50
			T	+40	50,51,50/45,42,44	46,41,44/41,41,40	50,50,50/40,40,40
			L	+30	88,86,56/81,53,56	46,61,56/48,42,62	50,50,50/50,50,50
			T	+10	34,30,40	31,30,27	30,30,30
			T	+60	50,53,50	46,46,45	40,40,40
			L	+10	58,44,32	44,31,36	40,40,40
			L	+20	58,36,42	45,31,35	40,40,40
			T	+212	78,76,71	66,64,61	99,98,99
				+100	73,65,73	61,61,65	90,90,90
				0	45,38,38	34,32,39	40,40,40
				-10	35,36,31	31,31,30	30,30,30
				-30	14,15,24	16,10,11	10,10,10
				-100	10,10,6	8,9,2	1,1,1

TABLE 5A-2
(Sheet 2 of 3)

Plate	Heat No./ Slab No.	Drop-weight NDT (Top/Bottom) (°F)	Charpy V-Notch Toughness (Top/Bottom)				
			Orien- tation (L or T)	Charpy Test Temp. (ft-lb)	Energy (ft-lb)	Lateral Expansion (mils)	% Shear
No. 2 Shell	C3147-1*	-20/-30	T	+40	38,40,41	33,40,36	40,40,40
22-1-3			T	+50	40,62,54/40,54,44	38,41,48/43,41,38	50,50,50/40,40,40
			T	+60	50,51,50/50,50,52	45,44,41/46,44,42	40,40,40/40,40,40
			L	+40	64,64,74	56,50,50	50,50,50
			T	+30	42,42,45	35,41,38	40,40,40
			L	+30	76,80,96	60,61,56	60,60,60
			T	+212	72,70,81	63,66,63	99,99,97
				+100	68,75,86	61,61,68	70,70,70
				+70	66,61,61	53,58,59	80,80,80
				0	39,38,36	36,35,36	40,40,40
				-20	17,23,20	18,20,15	20,20,20
				-150	2,2,2	1,1,1	1,1,1
No. 1	C3147-2*	-20/-30	T	+40	50,51,41	36,38,41	40,40,40
Shell			T	+50	47,56,40	40,42,36	50,50,50
(lower)			T	+60	52,50,50	48,44,44	60,60,60
21-1-1			L	+40	91,96,96	70,65,68	70,70,70
			T	+30	51,56,51	48,51,48	60,60,60
			L	+30	70,80,90	65,74,80	90,90,90
			T	+212	93,86,86	85,79,81	99,99,99
				+100	78,78,80	64,71,66	90,90,90
				+50	54,56,50	48,40,45	50,50,50
				0	34,35,32	31,33,32	30,30,30
				-20	31,23,31	21,21,16	20,20,20
				-150	5,4,6	2,3,2	1,1,1
No. 1	C3066-2	-30/-40	T	+30	45,38,43/38,54,43	36,38,38/40,44,36	40,40,40/40,40,40
Shell			T	+40	58,72,58/55,52,51	58,48,48/45,42,45	50,50,50/40,40,40
21-1-2			L	+30	78,80,90	65,61,66	70,70,70
			T	+20	60,56,36	34,50,59	50,50,50
			L	+20	77,84,80	62,51,60	60,60,60
			T	+212	80,91,86	67,69,71	99,99,99
				+150	88,93,84	76,84,79	99,99,99
				+70	58,65,64	45,56,50	50,50,50
				+30	56,50,40	43,34,46	40,40,40
				-30	33,33,26	21,28,29	30,30,30
				-100	9,7,7	5,4,4	1,1,1

TABLE 5A-2
(Sheet 3 of 3)

Plate	Heat No./ Slab No.	Drop-weight NDT (Top/Bottom) (°F)	Charpy V-Notch Toughness (Top/Bottom)				
			Orien-tation (L or T)	Charpy Test Temp. (ft-lb)	Energy (ft-lb)	Lateral Expansion (mils)	% Shear
21-1-3	C3065-2	-10/-40	T	+50	33,34,49/41,48,45	30,40,31/40,39,36	30,30,30/40,40,40
			T	+60	43,53,50	37,41,42	40,40,40
			T	+70	56,56,60/51,53,51	50,48,46/46,45,43	50,50,50/50,50,50
			L	+50	63,71,73	56,59,52	60,60,60
			T	+20	36,41,44	31,36,36	30,30,30
			T	+40	44,50,51	36,42,42	40,40,40
			L	+20	63,62,66	51,51,50	40,40,40
			T	+212	91,88,83	84,81,85	99,99,99
				+150	75,77,70	63,68,64	90,90,90
				+100	65,60,65	56,55,56	60,60,60
				+40	56,43,58	44,46,39	40,40,40
				-10	15,23,25	11,18,21	20,20,20
				-100	3,5,5	2,2,3	1,1,1

* Limiting plate. This material is also in the reactor vessel surveillance capsules. See Section 5.3.1.6 for a description of the reactor vessel material surveillance program.

