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REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

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REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

5.1 SUMMARY DESCRIPTION

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

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

a. Part of the reactor coolant system, or

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

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

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

    3. The reactor coolant system safety/relief valves.

Section 5.4 discusses the various subsystems to the RCPB.

The nuclear system pressure relief system protects the reactor coolant pressure boundary from damage due to overpressure. To protect against overpressure, pressure-operated relief valves are provided that can discharge steam from the nuclear system to the suppression pool. The pressure relief system also acts to automatically depressurize the nuclear system in the event of a loss-of-coolant accident (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. The major safety consideration for the reactor vessel is concerned with the ability of the vessel to function as a radioactive material barrier. Various combinations of loading are considered in the vessel design. The vessel meets the requirements of various applicable codes and criteria. The possibility of brittle fracture is considered, and suitable design, material selection, material surveillance activities, 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 coast down 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.

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

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

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

The residual heat removal (RHR) system includes a number of pumps and heat exchangers that can be used to cool the nuclear system under a variety of situations. During normal shutdown and reactor servicing, the RHR system removes residual and decay heat. The RHR system allows decay heat to be removed whenever the main heat sink (main condenser) is not available (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 operation is an engineered safety feature for use during a postulated LOCA. This operation is described in Section 6.3. The low-pressure core spray (LPCS) system also provides protection to the nuclear system.

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

5.1.1 SCHEMATIC FLOW DIAGRAM

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

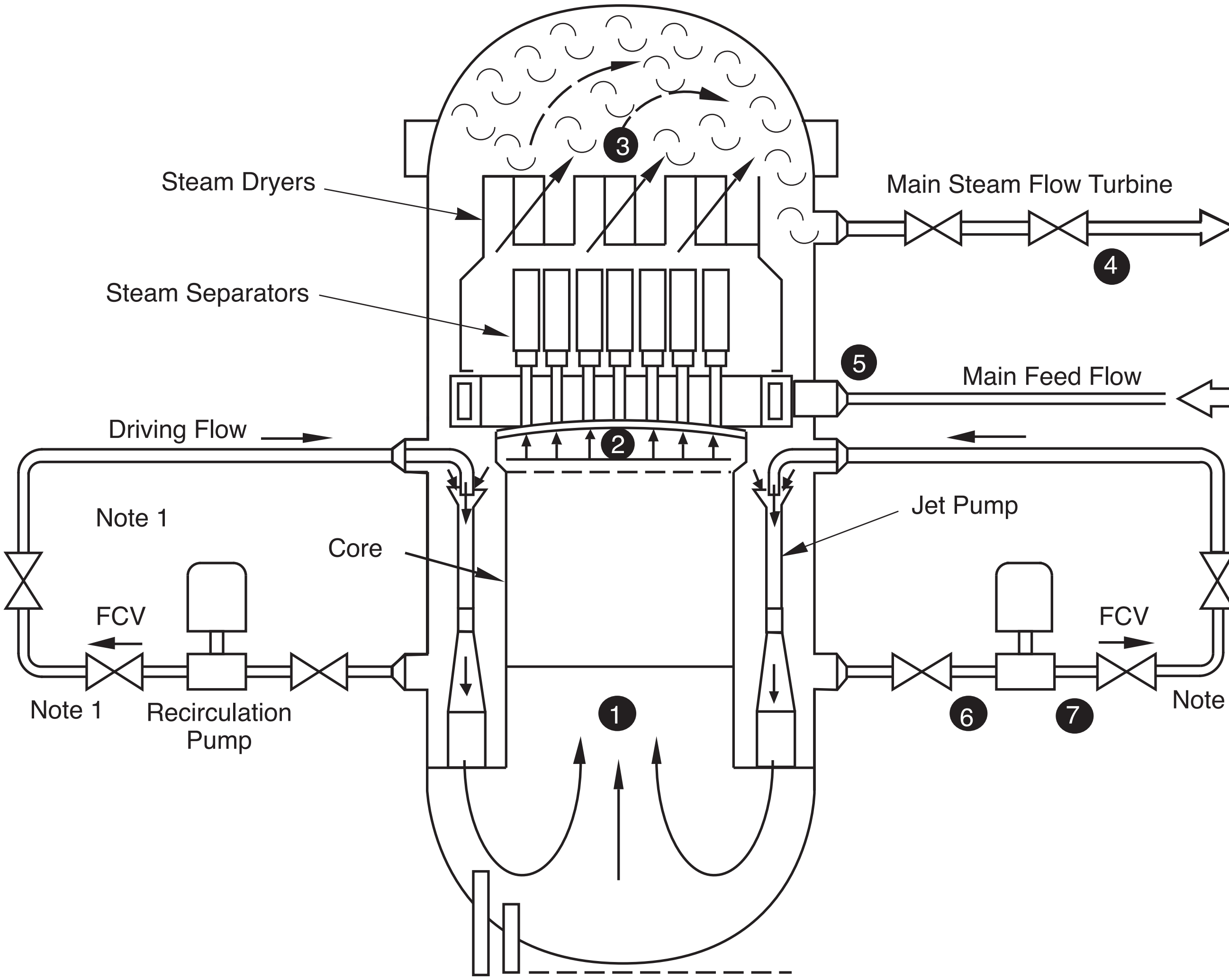
5.1.2 PIPING AND INSTRUMENTATION DIAGRAM

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

a. The nuclear boiler, main steam, and feedwater systems shown in Figure 10.3-2,
b. Recirculation system shown in Figure 5.4-7,
c. RCIC system shown in Figure 5.4-11,
d. RHR system shown in Figures 5.4-16 and 5.4-17,
e. Reactor water cleanup system shown in Figure 5.4-22,
f. HPCS system shown in Figure 6.3-4,
g. LPCS system shown in Figure 6.3-4, and
h. Standby liquid control system shown in Figure 9.3-14.

5.1.3 ELEVATION DRAWING

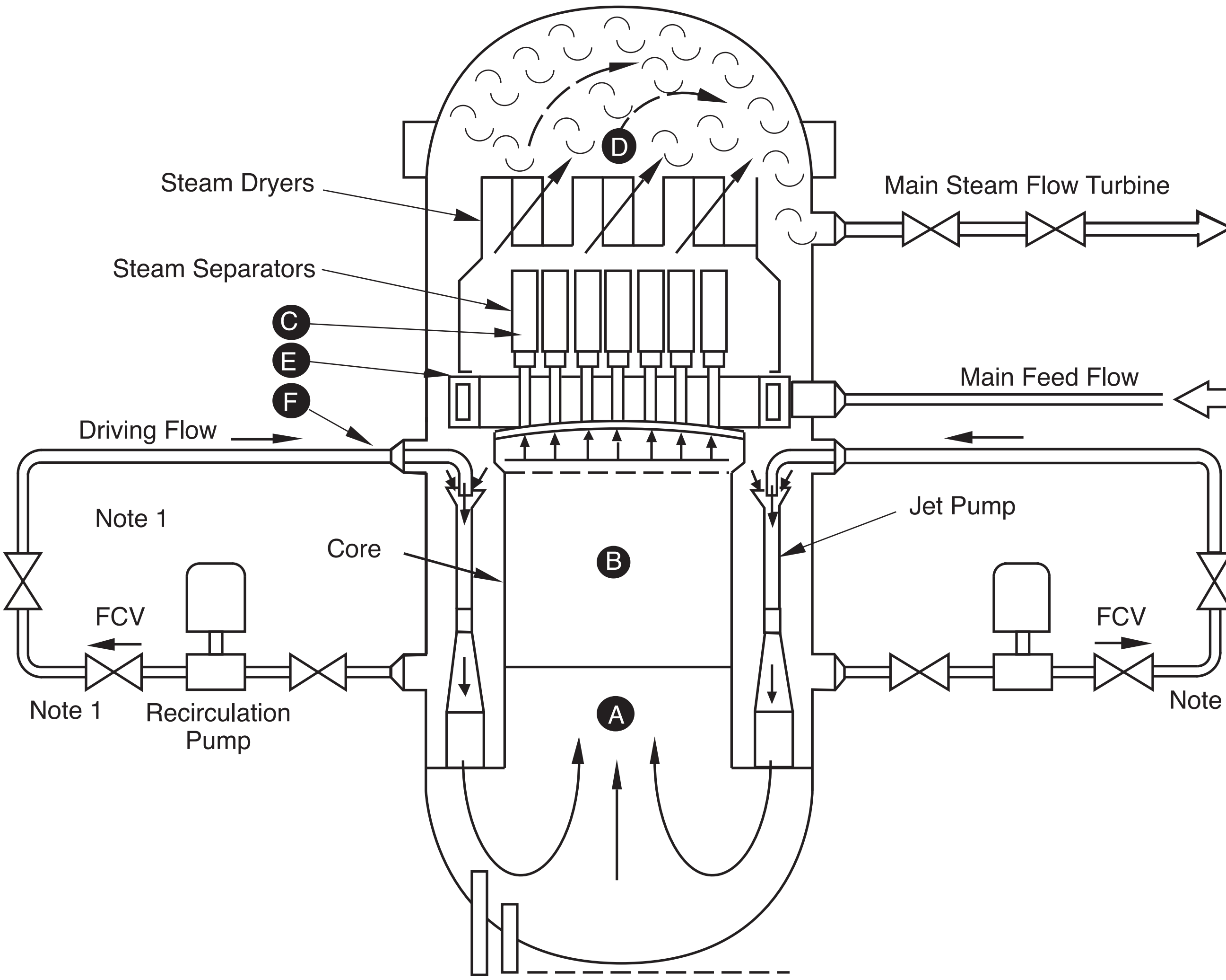
An elevation drawing showing the principal dimensions of the reactor and coolant system in relation to the containment is shown in Figures 1.2-11 and 1.2-12.



	PRESSURE (psia)	FLOW (lb/hr)	TEMP. (°F)	ENTHALPY (Btu/lb)
1. Core Inlet	1069	108.5×10^{6}*	534	528.7
2. Core Outlet	1047	108.5×10^{6}	550	639.9
3. Separator Outlet (Steam Dome)	1035	15.0×10^{6}	549	1191.0
4. Steam Line (2nd Isolation Valve)	1000	15.0×10^{6}	545	1191.0
5. Feedwater Inlet (Includes RWCU Return Flow)	1063	15.2×10^{6}	421	398.5
6. Recirculating Pump Suction	1037	32×10^{6}	534	528.4
7. Recirculating Pump Discharge	1327	32×10^{6}	535	529.8

* Channel Bypass - Nominally 10%
Note 1: The FCVs are kept in mechanically blocked full open position.

Rated Operating Conditions of the
Boiling Water Reactor



Volume of Fluid (ft^3)	
A. Lower Plenum	4010
B. Core	1990
C. Upper Plenum and Separators	2290
D. Dome (Above Normal Water Level)	7160
E. Downcomer Region	5210
F. Recirculating Loops and Jet Pumps	1010

Note1: The FCVs are kept in mechanically blocked full open position.

Columbia Generating Station
Final Safety Analysis Report

Coolant Volumes of the Boiling Water Reactor

Draw. No. 960690.04 | Rev. | Figure 5.1-2

5.2 INTEGRITY OF REACTOR COOLANT PRESSURE BOUNDARY

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

5.2.1 COMPLIANCE WITH CODES AND CODE CASES

5.2.1.1 Compliance with 10 CFR Part 50, Section 50.55a

Table 3.2-1 shows compliance with the rules of 10 CFR Part 50.55a "Codes and Standards." The American Society of Mechanical Engineers (ASME) Code edition, applicable addenda, and component dates are in accordance with 10 CFR 50.55a except for those RCPB components listed in Table 5.2-1. The design, fabrication, and testing of the RCPB components listed in Table 5.2-1 were in accordance with the recognized codes and standards in effect at the time the components were ordered as shown in the table. The code edition and applicable addenda that would be required by strict interpretation of the rules set forth in 10 CFR 50.55a are identified in Table 5.2-1.

Application for Columbia Generating Station (CGS) was filed with the Commission in August 1971. At that time a construction permit was expected before the end of the 1972, but requests for additional seismic data in August 1972 caused the issuance of the construction permit to go beyond the end of the year to March 19, 1973. As is common practice in the utility industry, Energy Northwest proceeded with the engineering, design, and material and components procurement in anticipation of the award of a construction permit to meet construction schedules. Had the construction permit been issued as initially expected, the requirements of 10 CFR 50.55a would have been met to the letter of the law.

However, in each instance of exception the ASME Code version applied was one addenda earlier (6 months) than the code version required by the rules of 10 CFR 50.55a. The changes embodied in the later ASME Code addenda were reviewed. It was concluded that the addenda required by the rules of 10 CFR 50.55a affected documentation format but imposed no new technical requirements or changes in quality control procedures from the code version applied in the procurement of the components. The level of safety and quality provided by conformance to the earlier code edition and addenda applied in procurement is equivalent to that which would be required by strict application of the rules of 10 CFR 50.55a. The effort and expense of recertification of these components, which had all been shipped to the construction site, would not have provided a compensating increase in the level of safety and quality.

5.2.1.2 Applicable Code Cases

The reactor pressure vessel (RPV) and appurtenances and the RCPB piping, pumps and valves, were designed, fabricated, and tested in accordance with the applicable edition of the ASME

Code, Section III, including the addenda that were mandatory at the order date for the applicable components. This is in compliance with the intent of Regulatory Guides 1.84 and 1.85. Section 50.55a of 10 CFR Part 50 requires code case approval only for Class 1 components. These code cases contain requirements or special rules which may be used for the construction of pressure-retaining components of Quality Group Classification A. The various ASME Code case interpretations that were applied to components in the RCPB are listed in Table 5.2-2. Code cases listed in Table 5.2-2 are those used in the original construction of CGS. Other code cases that are adopted for use, as approved by Regulatory Guides 1.147, 1.84, 1.85, or specifically approved by the Regulatory Authority for use at CGS, are specified in the component's design specification as required by ASME Section III.

5.2.2 OVERPRESSURIZATION PROTECTION

5.2.2.1 Design Bases

Overpressurization protection is provided in conformance with 10 CFR 50, Appendix A, General Design Criterion 15.

5.2.2.1.1 Safety Design Basis

The nuclear pressure-relief system is designed to

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

b. Provide automatic depressurization for small breaks in the nuclear system occurring with maloperation of the high-pressure core spray (HPCS) system so that the low-pressure coolant injection (LPCI) and the low-pressure core spray (LPCS) systems can operate to protect the fuel barrier (see Section 6.3.2.2.2),

c. Permit verification of its operability, and

d. Withstand adverse combinations of loadings and forces resulting from operation during abnormal, accident, or special event conditions.

5.2.2.1.2 Power Generation Design Bases

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

a. Discharge to the containment suppression pool, and

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

5.2.2.1.3 Discussion

The ASME Boiler and Pressure Vessel Code (B&PV Code) requires that each component designed to meet Section III be protected from overpressure under upset conditions. The code allows a peak allowable pressure of 110% of design pressure under upset conditions. The code specifications for safety valves require that (a) the lowest safety valve setpoint will be set at or below design pressure, and (b) the highest safety valve setpoint will be set so that total accumulated pressure does not exceed 110% of the design pressure for upset conditions. The SRVs are designed to open by means of either of two modes of operation as discussed in Chapter 15. The safety (spring) setpoints are listed in Table 5.2-3 and satisfy the first of the above-mentioned ASME Code specifications for safety valves because all valves open at less than the nuclear system design pressure of 1250 psig.

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

The following detailed criteria are used in selection of SRVs:

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

b. Valves must qualify for 100% of nameplate capacity credit for overpressure protection function, and

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

The SRV discharge piping is constructed in accordance with the ASME Code, Section III, 1971 Edition through the Winter 1973 Addenda.

5.2.2.1.4 Safety Valve Capacity

The safety valve capacity of this plant is adequate to limit the primary system pressure, including transients, to the requirements of the ASME B&PV Code, Section III, 1971 Edition through the Summer 1971 Addenda.

Table 5.2-4 lists the systems which could initiate during the safety valve capacity overpressure event.

5.2.2.2 Design Evaluation

5.2.2.2.1 Method of Analysis

To design the pressure protection for the nuclear boiler system, extensive analytical models representing all essential dynamic characteristics of the system are simulated on a large computing facility. These models include the hydrodynamics of the flow loop, the reactor kinetics (either a point kinetics or a one-dimensional kinetics simulation of the reactor core dynamics), the thermal characteristics of the fuel and its transfer of heat to the coolant, and all the principal controller features, such as feedwater flow, recirculation flow, reactor water level, pressure, and load demand. These are presented with all their principal nonlinear features in models that have evolved through extensive experience and favorable comparison of analysis with actual boiling water reactor (BWR) test data.

A detailed description of the models is documented in licensing topical report Reference 5.2-1. Safety/relief valves are simulated in the nonlinear representation, and the models thereby allow full investigation of the various valve response times, valve capacities, and actuation setpoints that are available in applicable hardware systems.

The typical capacity characteristic as modeled is represented in Figure 5.2-1 for the spring mode of operation. The associated turbine bypass, turbine control valve (TCV), and main steam isolation valve (MSIV) characteristics are also simulated in the models.

The associated bypass, TCV, main steam isolation characteristics, and anticipated transients without scram (ATWS) pump trip are also represented fully in the models.

5.2.2.2.2 System Design

The overpressure protection system must accommodate the most severe pressurization transient. There are two major transients, the closure of all MSIVs and a turbine generator trip with a coincident failure of the turbine steam bypass system valves, that represent the most severe abnormal operational transients resulting in a nuclear system pressure rise. The evaluation of transient behavior with final plant configuration has shown that the isolation valve closure is slightly more severe when credit is taken only for indirect derived scrams; therefore, it is used as the overpressure protection basis event and shown in Figure 5.2-2. Table 5.2-5 lists the sequence of events of the various systems assumed to operate during the main steam line isolation closure with flux scram event.

Compliance to ASME Code overpressure protection requirements for introduction of GE14 fuel has been conservatively demonstrated for the limiting overpressure event. The GE thermal-hydraulic and nuclear coupled transient code ODYN (Reference 5.2-1) was used to obtain system response and peak vessel pressure. The setpoints are listed in Table 5.2-3. The evaluation, based on reactor operation at 102% of uprated power, end-of-cycle nuclear

dynamic parameters, an initial dome pressure of 1050 psia (15 psia above the nominal uprated dome pressure), six SRVs with lowest safety setpoints out of service, and SRV opening pressures at 3% above nominal setpoint values resulted in a maximum reactor pressure of 1341 psig.

The scram reactivity curve is shown in Figure 5.2-2.

5.2.2.2.3 Evaluation of Results

5.2.2.2.3.1 Safety Valve Capacity. The required SRV capacity is determined by analyzing the pressure rise from an MSIV closure with flux scram transient. The plant is assumed to be operating at the turbine-generator design conditions at a maximum vessel dome pressure of 1050 psia. The analysis hypothetically assumes the failure of the direct MSIV position scram. The reactor is shut down by the backup, high neutron flux scram. For the analysis, the spring-action safety valve setpoints used are in the range of 1225 to 1256 psia. The ODYN analysis indicates that the design valve capacity is capable of maintaining adequate margin below the peak ASME Code allowable pressure in the nuclear system (1375 psig). Figure 5.2-2 shows the result of the ODYN analysis. The sequence of events in Table 5.2-5, assumed in these analyses, were investigated to meet code requirements and to evaluate the pressure relief system exclusively.

Under Section III of the ASME B&PV Code, credit can be allowed for a scram from the reactor protection system. In addition, credit is also taken for the protection circuits which are indirectly derived when determining the required SRV capacity.

The backup reactor high neutron flux scram is conservatively applied as a design basis in determining the required capacity of the pressure relieving dual purpose SRVs. Application of the direct position scrams in the design basis could be used since they qualify as acceptable pressure protection devices when determining the required SRV capacity of nuclear vessels under the provisions of the ASME Code. The SRVs are operated in a relief mode (pneumatically) at setpoints lower than those specified under the safety function. This ensures sufficient margin between anticipated relief mode closing pressures and valve spring forces for proper seating of the valves.

The parametric relationship between peak vessel (bottom) pressure and SRV capacity for the MSIV transient with high flux scram is described in Figure 5.2-3. Also shown in Figure 5.2-3 is the peak vessel (bottom) pressure for position scram with 18-valve capacity. Pressures shown for flux scram will result only with multiple failure in the redundant direct scram system.

The time response of the vessel pressure to the MSIV transient with flux scram is illustrated in Figure 5.2-4. This shows that the pressure at the vessel bottom exceeds 1250 psig for less than 7 sec and does not reach the limit of 1375 psig.

5.2.2.2.3.2 Pressure Drop in Inlet and Discharge. Pressure drop in the piping from the reactor vessel to the valves is taken into account in calculating the maximum vessel pressures.

Pressure drop in the discharge piping to the suppression pool is limited by proper discharge line sizing to prevent back-pressure on each SRV from exceeding 40% of the valve inlet pressure, thus ensuring choked flow in the valve orifice and no reduction of valve capacity due to the discharge piping. Each SRV has its own separate discharge line.