TABLE 5A-3a
(Sheet 1 of 1)

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TABLE 5A-3b
(Sheet 1 of 1)

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TABLE 5A-4
(Sheet 1 of 1)
SURVEILLANCE CAPSULE CONTENTS AND LOCATIONS

Capsule No.	Number of Transverse Charpy Specimens				Number of Specimens Flux Wires	
	Azimuth	Base	HAZ	Weld	Fe	Cu
1	3°	12	12	12	2	2
2	177°	12	12	12	2	2
3	183°	12	12	12	2	2

Note: Surveillance specimen capsule at 3° azimuth location was removed for testing in accordance with the original plant-specific material surveillance program.

CBI NUCLEAR COMPANY

CHICAGO BRIDGE & IRON CO.

WELD PROCEDURE SPECIFICATION

Low Alloy SMA & SA Grooves & Buildup

PROCEDURE NUMBER WPS 323-2F4F6

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DATE 2-17-69

REVISION NO. 8 (7-23-74) RJO

CUSTOMER General Electric Company

PRODUCT NUCLEAR VESSELS (Class 1)

DESCRIPTION Shielded Metal Arc and Submerged Arc Welding of ASME P12B Subgroup 1 Material

REFERENCE SPECIFICATIONS

General WPS 800 Latest Revision
General WPS 820 Latest Revision

PROCEDURE QUALIFICATION

NO.	POSITION	THICKNESS RANGE
1890 (SMA)	V	3/16" to 8"
1891 (SMA)	H	3/16" to 8"
1892 (SMA)	OH, F	3/16" to 8"
1893 (SA-1)	F	3/16" to 8"
2200 (SA-2)	F	3/16" to 8"

8

POST HEAT TREATMENT –
Procedure qualified with 50 hrs. at 1150°F +25°/-50°F.
Postweld heat treatment of the weldment shall be in accordance with a CB&I approved procedure.

BASE METAL –
ASME SA-533 Gr B Class 1 or
SA-508 Class 2
ASME Group No. P12B Subgroup 1

FILLER METAL – ASME

Shielded Metal Arc
Specification – SFA-5.5
Classification – E8018-G
Analysis – A3 (except Ni 0.50 to 1.25)
Usability – F4
Trade Name – Alloy Rods E8018NM

Submerged Arc
See Adjacent Column.

ELECTRICAL CHARACTERISTICS –
See Adjacent Column.

SHIELDING GAS – None

BACKUP GAS – None

FLUX – Linde 124

CUSTOMER APPROVAL

PREHEAT REQUIREMENTS:

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

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

INTERPASS TEMPERATURE REQUIREMENTS:

The interpass temperature shall not exceed 500°F maximum.

FILLER METAL:

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

ELECTRICAL CHARACTERISTICS:

SMA – DCRP

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

ATTACHMENT 5A-1

SURVEILLANCE SPECIMEN
WELD PROCEDURE

NIAGARA MOHAWK POWER CORPORATION
NINE MILE POINT-UNIT 2
FINAL SAFETY ANALYSIS REPORT

APPENDIX 5B

LEAD FACTORS FOR SURVEILLANCE CAPSULES

APPENDIX 5B

LEAD FACTORS FOR SURVEILLANCE CAPSULES (ORIGINAL PLANT-SPECIFIC MATERIAL SURVEILLANCE PROGRAM)

CONCERN

During a NRC conference call with NMPC, the NRC indicated that NMPC needed to provide some additional information regarding the lead factors for the Unit 2 surveillance coupon. Additionally, the NRC wanted some information relative to the justification for the lead factors for Unit 2 and their compliance with 10CFR50 Appendix H; whether test results from another reactor could be utilized for Nine Mile Point; and whether there were constraints on relocating the Unit 2 surveillance capsules. These were followed by a letter dated November 16, 1984, which had specific requests. The information below addresses the staff concerns regarding the Unit 2 lead factors.

RESOLUTION

The Unit 2 neutron materials surveillance samples provide a reactor vessel neutron lead factor of 0.29 for the inside surface of the reactor vessel and 0.41 for the 1/4 T position.