5.2.2.2.3.3 Reload Specific Confirmatory Analysis. The calculated vessel pressure for MSIV inadvertent closure may be dependent upon the fuel design and core loading pattern. Compliance with the ASME upset limit is demonstrated by cycle-dependent analysis just prior to the operation of that cycle. The results are reported in Supplemental Reload Licensing Report (Reference 5.2-11).

5.2.2.3 Piping and Instrumentation Diagrams

See Figure 5.2-5 which shows the schematic location and number of pressure-relieving devices. The schematic arrangement of the SRVs is shown in Figures 5.2-6 and 5.2-7.

5.2.2.4 Equipment and Component Description

5.2.2.4.1 Description

The nuclear pressure relief system consists of SRVs located on the main steam lines between the reactor vessel and the first isolation valve within the drywell.

Chapter 15 discusses the events which are expected to activate the primary system SRVs. The chapter also summarizes the number of valves expected to operate during the initial blowdown of the valves and the expected duration of this first blowdown. For several of the events it is expected that the lowest set SRV will reopen and reclose as generated heat drops into the decay heat characteristics. The pressure increase and relief cycle will continue with lower frequency and shorter relief discharges as the decay heat drops off and until such time as the residual heat removal (RHR) system can dissipate this heat. The duration of each relief discharge should, in most cases, be less than 30 sec. Remote manual actuation of the valves from the control room is recommended to minimize the total number of these discharges, with the intent of achieving extended valve seat life and reducing challenges to the SRV.

A schematic of the main SRV is shown in Figure 5.2-8. It is opened by either of two modes of operation:

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

pressure force exceeds the spring force. Figure 5.2-9 depicts typical valve lift versus opening time characteristics; and

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

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

For overpressure SRV operation (self-actuated or spring lift mode), the spring load establishes the safety valve opening setpoint pressure and is set to open at setpoints designated in Table 5.2-3. In accordance with the ASME Code, full lift in this mode of operation is attained at a pressure not greater than 3% above the setpoint.

To prevent backpressure from affecting the spring lift setpoint, each valve is provided with a bellows and balancing piston to counteract the effects of any static backpressure which may be present in the discharge line before the valve is opened to discharge steam. The bellows isolates steam in the valve discharge chamber from the valve's internals. If the bellows fails, the balancing piston serves as a functional backup by presenting an effective piston area to the back pressure equal to the valve seat area, thus balancing it so there is essentially no net back pressure effect on the setpoint (Figure 5.2-8).

The safety function of the SRV is a backup to the relief function described below. The spring-loaded valves are designed and constructed in accordance with ASME III, 1971 Edition, Paragraph NB-7640, as safety valves with auxiliary actuating devices.

Each SRV is provided with its own pneumatic accumulator and inlet check valve to provide high assurance the valve will actuate in the power-actuated (relief) mode when its pneumatic solenoid valve is energized. The pneumatic accumulator has sufficient capacity to provide one SRV actuation at approximately 1000 psig valve inlet pressure. Although no credit is taken under ASME Code Section III for overpressure protection by the SRVs in their power-actuated mode, power actuation of the SRV will limit peak reactor pressure in the majority of overpressure transients.

Safety/relief valve actuation in the relief mode is initiated by pressure switches (one per valve) which sense reactor steam space pressure at lower values than the spring mode inlet steam opening pressure. The pressure switches initiate the opening of the SRVs by energizing the pneumatic solenoids (one per valve) at the relief setpoints designated in Table 5.2-3.

When the solenoid is actuated, the delay time, maximum elapsed time between receiving the overpressure signal at the valve actuator and the actual start of valve motion, will not exceed

0.1 sec. The maximum full stroke opening time will not exceed 0.15 sec with 1000 psig steam at the valve inlet.

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

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

a. 340°F for a 3-hr period, at drywell design pressure,
b. 320°F for an additional 3-hr period, at drywell design pressure,
c. 250°F for an additional 18-hr period, at 25 psig drywell pressure, and
d. 200°F during the next 99 days at 20 psig drywell pressure.

The automatic depressurization system (ADS) utilizes selected SRVs for depressurization of the reactor (see Section 6.3). Each of the SRVs utilized for automatic depressurization is equipped with an air accumulator and check valve arrangement. These accumulators ensure that the valves can be held open following failure of the air supply to the accumulators. The designed pneumatic supply to the ADS accumulator is such that, following a failure of the safety-related pneumatic supply to the accumulator, at least two valve actuations can occur with the drywell at 70% of design pressure. For a discussion of the noninterruptible air supply to the ADS valves, see Section 9.3.1. Three ADS SRVs and their associated solenoid pilot valves (SPV) are qualified for the full post-LOCA time frame for long-term cooling. All other SRVs and their SPVs are qualified for 24 hr post-LOCA to provide overpressure protection capability.

The valve position indication (VPI) and the tailpipe temperature indication systems are discussed in Section 7.5.2.

Each SRV discharges steam through a discharge line to a point below the minimum water level in the suppression pool. Safety/relief valve discharge line piping from the SRV to the suppression pool consists of two parts. The first part is attached at one end to the SRV and at its other end penetrates and is welded to a 28-in. downcomer (considered a pipe anchor). The main steam piping, including this portion of the SRV discharge piping, is analyzed as a complete system. This portion of the SRV discharge line is classified as Quality Group C and Seismic Category I down to the jet deflector plate just above the diaphragm floor (through which it is rigidly guided) and Quality Group B and Seismic Category I from the jet deflector plate to the downcomer.

The second part of the SRV discharge piping extends from the downcomer (anchor) to the suppression pool. Because of the anchor on this part of the line, it is physically decoupled from the main steam header and is, therefore, analyzed as a separate piping system. In analyzing this part of the discharge piping in accordance with the requirements of Quality Group B, the following load combination was considered as a minimum:

a. Pressure and temperature,
b. Dead weight, and
c. Fluid dynamic loads due to SRV operation.

As a part of the preoperational and startup testing of the main steam lines, movement of the SRV discharge lines were inspected with negligible vibration observed.

The SRV discharge piping is designed to limit valve outlet pressure to 40% of maximum valve inlet pressure with the valve open. Water in the line more than a few feet above suppression pool water level would cause excessive pressure at the valve discharge when the valve is again opened. For this reason, redundant 10-in. vacuum relief valves are provided on each SRV discharge line to prevent drawing an excessive amount of water up into the line as a result of steam condensation following termination of relief operation. Each vacuum relief valve pair is situated with the valves in parallel, the discharge being routed to a common tee in the SRV discharge line.

The nuclear pressure relief system automatically depressurizes the nuclear system sufficiently to permit the LPCI and LPCS systems 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 7.3.1.1.1.

5.2.2.4.2 Design Parameters

Table 5.2-6 lists design temperature, pressure, and maximum test pressure for the RCPB components. The specified operating transients for components within the RCPB are given in Section 3.9. Refer to Section 3.7 for discussion of the input criteria for design of Seismic Category I structures, systems, and components.

A summary of the number of cycles for transients used in design and fatigue analysis is listed in Table 3.9-1 and categorized under the appropriate design condition (i.e., normal, upset, emergency, and faulted).

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

5.2.2.4.2.1 Safety/Relief Valve. The discharge area of the valve is 16.117 in.^{2} and the coefficient of discharge KD is equal to 0.966, as certified by the National Board of Boiler and Pressure Vessel Inspectors.

The design pressure and temperature of the valve inlet and outlet are 1250 psig at 575°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. Cyclic testing has demonstrated that the valves are capable of at least 60 actuation cycles between required maintenance.

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

5.2.2.5 Mounting of Pressure Relief Devices

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

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

5.2.2.6 Applicable Codes and Classification

The RCPB overpressure protection system is designed to satisfy the requirements of Section III, Subsection NB, of the ASME B&PV Code. The general requirements for protection against overpressure as given NB-7120 of Section III of the code recognize that RCPB 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.

5.2.2.7 Material Specification

Pressure retaining components of SRVs are constructed only from ASME Section III, Class 1 designated materials.

5.2.2.8 Process Instrumentation

Overpressure protection process instrumentation is listed in Table 1 of Figure 5.2-5 and shown in Figure 10.3-2.

5.2.2.9 System Reliability

Overpressure protection system reliability is principally a function of the SRVs in their spring-opening mode of operation. No credit is taken in the ASME Code Section III required overpressure protection report for power actuation of the SRVs to provide protection against overpressure.

Section 5.2.2.10 discusses the inspection and testing conducted to ensure high SRV reliability. As demonstrated by the extensive qualification and production testing, the valves are very reliable.

In addition to SRV testing to ensure high SRV quality, an extensive in-depth quality assurance program was followed in the manufacture and production testing of the valves to provide assurance of high quality.

A significant amount of BWR operating experience was been accumulated on this type of SRV, approximately 150 individual valve years, only one “stuck-open relief valve” had occurred. This was due to an air solenoid valve sticking open after it was deenergized, thus holding the SRV open in the power-actuated mode. Proper maintenance procedures are incorporated into the instruction manual to preclude recurrence.

This type of SRV has demonstrated good inservice operability similar to that demonstrated by the qualification test program.

In summary, this type of SRV has demonstrated excellent reliability, both in qualification testing and in actual BWR operation.

5.2.2.10 Inspection and Testing

To verify the design of the SRV used will reliably operate, several SRVs were subjected to qualification test programs. These qualification test programs demonstrated the design of the valve is capable of performing its overpressure protection function under normal, upset, emergency, and faulted conditions and its designated mechanical motion(s) to fulfill its safety function to shut down the plant or mitigate the consequence of a postulated event. To ensure that valves to be installed are operable, each valve is manufactured, inspected, and production tested in accordance with quality control procedures to verify compliance with both ASME Code and operability assurance acceptance criteria.

The SRV design used at CGS successfully completed the following qualification tests:

a. Life Cycle Test

   Following the prequalification production tests, each modified SRV was then subjected to life cycle qualification tests with saturated steam conditions, in accordance with GE specification 22A6595. This included approximately 300 relief (power) and safety (pressure) actuations to demonstrate and characterize each valve for acceptable BWR service. Tests parameters included:

   1. Seat tightness/leakage characteristics,

2. Set pressure,

3. Opening and closing response time,

4. Blowdown,

5. Safety/relief valve lift-achieving rated flow capacity lift during each activation,

6. Safety/relief valve reclosure without chattering, disc oscillation, or sticking open, and

7. Capability to open without inlet steam when activated on demand.

Test conditions were varied according to facility capability to ensure valve operability across the design limits to which the SRV may be subjected while in service. These included temperature, pressure ramp rates, pneumatic operating pressure, solenoid voltage, inlet pressure, and the dynamically imposed backpressure.

Test results indicate essentially zero leakage for both the relief (power) and safety (pressure) modes of SRV operation. All valves demonstrated seat-tightness capability to meet the 20 lb/hr specification limit under saturated steam conditions. Each valve demonstrated safety actuation within the nameplate value plus 1% at a confidence level of 0.95. The response is also linear with ambient temperature in the negative direction; i.e., at temperatures above 135°F the actual pop pressure is lower than the nameplate value. The temperature correction value is 0.2 psi/°F for this SRV. Set pressure is independent of ramp rate variance. Response of the SRV is directly related to the effective differential pressure force acting to open the SRV; therefore, outlet static pressure at the exit can be accurately accounted for.

Opening times were as follows during the test set up:

Safety actuation time - $0.020 \leq t \leq 0.30$ sec

Relief actuation time - $0.020 \leq t \leq 0.15$ sec

Actual installation times could result in a delay time > 0.10 sec due to wire lengths and other non-SRV wire losses. Closing times were:

Safety actuation - none, controlled by blowdown requirement.

Relief actuation -	time to deenergize solenoid	< 0.90 sec
	disc travel after solenoid was deenergized	< 1.50 sec

Blowdown within the required range of 2% to 11% was demonstrated. Each SRV is adjusted by full flow testing for acceptable blowdown.

Qualification test results demonstrate the SRV will open to rated capacity lift in either the relief or safety modes of operation when actuated.

The SRV reclosure was demonstrated throughout the qualification tests without sticking, chatter, or disc oscillation during the closure stroke. When inlet pressure was increased to repressurize to the set pressure, the SRV reactuated to the full open position. The modified SRV will open to its full rated capacity lift position when operated in the relief mode with the inlet pressure at zero psig, thus demonstrating its emergency operability capability.

Six SRVs were included in this life cycle qualification test program. Test anomalies corrected during this demonstration do not invalidate the adequacy of the test results obtained; the finalized modified SRV design is considered acceptable for BWR main steam applications.

b. Seismic and Moment Transfer Test

One valve specimen was subjected to operating basis earthquake (OBE) and safe shutdown earthquake (SSE) accelerations and flanged end connection moment loading with valve inlet pressurized with saturated steam. Valve operability was demonstrated during and after application of loading. Maximum test loads were 8×10^5 in. pound moment at valve inlet and 6×10^5 in. pound moment at valve outlet. Seismic accelerations of 5.0g horizontal and 4.2g vertical are the established maximum for any frequency between 5 to 200 Hz unless otherwise specified for a smaller frequency range.

c. Emergency Environmental Qualification Test

The solenoid valves and the pneumatic actuator assembly were subjected to a test sequence as follows:

1. Thermal aging equivalent to 343°F for 96 hours,

2. Radiation aging to greater than or equal to 30×10^6 rads,

3. Mechanical aging for 1000 cycles (500 per solenoid),

4. Seismic testing as described in item b. above,

5. Exposure to emergency environmental conditions of 340°F at 65 psig decreasing to 250°F at 25 psig for 4 days, and

6. Separate solenoid valve test
   340°F, 3 hrs, 45 psig
   320°F, 3 hrs, 45 psig
   250°F, 18 hrs, 25 psig
   200°F, 99 days, 20 psig.

Operability of the actuator assembly was demonstrated during and after exposure to the emergency environment.

d. Low-Pressure Water Discharge Test

Low-pressure water discharge tests as described and reported in GE Report NEDE-24988 to satisfy the requirements of II.D.1 of NUREG-0737.

Test reports/records of the above qualification tests are available for inspection.

Each SRV is production tested at the vendor's shop to ensure, by demonstration, each SRV manufactured will reliably perform its required function(s). The SRV production test consist of

a. Inlet and outlet hydrostatic tests at specified conditions to satisfy ASME Code requirements,

b. Emergency operability test to verify capability of actuator to open the SRV without inlet pressure applied to the valve,

c. Actuator system leakage test to assure pneumatic leaktightness is compatible with plant air system make-up requirements,

d. Nitrogen set pressure and leakage test to rough adjust setpoint and ensure seat quality of seating surface prior to steam tests (optional),

e. Set pressure and blowdown test under thermally stabilized and saturated steam conditions,

f. Response time tests to verify relief opening and closing times under thermally stabilized and saturated steam conditions, and

g. Steam leakage tests to verify leaktightness.

The valves are normally installed as received from the factory providing there is no apparent evidence of damage during transportation, handling, and storage. For valves stored longer than one year, it is recommended they be recertified to ensure operability. The GE equipment specification requires certification from the valve manufacturer that design and performance requirements have been met.

Testing to satisfy the ASME Code requirements is normally performed in situ. Testing can be performed locally or remotely. The local test method is conducted using a test fixture that is temporarily mounted on the SRV and then removed on completion of the test. Remote testing is accomplished using a permanently mounted pneumatic head assembly that is controlled by a remote computer. This method does not require any personnel entry into the containment for the purpose of testing.

During the startup test program, all of the main steam SRVs were tested for proper operation. These tests include a documentation review to ensure that the valves were properly installed, properly handled during transportation, storage, and installation, and were properly maintained as to cleanliness prior to performance of any tests. In addition, the air accumulator capacity, SRV nameplate set pressure, and capacity were compared with the system design documentation for compliance.

Actual mechanical tests included an operability check of the SRV discharge line vacuum breakers, actuation of the individual SRVs by each remote manual switch (main control room and/or remote shutdown panel) to demonstrate full lift, smooth stroke, and opening time characteristics, actuation of each SRV in the relief mode by stimulating its pressure switch, and a demonstration that each SRV accumulator (ADS and/or normal) has sufficient capacity to operate the SRV air actuator as required by the system design documentation. Finally, the ADS logic was fully tested for proper performance. Note that only the air actuator was exercised during many of the startup tests. This minimizes valve wear and unnecessary maintenance.

During the power ascension phase of the startup test program, each SRV was manually actuated at approximately 250 psig reactor pressure to demonstrate valve operability. At approximately 50% power each SRV was actuated a second time to measure discharge capacity and to demonstrate that no blockage in the SRV discharge line existed.

At commercial turnover the scope of SRV testing was governed by ASME B&PV Code Section XI, Article IWV and the Technical Specifications. This article specifies the rules and requirements for inservice testing to verify operational readiness of the SRVs. This code section is applied to both ADS and non-ADS valves alike. Supplemental tests of the ADS

valves each operating cycle are required by the Technical Specifications. Applying Section XI, the SRV test schedule (in part) is as follows:

Time Period (Cycle)	Number of Valves Tested	Total Tested	Elapsed Time (years)
1	6	6	1.5
2	4	10	2.5
3	4	14	3.5
4	4	18	4.5
5	4	4	1.0
6	4	8	2.0
7	4	12	3.0
8	4	16	4.0
9	2	18	5.0

Note that following the return to service of the testing SRVs, an operability demonstration will be performed in compliance with Section XI, Article IWV-3200.

This combination of the startup test program, Technical Specifications surveillance, and inservice inspection testing satisfies industry standards for SRV operability demonstrations. Energy Northwest participated in the BWR Owners' Group for TMI concerns on SRV reliability. The final test program description was submitted to the NRC by the BWR Owners' Group and is endorsed by Energy Northwest.

5.2.3 REACTOR COOLANT PRESSURE BOUNDARY MATERIALS

5.2.3.1 Material Specifications

Table 5.2-7 lists the principal pressure retaining materials and the appropriate material specifications for the RCPB components.

5.2.3.2 Compatibility with Reactor Coolant

5.2.3.2.1 Pressurized Water Reactor Chemistry of Reactor Coolant

Not applicable to BWRs.

5.2.3.2.2 Boiling Water Reactor Chemistry of Reactor Coolant

Regulatory Guide 1.56 compliance is addressed in Section 1.8.

Reactor feedwater (RFW) quality is maintained in accordance with the Licensee Controlled Specifications (LCS) and as described in Section 10.4.6.

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

Periodically an On-Line NobleChem™ application will be performed to create a catalytic layering of the noble metal platinum to reduce the hydrogen injection rate required to achieve a low electrochemical corrosion potential (ECP). The low ECP achieves intergranular stress corrosion cracking (IGSCC) and irradiation assisted stress corrosion cracking (IASCC) protection while minimizing the effects of high dose rates attributed to regular hydrogen injection rates.

When conductivity is in its normal range, pH, chloride, and other impurities affecting conductivity will also be within their normal range. When conductivity becomes abnormal, chloride measurements are made to determine whether or not they are also out of their normal operating values. Conductivity could be high due to the presence of a neutral salt, which would not have an effect on pH or chloride. In such a case, high conductivity alone is not a cause for shutdown. In some types of water-cooled reactors, conductivities are high because of the purposeful use of additives. In BWRs, however, where no additives which significantly affect conductivity are used and where near neutral pH is maintained, conductivity provides a good and prompt measure of the quality of the reactor water. A depleted zinc oxide (DZO) skid is connected to the RFW system which maintains DZO concentration in reactor water. This has a small effect on conductivity. Significant changes in conductivity provide the operator with a warning mechanism so he can investigate and remedy the condition before reactor water limits are reached. Methods available to the operator for correcting the off-standard condition include operation of the reactor cleanup system, reducing the input of impurities, and placing the reactor in the cold shutdown condition. The major benefit of cold shutdown is to reduce the temperature-dependent corrosion rates and provide time for the cleanup system to reestablish the purity of the reactor coolant.

During normal plant operation, the dynamic oxygen equilibrium, in the reactor vessel water phase, established by steam-gas stripping and radiolytic formation (principally) rates, corresponds to a nominal value of approximately $\boxed{\text{200 ppb}}$ (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 reactor water cleanup (RWCU) system is provided for removal of feedwater input impurities plus corrosion and fission products originating from primary system components. The cleanup process consists of filtration and ion exchange and serves to maintain a high level of water purity in the reactor coolant.

Additional water input to the reactor vessel originates from the control rod drive (CRD) cooling water. The CRD water is of feedwater quality. Additional filtration of the CRD water to remove insoluble corrosion products takes place within the CRD system prior to entering the drive mechanisms and reactor vessel.

An iron addition system is used to inject an iron oxalate/demineralized water solution into the suction line of the condensate booster pumps. The injection flow rate is extremely small when compared to condensate system flow rate. This iron injection system will have a negligible affect on the oxygen concentration in the RFW.

A hydrogen injection system is installed across the condensate booster pumps. This hydrogen injection system will have a negligible affect on the oxygen concentration in the RFW.

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 reactor coolant water could contain up to 8 ppm dissolved oxygen.

Conductivity of the primary coolant is continuously monitored with instruments connected to the reactor water recirculation loop and the RWCU system inlet. The effluent from the RWCU system is also monitored for conductivity on a continuous basis. These measurements provide reasonable surveillance of the reactor coolant.