There should be no significant temperature difference between the capsule and RPV inner wall. The downcomer fluid flow, during normal operation, is very turbulent and well mixed before it reaches the vessel beltline.

There is no significant neutron spectrum difference between the surveillance material and RPV inner wall. The calculated shift for any energy group above 1.0 MEV is ± 2.5 percent max.

Currently, 10CFR50 Appendix H requires that "surveillance specimen capsules must be located near the inside vessel wall in the beltline region so that the radiation history duplicates to the extent practical within the physical constraints of the system, the neutron spectrum, temperature history, and maximum neutron fluence experienced by the reactor vessel inner surface."

For Unit 2, surveillance specimen baskets are located about the core midplane at azimuths (i.e., 3 deg, 177 deg, and 183 deg) that are physically advantageous for specimen withdrawal and yet duplicate as much as possible the neutron spectrum and temperature history of the vessel inner surface. These locations were specifically located to ease removal and, thus, reduce occupational radiation to the technicians removing the sample. Specifically, the holder was located to avoid interferences from the jet pumps, core spray lines, and other reactor vessel internals to ensure that the vessel sample could be removed expeditiously.

The ASTM E185-73, ASTM E185-79, and ASTM 185-82 standards, which are incorporated by reference into 10CFR50 Appendix H, provide the standard practice for conducting surveillance tests for light-water-cooled nuclear power reactor vessels. This specification recommends that "the surveillance capsule lead factors (the ratio of the instantaneous neutron flux density at the specimen located to the maximum calculated neutron flux density at the inside surface of the reactor vessel wall) be in the range of 1 to 3." However, it is the Unit 2 position that the effects of neutron radiation on RTNDT and upper shelf energy can still be reliably predicted, given the somewhat lower lead factor readings. Using RG 1.99 as a model, the results of the fracture toughness test data obtained from the specimens can be adjusted to the fluence levels that correspond to present and future periods of vessel service. Therefore, the Unit 2 surveillance program can effectively monitor changes in the fracture toughness properties of the beltline materials.

Unit 2 has confirmed, based upon information from General Electric, that the capsule bracket can be moved to improve the lead factor ranges to about 0.8 to 0.9. However, this relocation could change the flux spectrum in certain energy ranges by as much as 40 percent. Further, moving the capsule bracket raises several other issues that have not yet been evaluated:

- Relocation of the surveillance holder will put it closer to jet pump and core annulus flow stream. This may require a redesign of the neutron surveillance holder.

- Annulus flow obstruction and flow-induced vibration of the holder may change.

- The effect of annulus flow on the dosimeter, attached to the side of the holder and currently held by gravity, is unknown.

- There is a potential for interference of the holders for removal of a jet pump and other internals for their repair.

Unit 2 removed the first capsule at 8.72 effective full-power years (EFPY). The shift and reference transition temperature in the specimen and reactor vessel were determined using RG 1.99 and the results reported in accordance with 10CFR50 Appendix H. Based on analysis of the test results, the P-T limits were determined to be valid for their expressed duration[1]. Future changes to P-T limits will be made in accordance with the requirements of 10CFR50 Appendix G.

Additionally, several operating BWR plants with 251 series (764 bundle) vessels are available to provide supplemental surveillance data. These plants include WNP-2 and LaSalle 1 and 2. The surveillance data for these plants will be utilized to supplement Unit 2 data.

Finally, the NRC has previously accepted the current locations of another similar plant previously licensed. This includes a BWR 6 plant which has a lead factor of 0.4.

In conclusion, the current location of the capsule meets the requirements of 10CFR50 Appendix H. Unit 2 originally committed to supplement the data from Unit 2 with data from other operating BWR 5 251 series vessels. This supplemental data was to be used to provide a trending estimate for Unit 2. The supplemental data evaluation was to consider operational history, fluence values, neutron spectrum, and material similarity.

Subsequent to development of the Unit 2 plant-specific surveillance program, the BWR Vessel and Internals Project (BWRVIP) developed an integrated surveillance program (ISP) to comply with the requirements of 10CFR50 Appendix H. No capsules from the Unit 2 vessel are included in the BWRVIP ISP. Capsules from other plants will be removed and specimens will be tested in accordance with the ISP implementation plan. The results from these tests will provide the necessary data to monitor embrittlement of the Unit 2 vessel. See Section 5.3.1.6 for further description of the BWRVIP ISP.

Reference

1. Niagara Mohawk Power Corporation, Nine Mile Point Nuclear Station - Unit 2, Docket 50-410, 10CFR50 Appendix H, Reactor Vessel Material Surveillance Program Requirements, Report of Test Results. March 8, 2001.