Grab sample points are provided at the locations shown in Table 5.2-8, for special measurements such as pH, oxygen, chloride, and radiochemical content.

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 (see Figure 5.2-10). Values for these compounds essentially bracket values of other common chloride salts or mixtures at the same chloride concentration. Surveillance requirements are based on these relationships.

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

The sampling frequency established for primary coolant at normal conductivity levels is adequate for instrument checks and routine audit purposes. When specific conductance increases and higher chloride concentrations are possible or when continuous conductivity monitoring is unavailable, sampling frequency is increased according to LCS.

The primary coolant conductivity monitoring instrumentation, ranges, sensor, and indicator locations are shown in Table 5.2-8. The sampling is coordinated in a reactor sample station especially designed with constant temperature control and sample conditioning and flow control equipment.

Water Purity During a Condensate Leakage

Due to improved water quality limits, any appreciable circulating water inleakage would result in water chemistry conditions outside acceptable limits and require action(s) to return the water quality to within applicable limits for continued plant operation.

5.2.3.2.3 Compatibility of Construction Materials with Reactor Coolant

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

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

b. Nickel base alloys - Inconel 600 and Inconel X750 and Inconel 82 and 182 weld metal,

c. Carbon steel and low alloy steel,

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

e. Cobalt, chromium, nickel, and iron based alloy hardfacing material

All of these materials of construction are generally 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 on any of the materials from allowable contaminant levels in the high purity reactor coolant. Radiolytic products in the BWR have no adverse effects on the construction materials.

The recirculation system piping and normally flooded sections of the reactor vessel are coated as needed utilizing the GEH On-Line NobleChem™ application process with a microscopic layer of noble metals. This coating serves to prevent as well as mitigate IGSCC by eliminating the dissolved oxygen at the metal surface when an amount of hydrogen gas is added in a molar ratio of greater than 2 to 1 hydrogen to oxygen.

Type 304 stainless steel has been replaced with type 316L stainless steel in the recirculation inlet line safe ends. The bypass lines and the CRD hydraulic return line were eliminated and nozzles capped. The core spray lines are fabricated of carbon steel. The piping components that do not comply with the requirements of the Generic Letter 88-01 (GL 88-01), NRC Position on IGSCC BWR austenitic Stainless Steel Piping, will be subjected to the augmented inspection requirements of GL 88-01 as modified in Energy Northwest response (see Section 5.2.4 and Tables 5.2-9 and 5.2-10).

5.2.3.2.4 Compatibility of Construction Materials with External Insulation and Reactor Coolant

The materials of construction exposed to external insulation are

a. Solution annealed austenitic stainless steels (e.g., types 304, 304L, and 316), and

b. Carbon and low alloy steel.

Two types of external insulation are used. Reflective metal insulation used does not contribute to any surface contamination and has no effect on construction materials. The fibrous insulation used meets the requirements of Regulatory Guide 1.36.

DZO and iron are additives in the BWR coolant. Leakage would expose materials to high purity demineralized water, DZO, and iron. Exposure to demineralized water, DZO, and iron would cause no detrimental effects.

5.2.3.3 Fabrication and Processing of Ferritic Materials and Austenitic Stainless Steels

Fracture toughness requirements for the ferritic materials used for piping and valves (no ferritic pumps in RCPB) of the RCPB were as follows:

Safety/relief valves were exempted from fracture toughness requirements because Section III of the 1971 ASME B&PV Code did not require impact testing on valves with inlet connections of 6 in. or less nominal pipe size.

Main steam isolation valves were also exempted because the mandatory ASME Code, 1971 Edition through the Winter 1971 Addenda, required brittle fracture testing on ferritic pressure

boundary components only if required in the Design Specification. The Design Specification did not require brittle fracture testing because the system temperature is in excess of 250°F at pressure above 20% of the design pressure. Material information pertaining to the MSIVs is contained in Table 5.2-11.

Main steam piping was tested in accordance with and met the fracture toughness requirements of Paragraph NB-2300 of the 1972 Summer Addenda to ASME Code, Section III.

The ferritic pressure boundary material of the RPV was qualified by impact testing in accordance with the 1971 Edition of Section III ASME Code and Addenda to and including the Summer 1971 Addenda.

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

The degree of compliance with Regulatory Guides 1.31, 1.34, 1.37, 1.43, 1.44, 1.50, 1.66, and 1.71 is addressed in Section 1.8.

5.2.4 INSERVICE INSPECTION AND TESTING OF THE REACTOR COOLANT PRESSURE BOUNDARY

The structural integrity of ASME Code Class 1, 2, and 3 components are maintained as required by the ISI program in accordance with 10 CFR 50.55a. With the structural integrity of any component not conforming to the above requirements, the structural integrity will be restored to within its limits or the affected component will be isolated. For Class 1 components, this isolation will be accomplished prior to increasing reactor coolant system temperature more than 50°F above the minimum temperature required by nil-ductility transition (NDT) considerations. For Class 2 components, isolation will be accomplished prior to increasing reactor coolant system temperature above 200°F.

Inservice Inspections are performed in accordance with the requirements of 10 CFR 50.55a subparagraph (g) as described in the Inservice Inspection Program Plan.

5.2.4.1 System Boundary Subject to Inspection

The system boundary subject to inspection is defined in the Inservice Inspection Program Plan. *The RPV was examined prior to service in accordance with the requirements of the 1974 Edition of the ASME B&PV Code, Section XI, including the Summer 1975 Addenda. All Class 1 piping, pumps, and valves were examined prior to service in accordance with the requirements of the 1974 Edition of the ASME B&PV Code, Section XI, with Addenda through Summer 1975, including Appendix III from the Winter 1975 Addenda.*

The design of the RPV shield wall and external inservice inspection system was completed prior to the promulgation of amendments to 10 CFR 50.55a which require the upgrading of the

utility's inservice inspection code commitment for examinations subsequent to the baseline examination. The design has allowed some additional access for inspections and coverages anticipated to be required by later codes, where possible. The result of this effort has increased the areas on the RPV available to inservice inspection (approximately 84% of the vessel weld volume is accessible) and has allowed the piping examination to be upgraded to conform to the requirements of the Summer 1975 Addenda to Section XI as far as practical.

The preservice examination was performed on Class 1 components and piping pursuant to the requirements of the 1974 Edition of the ASME B&PV Code, Section XI, including the Summer 1975 Addenda for both the RPV and associated piping, pumps, and valves. It is described in the Preservice Inspection Program Plan (Reference 5.2-6).

5.2.4.2 Arrangement of Systems and Components to Provide Accessibility

Access for the purpose of inservice inspection is defined as the design of the plant with the proper clearances for examination personnel and/or equipment to perform inservice examinations. The RCPB for the RPV is designed to provide compliance with the provisions for access as required by Subarticle IWA-1500 of the 1974 Edition of the ASME B&PV Code, Section XI, including the Summer 1975 Addenda. The RCPB for piping, pumps, and valves is designed to provide compliance with the provisions for access as required by Subarticle IWA-1500 of the 1974 Edition of the ASME B&PV Code, Section XI, with addenda through Summer 1975.

Access is provided for volumetric examination of the pressure containing welds from the external surfaces of components and piping by means of removable insulation, removable shielding, and permanent tracks for remote inspection devices in areas where personnel access is restricted. The provisions for suitable access for inservice inspection examinations minimizes the time required for these inspections and, hence, reduces the amount of radiation exposure to both plant and examination personnel. Working platforms are provided at most strategic locations in the plant which permit ready access to those areas of the RCPB which are designated as inspection points in the inservice inspection program. Temporary scaffolding will be used as required to gain access for examination.

Energy Northwest retained Southwest Research Institute to provide an independent assessment as to the suitability of plant access provisions for inservice inspection. This overview provided for identification of design modification or inspection technique development needs to ensure maximum practical compliance with code requirements.

5.2.4.2.1 Reactor Pressure Vessel

Access for inspection of the RPV is as follows:

a. *Access to the exterior surface of the RPV for inservice inspection is provided by removable insulation and shield plugs. Hinged shield wall plugs around nozzles are used to gain access for nozzle inspection. A minimum annular space of 8.25 in. is provided between the vessel exterior surface and the insulation interior surface to permit the insertion of remotely operated inspection devices between the insulation and the reactor vessel. The RPV nozzle insulation is removable. This design allows sufficient clearances for the mounting of a nozzle-to-shell examination device from tracks located either at the nozzle safe-end or at the pipe area. Examinations that can be performed from these tracks include the required coverage of the nozzle-to-shell welds and depending on technique, could provide examination coverage of the nozzle inner radius section and nozzle-to-safe-end weld. Access, geometry and radiation level considerations will determine those nozzles scheduled for manual examination.*

b. *The vessel flange area and vessel closure head can be examined during refueling outages using manual ultrasonic techniques. With the closure head removed, access is afforded to the upper interior clad surface of the vessel by removal of a steam dryer and steam separator assembly. Removal of these components also enables the examination of remaining internal components by remote visual techniques. The volumetric examination of the vessel-to-flange weld and closure head-to-flange weld can be performed by applying the search units directly to the seal surface areas. The vessel-to-flange weld is also examined from vessel shell surface.*

c. *The closure head is dry stored during refueling which facilitates direct manual examination. Removable insulation allows examination of the head welds from the outside surface. Reactor vessel nuts and washers are removed to dry storage for examination during refueling.*

   Selected studs are examined during refueling in accordance with the Inservice Inspection Program Plan.

d. Openings in the RPV support skirt are provided to permit access to the RPV bottom head for purposes of inservice examination. The examinations performed include volumetric examinations of circumferential welds, portions of the meridional welds, portions of the dollar plate longitudinal welds, and visual examination of accessible penetration welds.

5.2.4.2.2 Piping, Pumps, and Valves

The physical arrangement of piping, pumps, and valves is designed to allow personnel access to welds requiring inservice inspection. Modifications to the initial plant design have been incorporated where practicable to provide inspection access on Class 1 piping systems. Removable insulation is provided on those piping systems requiring inspection. In addition, the placement of pipe hangers and supports with respect to those welds requiring inspection have been reviewed and modified where necessary to reduce the amount of plant support required in these areas during inspection. Working platforms are provided to facilitate servicing most of the pumps and valves. Temporary platforms, scaffolding, and ladders will be provided to gain additional access for piping and some pump and valve examinations. An effort has been made to minimize the number of fitting-to-fitting welds within the inspection boundary. Welds requiring inspection are located to permit ultrasonic examinations from at least one side, but where component geometries permit, access from both sides of the weld is provided. The surface of welds within the inspection boundary are prepared to permit effective ultrasonic examination.

5.2.4.3 Examination Techniques and Procedures

Examination techniques and procedures for the preservice examination, including any special technique and procedure, met the requirements of Table IWB-2600 of the 1974 Edition of the ASME B&PV Code, Section XI, including the Summer 1975 Addenda for both the RPV and the associated piping, pump, and valve examinations. Examination techniques and procedures for inservice inspections are in accordance with the Inservice Inspection Program Plan. *During plant design, an effort was made to upgrade the requirement for calibration standards. Where upgrading was not feasible, material of the same P series with similar acoustic characteristics were used.*

5.2.4.3.1 Equipment for Inservice Inspection

Access for inservice inspection of the RPV seam welds is accomplished through openings in the sacrificial shield. These openings are provided at each nozzle location. Permanently installed tracks between the vessel surface and the insulation can be used for mounting remotely operated devices. Access is also provided for devices that do not require use of these tracks.

Remote ultrasonic scanning equipment for examination of the nozzle-to-vessel welds will be supported and guided from tracks temporarily mounted on the pipe connected to the nozzle. The examination equipment will provide radial and circumferential motion to the ultrasonic transducer while rotating about the nozzle. Installation of the equipment will be accomplished through the access openings in the sacrificial shield which are provided at each nozzle location.

5.2.4.3.2 Coordination of Inspection Equipment With Access Provisions

Access to areas of the plant requiring inservice inspection is provided to allow use of standard equipment wherever practicable. Design in general provides for free space envelopes both radially and axially from welds to be examined so standard manual examination equipment may be utilized. Any special equipment or techniques used will achieve the sensitivities required by the codes.

5.2.4.3.3 Manual Examination

In areas where manual ultrasonic examination is performed, all reportable indications are recorded consistent with current inservice inspection codes in effect. Radiographic techniques may be used where ultrasonic techniques are not practical. In areas where manual surface or direct visual examinations are performed, all recordable indications will be in accordance with the Inservice Inspection Program Plan.

5.2.4.4 Inspection Intervals

Inspection intervals are defined in the Inservice Inspection Program Plan.

5.2.4.5 Examination Categories and Requirements

Examination categories and requirements for the preservice inspection are defined in the Preservice Inspection Program Plan and closely follow the categories and requirements specified in Tables IWB-2500 and IWB-2600 of the 1974 Edition with Addenda through Summer 1975 of the ASME B&PV Code, Section XI, for the RPV and the associated piping, pumps, and valves.

Examination categories and requirements for inservice inspections are in accordance with the requirements of ASME Section XI and are contained in the Inservice Inspection Program Plan.

5.2.4.6 Evaluation of Examination Results

Evaluation of results for the RPV, pump, and valve baseline examinations were conducted in accordance with Article IWB-3000 of the 1974 Edition of the ASME B&PV Code, Section XI, including the Summer 1975 Addenda. Evaluation of examination results for piping baseline examinations were conducted in accordance with Article IWB-3000 of the 1974 Edition of the ASME B&PV Code, Section IX, with Addenda through Winter 1975. Energy Northwest recognized that Section XI had been promulgated as an effective code by 10 CFR 50.55a, for the baseline examinations, only through the Summer 1975 Addenda. However, Energy Northwest also recognized that even though the code through Summer 1975 Addenda included evaluation criteria which could be interpreted to apply to piping (Category B-J) welds, the evaluation criteria found in the Winter 1975 Addenda clearly provides evaluation criteria which

are applicable to these welds. Energy Northwest was unaware that the NRC staff was opposed to these evaluation criteria and anticipates that the criteria which will appear in the future codes will be consistent therewith. Evaluations are performed in accordance with the Inservice Inspection Program Plan.

5.2.4.7 System Leakage and Hydrostatic Pressure Tests

The requirement for baseline hydrostatic test for the RPV was satisfied by the hydrostatic test performed in accordance with the requirements of ASME Section III. Similarly, the requirements for the baseline piping system leakage and hydrostatic tests were satisfied by reference to the Section III hydrostatic test report as permitted by ASME Section XI, IWA-5210(b). Subsequent hydrostatic and system leak tests are conducted to the code in effect in accordance with the Inservice Inspection Program Plan.

5.2.4.8 Inservice Inspection Commitment

All quality Group A components were examined once prior to startup in accordance with the above requirements. This preoperational examination served to satisfy the requirements of IWB-2100 of the 1974 Edition of the ASME B&PV Code, Section XI, including the Summer 1975 Addenda for the RPV and associated piping, pumps, and valves. Inservice inspection of Columbia Generating Station is performed in accordance with the Inservice Inspection Program Plan.

5.2.4.9 Augmented Inservice Inspection to Protect Against Postulated Piping Failures

An augmented Inservice Inspection Program Plan has been implemented for Columbia Generating Station, on high energy* Class 1 piping systems which penetrate containment for which the effects of postulated pipe breaks would be unacceptable. This program is described in the Inservice Inspection Program Plan.

* High-energy lines include those systems that, during normal plant conditions, are either in operation or maintained pressurized and where either the maximum operating pressure exceeds 275 psig or maximum operating temperature exceeds 200°F. If, for a particular line, the above pressure and temperature limits are not exceeded more than 2% of the time that the system is in operation, then that line is considered moderate energy and is exempt from the requirement for augmented inservice inspection.

5.2.4.10 Augmented Inservice Inspection of Reactor Pressure Vessel Feedwater Nozzles

5.2.4.10.1 Preservice Examination

Energy Northwest performed a preservice inspection ultrasonic examination of the RFW nozzle inner radii, bore, and safe end regions as described in the Preservice Inspection Program Plan.

In addition, a preservice liquid penetrant examination was performed on the accessible areas of all RFW nozzle inner radius surfaces.

5.2.4.10.2 Inservice Examination

Inservice examinations of RFW nozzles are performed in accordance with the Inservice Inspection Program Plan.

5.2.4.11 Augmented Inservice Inspection for Intergranular Stress Corrosion Cracking

Energy Northwest performed an ultrasonic examination of all Code Class 1 piping which is considered susceptible to IGSCC. The results are reported in the Preservice Inspection Summary Report (References 5.2-9 and 5.2-10).

GL 88-01 weld categories and augmented inspection requirements are described in the Inservice Inspection Program Plan.

5.2.4.12 ASME Section XI Repairs/Replacements

The repair or modification of N-stamped components will be performed in accordance with the Edition and Addenda of ASME Section XI defined in the Inservice Inspection Program Plan and in accordance with ASME Section III (Code Edition and Addenda to which the component was fabricated).

Deviations to the above referenced code edition and addenda as allowed by code will be reviewed by Energy Northwest and authorized on a case-by-case basis.

5.2.5 DETECTION OF LEAKAGE THROUGH REACTOR COOLANT PRESSURE BOUNDARY

5.2.5.1 Leakage Detection Methods

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

a. Main steam lines,
b. RWCU system,
c. RHR system,
d. Reactor core isolation cooling (RCIC) system,
e. Feedwater system,
f. HPCS,
g. LPCS, and
h. Coolant system within the primary containment.

Isolation and/or alarm of affected systems and the detection methods used are summarized in Table 5.2-12.

Small leaks (5 gpm and less) are detected by temperature and pressure changes, drain sump pump activities, floor drain flow monitoring, and fission product monitoring. Large leaks are also detected by changes in reactor water level and changes in flow rates in process lines.

The 5-gpm leakage rate is the limit on unidentified leakage. The leak detection system sensitivity and response is discussed in Section 7.6.2.4.

Compliance with Regulatory Guide 1.45 is described in Section 1.8.

Table 5.2-12 summarizes the actions taken by each leakage detection function. The table shows that those systems which detect gross leakage initiate immediate automatic isolation. The systems which are capable of detecting small leaks initiate an alarm in the control room. The operator can manually isolate the violated system or take other appropriate action.

5.2.5.1.1 Detection of Abnormal Leakage Within the Primary Containment

Leaks within the drywell are detected by monitoring for abnormally high-pressure and temperature within the drywell, high fillup rates of equipment and floor drain sumps, excessive temperature difference between the inlet and outlet cooling water for the drywell coolers, a decrease in the reactor vessel water level, and high levels of fission products in the drywell atmosphere. Temperatures within the drywell are monitored at various elevations. Also the temperature of the inlet and exit air to the atmosphere is monitored. Excessive temperatures in

the drywell, increased drywell drain sump flow rate, and drywell high-pressure are annunciated by alarms in the control room. Drywell high pressure and low reactor vessel water level will cause automatic primary containment isolation. In addition, low reactor vessel water level will isolate the main steam lines. The systems within the drywell share a common area; therefore, their leakage detection systems are common. Each of the leakage detection systems inside the drywell is designed with a capability of detecting leakage rates less than those established by the Technical Specifications.

5.2.5.1.2 Detection of Abnormal Leakage Outside the Primary Containment

Outside the drywell, the piping within each system monitored for leakage is in compartments or rooms, separate from other systems where feasible, so that leakage may be detected by area temperature indications. Each leakage detection system discussed in the following paragraphs is designed to detect leak rates that are less than those established by the Technical Specifications. The method used to monitor for leakage for each RCPB component is described in Table 5.2-12.

a. Ambient and differential room ventilation temperature

A differential temperature sensing system is installed in each room containing equipment that is part of the RCPB. These rooms are the RCIC, RHR, and the RWCU systems equipment rooms and main steam line tunnel. Temperature sensors are placed in the inlet and outlet ventilation ducts or across room boundaries. Other sensors are installed in the equipment areas to monitor ambient temperature. A differential temperature monitor reads each set of sensors and/or ambient temperature and initiates an alarm and isolation when the temperature reaches a preset value. Annunciator and remote readouts from temperature sensors are indicated in the control room.

Spurious isolations of systems due to a relatively sharp drop in outside ambient temperature is highly unlikely. For example, the normal approximate operating differential temperature for the RHR and RCIC pump rooms is 26°F and 32°F respectively. The temperature elements are located at the face of the supply and return ductwork in each pump room. The setpoint differential for isolation is 50°F and 55°F for RCIC and RHR to allow for heat released from a predetermined steam leak. Analysis has shown that it would take a 30°F/hr ambient (outside) temperature decrease for about 2 hr to cause isolation. This magnitude of temperature drop is not supported historically because meteorological data for Hanford has not recorded changes of this magnitude.

b. Reactor building sump flow measurement

Instrumentation monitors and indicates the amount of leakage into the reactor building floor drainage system. The normal leakage collected in the system consists of leakage from the RWCU and CRD systems and from other miscellaneous vents and drains.

c. Visual and audible inspection

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

d. Differential flow measurement (cleanup system only)

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

5.2.5.2 Leak Detection Devices

a. Drywell floor drain sump measurement

The normal design leakage collected in the floor drain sump consists of leakage from the CRDs, valve flange leakage, floor drains, closed cooling water system drywell cooling unit drains, and potential valve stem leaks. The floor drain sump collects unidentified leakage. Design details are given in Section 9.3.3.

b. Drywell equipment drain sump

The equipment drain sump collects only identified leakage. This sump receives condensate drainage from pump seal leakoff and the reactor vessel head flange vent drain. Collection in excess of background leakage would indicate reactor coolant leakage. Design details are given in Section 9.3.3.

c. Drywell air sampling

The primary containment radiation monitoring system is used to supplement the temperature, pressure, and flow variation method described previously to detect

leaks in the nuclear system process barrier. This system is described in Sections 11.5 and 7.6.

Radiation monitors are useful as leak detection devices because of their sensitivity and rapid response to leaks. After several weeks of full power operation, a set level of background radiation is established. Any sudden or unexplained increase in background radiation indicates a possible primary coolant leak within the primary containment. If an increase is noted, a comparison with other leak detection devices having a relationship to each other is made, particularly the equipment and floor drain flow rate monitors, and the reactor building sump pumps activation on high sump level. Using the flow rate monitors as a reference, the comparisons provide independent indications of a leak within the primary containment. This provides diversity in leak detection.

d. Reactor vessel head closure

The reactor vessel head closure is provided with double seals with a leak off connection between seals that is piped to the equipment drain sump. Leakage through the first seal is annunciated in the control room. When pressure between the seals increases, an alarm in the control room is actuated. The second seal then operates to contain the vessel pressure.

e. Reactor water recirculation pump seal

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

f. Safety/relief valves

Tail pipe temperature sensors connected to a multipoint recorder are provided to detect SRV leakage during reactor operation. Safety/relief valve temperature elements are mounted, using a thermowell, in the SRV discharge piping several feet from the valve body. Temperature rise above ambient is recorded in the main control room.

5.2.5.3 Indication in the Control Room

Details of the leakage detection system indications are included in Section 7.6.1.3.

5.2.5.4 Limits for Reactor Coolant Leakage

5.2.5.4.1 Total Leakage Rate

The total leakage rate consists of all leakage, identified and unidentified, that flows to the drywell floor drain and equipment drain sumps. The total leakage rate limit is established so that, in the absence of normal ac power with loss of feedwater supply, make-up capabilities are provided by the RCIC system.

The equipment sump and the floor drain sump collect all leakage. The equipment sump is drained by one 50-gpm pump and the floor drain sump is drained by two 50-gpm pumps. The total leakage rate limit from inside containment is established at 25 gpm, which includes no more than 5 gpm unidentified leakage. The total leakage rate limit is low enough to prevent overflow of the drywell sumps.

5.2.5.4.2 Normally Expected Leakage Rate

The pump packing glands and other seals in systems that are part of the RCPB and from which normal design leakage is expected, are provided with drains or auxiliary sealing systems. Nuclear system pumps inside the drywell are equipped with double seals. Leakage from the primary recirculation pump seals is piped to the equipment drain sump. Leakage in the discharge lines from the main steam SRVs is monitored by temperature sensors that transmit a signal to the control room. Any temperature increase above the drywell ambient temperature detected by these sensors indicates valve leakage.

Thus, the leakage rates from pumps and the reactor vessel head seal are measurable during plant operation. These leakage rates, plus any other leakage rates measured while the drywell is open, are defined as identified leakage rates.

The identified leakage is measured continuously and the leakage rate will be calculated and recorded on a frequency of at least once per 12 hr in accordance with the Technical Specifications. The procedures describing how the identified leakage rate is determined include provisions for showing the identified leakage rate has not exceeded the maximum allowable value of 25 gpm, including no more than 5 gpm unidentified leakage.

Each equipment leak-off connection has been provided with a temperature element which will identify to the operator that a higher than normal temperature exists at that particular location.

5.2.5.5 Unidentified Leakage Inside the Drywell

5.2.5.5.1 Unidentified Leakage Rate

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

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

The unidentified leakage rate limit is established at the 5-gpm rate to allow time for corrective action before the process barrier could be significantly compromised.

The following indications are available to the control room operator for evaluating and detecting unidentified leakage:

Drywell pressure recorders,
Drywell temperature recorders,
Drywell floor drain total flow recorder,
Reactor building floor drain sump fillup rate timer,
Reactor building floor drain sump pump out rate timer,
Drywell cooler cooling water differential temperature recorder,
Reactor vessel water level, and
Drywell atmosphere radiation monitors.

While the indications listed above have no definitive correlation between their engineering units, they provide an early warning of a potential leak to the operator. The actual unidentified leak rate is determined by observing the drywell floor drain system flow rate recorders provided in the control room. Since the monitoring is not computerized, a computer failure would not affect indications.

5.2.5.5.2 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 (References 5.2-4 and 5.2-5). This analysis relates to axially oriented through-wall cracks and provides a realistic estimate of the leak rate to be expected from a crack of critical size. In every case, the leak rate from a crack of critical size is significantly greater than the 5-gpm criterion.

If either the total or unidentified leak rate limits are exceeded, an orderly shutdown would be initiated and the reactor would be placed in a cold shutdown condition in accordance with the Technical Specifications.

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

For process lines that are normally open, there are at least two different methods of detecting abnormal leakage from each system within the nuclear system process barrier located in the drywell, reactor building, and auxiliary building as shown in Table 5.2-12. 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. Some alarm points require hot operation data for their determination. Preoperational testing verified proper operation of the instrumentation for the alarm point used.

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

The leak detection system sensitivity and response time is discussed in Section 7.6.2.4 such that an unidentified leakage rate increase of 1 gpm in less than 1 hr will be detected.

5.2.5.6 Safety Interfaces

The balance of plant/GE nuclear steam supply system safety interfaces for the leak detection system are the signals from the monitored balance-of-plant equipment and systems that are part of the nuclear system process barrier and associated wiring and cable lying outside the nuclear steam supply equipment. These balance-of-plant systems and equipment include the main steam line tunnel, the SRVs, and the turbine building sumps.

5.2.5.7 Testing and Calibration

Provisions for testing and calibration of the leak detection system are described in Section 7.6.

5.2.6 REFERENCES

5.2-1 “Qualification of the One–Dimensional Core Transient Model (ODYN) for BWR’s,” NEDO-24154–A, Vol. 1 and 2, General Electric, August 1986.

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

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

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

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

5.2-6 Washington Public Power Supply System, 1985, "WNP-2 Preservice Inspection Program Plan," Washington Public Power Supply System, Richland, Washington.

5.2-7 Deleted.

5.2-8 Deleted.

5.2-9 Letter GO2-85-110 from G. C. Sorenson, Supply System, to A. Schwencer, NRC, Subject: Nuclear Project No. 2, CPPR-93 Preservice Inspection Program Plan, Amendment No. 4, Summary Report Supplement No. 1, NIS-1 Code Data Report, dated February 28, 1985.

5.2-10 Letter GO2-83-401 from G. D. Bouchay, Supply System, to A. Schwencer, NRC, Subject: Nuclear Project No. 2, CPPR-93, Preservice Inspection Program Plan, Volume No. 4, "Preservice Inspection Summary Report", dated May 3, 1983.

5.2-11 "Supplemental Reload Licensing Report for Columbia" (most recent version referenced in COLR).

Table 5.2-1

Exceptions to Conformance to 10 CFR 50.55a
Reactor Coolant Pressure Boundary Components

Component Description	Quantity	Plant Identification System Number	Purchase Order Date	Code Applied ASME Section III	Code Required by 10 CFR 50.55(a)	Component Status
Main steam safety relief valves	18	MS-RV-1 A-D MS-RV-2 A-D MS-RV-3 A-D MS-RV-4 A-D MS-RV-5 B-C (B22-F013 A-V)	April 1971	1971 Edition	1971 Summer Addenda	FS[a]
Recirc pumps	2	RRC-P-1A (B35-C001)	April 1971	1971 Edition	1971 Summer Addenda	FS
Recirc gate valves	4	RRC-V-23/ RRC-V-67 (B35-F023/F067)	June 1971	1971 Edition	1971 Summer Addenda	FS
Recirc flow control valve	2	RRC-V-60 (B35-F060)	June 1971	1971 Edition	1971 Summer Addenda	FS
Recirc piping	1 lot	B35-G001	October 1971	1971 Summer Addenda	1971 Winter Addenda	FS

[a] FS = Fabricated and Shipped

Table 5.2-2

Reactor Coolant Pressure Boundary
Component Code Case Interpretations

	Number	Title	Remarks
1.	1332 - Revision 6	Requirements for steel forgings	Regulatory Guide 1.85, Revision 6
2.	1401 - Revision 0	Welding repairs to cladding of Class I, Section III, components after heat treating	
3.	1420 - Revision 0	5b-167 Ni-Cr-Fe alloy pipe or tube	
4.	1441 - Revision 1	Waiving of 3 S_m requirement for Section III construction	
5.	1141 - Revision 1	Foreign produced steel	Regulatory Guide 1.85, Revision 5
6.	1361 - Revision 2	Socket welds, Section III	Regulatory Guide 1.84, Revision 9
7.	1525	Pipe descaled by means other than pickling, Section III	
8.	1535 - Revision 2	Hydrostatic test of Class 1 nuclear valves, Section III	Regulatory Guide 1.84, Revision 9
9.	1567	Testing lots of carbon and low alloy steel covered electrodes, Section III	Regulatory Guide 1.85, Revision 6
10.	1621 - Revision 1	Internal and external valve items, Section III, Class 1	Regulatory Guide 1.84, Revision 12 (for 1621-2)
11.	1588	Electro-etching of Section III code symbols	Regulatory Guide 1.84, Revision 9
12.	1820	Alternative ultrasonic examination technique Section III, Division 1	Regulatory Guide 1.85, Revision 11
13.	N181	Steel castings refined by the argon oxygen decarbonization process Section 3, Division 1 construction	
14.	1711	Pressure relief valve, design rules, Section III, Division 1, Class 1, 2, 3	

Table 5.2-3

Nuclear System Safety/Relief Setpoints

Number of Valves	Spring Set Pressure (psig)	ASME Rate Capacity at 103% Spring Set Pressure (lb/hr each)	Pressure Setpoint for the Power Actuated Mode (psig)
2	1165	876,500	1091
4	1175	883,950	1101
4	1185	891,380	1111
4	1195	898,800	1121
4	1205	906,250	1131

Note: Seven of the safety/relief valves serve in the automatic depressurization function.

Table 5.2-4

Systems Which May Initiate During
Safety Valve Capacity Overpressure Event

System	Initiating/Trip Signal(s)[a]
Reactor Protection System	Reactor trips "OFF" on high flux
RCIC	"ON" when reactor water level ≤L2
	"OFF" when reactor water level ≥L8
HPCS	"ON" when reactor water level ≤L2
	"OFF" when reactor water level ≥L8
Recirculation system	"OFF" when reactor water level ≤L2
	"OFF" when reactor pressure ≤1143 psig
RWCU	"OFF" when reactor water level ≤L2

[a] Note: Vessel level trip settings L2 and L8 shown in Figure 5.3-3.

Table 5.2-5

Sequence of Events for Figure 5.2-2

Time-Sec	Event
0	Initiate closure of all main steam isolation valves (MSIV).
0.45	MSIVs reached 85% open and initiated reactor scram. However, hypothetical failure of this position scram was assumed in this analysis.
1.7	Neutron flux reached the high APRM flux scram setpoint and initiate reactor scram.
2.9	Steam line pressure reached the group safety relief valve pressure setpoint (spring-action mode and safety relief valves started to open).
3.0	MSIVs completely closed.
3.5	All safety relief valves opened.
4.5	Vessel bottom pressure reached its peak value.

Table 5.2-6

Design Temperature, Pressure and Maximum Test Pressure for RCPB Components

Component	Design Temperature (°F)	Design Pressure (psig)	Maximum Test Pressure (psig)
<u>Reactor vessel</u>	575	1250	1563
<u>Recirculation system</u>			
Pump discharge piping, through valves	575	1650	(a)
Pump discharge piping, beyond valves	575	1550	(a)
Pump suction piping	575	1250	(a)
Pump and discharge valves	575	1650	(b)
Suction valves	575	1250	(b)
Flow control valve	575	1675	(a)
Vessel drain line	575	1275	(a)
<u>Main steam system</u>			
Main steam line	575	1250	(a)
Main steam line valves	575	1250	(b)
<u>Residual heat removal system</u>			
Shutdown suction			
Recirculation header to second isolation valve			
Piping	575	1250	(a)
Valves	575	1250	(b)
Pump discharge			
Reactor vessel to second isolation valve			
Piping	575	1250	(a)
Valves	575	1250	(b)

Table 5.2-6

Design Temperature, Pressure and Maximum Test Pressure for RCPB Components (Continued)

Component	Design Temperature (°F)	Design Pressure (psig)	Maximum Test Pressure (psig)
Shutdown return			
Recirculation header to second isolation valve			
Piping	575	1575	(a)
Valves	575	1575	(b)
Reactor feedwater			
Reactor vessel to manual valve (F011)			
Piping	575	1300	(a)
Valves	575	1300	(b)
Reactor core isolation cooling system			
Steam to RCIC.	575	1250	(a)
Pump turbine			
Reactor vessel to second isolation valve			
Piping	575	1250	(a)
Valves	575	1250	(b)
Pump discharge to reactor	170	1500	(a)
Reactor vessel to second isolation valve			
Piping	575	1500	(a)
Valves	575	1500	(b)
High-pressure core spray system			
Outboard containment isolation valve to and including maintenance valve inside containment[c]			
Piping	575	1250	(a)
Valves	575	1250	(b)

Table 5.2-6

Design Temperature, Pressure and Maximum Test Pressure for RCPB Components (Continued)

Component	Design Temperature (°F)	Design Pressure (psig)	Maximum Test Pressure (psig)
From maintenance valve to reactor vessel			
Piping	575	1250	(a)
Valves	575	1250	(b)
<u>Low-pressure core spray system</u>			
Outboard isolation valve to reactor vessel			
Piping	575	1250	(a)
Valves	575	1250	(b)
<u>Standby liquid control</u>			
Pump discharge to reactor vessel			
Reactor to second isolation valve[d]			
Piping	150	1400	(a)
Valves	150	1400	(b)
<u>Reactor water cleanup system</u>			
Pump suction			
Recirculation piping to isolation valve outside drywell			
Piping	575	1250	(a)
Valves	575	1250	(b)
<u>Control rod drive system</u>			
Piping to HCUs	150	1750	2187

[a] Test pressure at the bottom of the reactor vessel is nominally 1565. The piping is field tested with the reactor vessel.

[b] Test pressure is based on ASME III Table NB-3531-9 (1971 Edition through Winter 1973 Addenda).

[c] For dual design conditions, see Figure 6.3-3.1.

[d] The design temperature and pressure of the original injection piping were 575°F and 1250 psig. This portion of piping was rerouted to the HPCS injection and was tested in accordance with ASME Section XI, 1980 Edition, Winter Addenda.

Table 5.2-7

Reactor Coolant Pressure Boundary Materials

Component	Form	Material	Specification (ASTM/ASME)
Reactor vessel	Rolled plate	Low alloy steel	SA-533 grade B class 1
Heads, shells	Forgings Welds	 Low alloy steel	SA-508 class 2 SFA-5.5
Closure flange	Forged ring Welds	Low alloy steel Low alloy steel	SA-508 class 2 SFA-5.5
Nozzle safe ends	Forgings or Plates	Stainless steel	SA-182, F304 or F316 SA-336, F8 or F8 M SA-240, 304 or 316
	Welds	Stainless steel	SFA-519, TP-308L or 316L
Nozzle safe ends	Forgings Welds	Ni-Cr-Fe Ni-Cr-Fe	SB-166 or SB-167 SFA-5.14, TP ERNiCr-3 or SFA-5.11, TP ENCrFe-3
Nozzle safe ends	Forgings	Carbon steel	SA-105 grade 2, SFA-5.18 grade A, or SFA-5.17 F70
Nozzle safe ends	Forgings	Austenitic stainless steel	SA-182 grade F, 316L
Cladding	Weld overlay	Austenitic stainless steel	SFA-5.9 or SFA-5.4 TP-309 with carbon content on final surface limit to 0.09% maximum
Control rod drive housings	Pipe Forgings Welds	Austenitic stainless steel Inconel	SA-312 type 304 SFA-5.11 type ENiCrFe-3 or SFA-5.14 type ERNiCr-3

Table 5.2-7

Reactor Coolant Pressure
Boundary Materials (Continued)

Component	Form	Material	Specification (ASTM/ASME)
In-core housings	Pipe Forgings Welds	Austenitic stainless steel Inconel	SA-312 type 304 SFA-5.11 type ENiCrFe-3 or SFA-5.14 type ERNiCr-3

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

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

Pipe SA-106 grade B and C; SA-333 grade 5; SA-155 grade KCF 70

Valves SA-105 grade II-normalized; SA-350 grade LF1 or LF2 and SA-216 grade WCB, normalized; and SA-352 grade LCB

Fittings GA-105 grade II-normalized; SA-350 grade LF1 or LF2-normalized; SA-234 grade WPB-normalized; and SA-420 grade WPL1

Bolting SA-193 grade B7; and SA-194 grades 7 and 2H

Welding Material Welding materials conform to the applicable SFA specifications listed in ASME B&PV Code Section IIc. Individual selection of filter metals are reviewed for conformity to the base materials being welded by the Consulting Engineers' review of welding procedures.

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

Pipe SA-376 type 304; SA-312 type 304; SA-358 type 304

Valves SA-182 grade F-304 and F-316; SA-351 grades CF-3, CF-3M, CF-8 and CF-8M

Table 5.2-7

Reactor Coolant Pressure
Boundary Materials (Continued)

Pump	SA-182 grade F-304; SA-351 grades CF-8 and CF-8M
Flanges	SA-182 grade F-316
Bolting	SA-193 grade B7; SA-194 grades 7 and 2H
Welding	SFA-5.4 (E308-15, E308L-15, E316-15); SFA-5.9 (ER-308, ER-308L, ER-316)
Fittings	SA-182 grade F304; SA-351 grade CF-8; SA-403 grade WP-304, 304W

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Table 5.2-8

Water Sample Locations

Sample Origin	Sensor Location	Indicator Location	Recorder Location	Range µmho/cm	Alarm Low	Alarm High
Reactor water recirculation loop	Sample line	Sample station	Control room	0-1	0.0	1.0
Reactor water cleanup system inlet	Sample line	Sample station	Control room	0-1	0.0	1.0
Reactor water cleanup system outlets	Sample line	Sample station	Control room	0-0.3	NA	0.15

Table 5.2-9

IHSI Summary
Prior to First Refueling GL 88-01,
Category B Welds

Energy Northwest ISI Weld Number	Welds
Stainless steel to stainless steel	
24RRC(2)-A-2 thru 24RRC(2)-A-12	11
24RRC(1)-A-13 thru 24RRC(1)-A-22	10
16RRC(1)-A-1 thru 16RRC(1)-A-4	4
12RRC(1)-N2A-1, 1A	2
12RRC(1)-N2B-1, 1A	2
12RRC(1)-N2C-1, 1A	2
12RRC(1)-N2D-1, 1A	2
12RRC(1)-N2E-1, 1A	2
20RRC(6)-1 thru 20RRC(6)-7, 7A, 8	9
4RRC(8)-2A-1, 2	2
4RRC(8)-1A-1, 2	2
12RRC(7)-A-1 thru 12RRC(7)-A-6	6
12RHR(1)-A15 thru 12RHR(1)-A18	4
24RRC(2)-B-2 thru 24RRC(2)-B-10	9
16RRC(1)-B-1 thru 16RRC(1)-B-4	4
24RRC(1)-B-11 thru 24RRC(1)-B-20	10
12RRC(1)-N2F-1, 1A	2
12RRC(1)-N2G-1, 1A	2
12RRC(1)-N2H-1, 1A	2
12RRC(1)-N2J-1, 1A	2
12RRC(1)-N2K-1, 1A	2
4RRC(8)-2B-1, 2	2
4RRC(8)-1B-1, 2	2
12RRC(7)-B-1, 2, 2A thru 12RRC(7)-B-6	7
12RHR(1)-B-11 thru 12RHR(1)-B-13	3
20RHR(2)-1	1
Stainless steel to stainless steel caps	
24RRC(1)-A13/8CAP-1, A20/12CAP-1	2
24RRC(1)-B-11/CAP-1, 18/12CAP-1	2
Stainless steel to carbon steel	
20RHR(2)-2	1
12RHR(1)-A14	1
12RHR(1)-B-10	1
TOTAL	113

Table 5.2-10

IHSI Summary
During First Refueling GL 88-01,
Category B Welds

Energy Northwest ISI Weld Number		Welds
4RRC(4) A-1 thru 4RRC(4) A-11		11
4RRC(4) B-1 thru 4RRC(4) B-12		12
24RRC(2) A-10/4RRC(8)-4S		1
24RRC(2) A-10/4RRC(4)-4S		1
24RRC(1) A-13/4RRC(8)-4S		1
24RRC(1) A-13/8 Cap		1
24RRC(1) A-20/12 Cap		1
24RRC(1) A-20/12RRC(7)-4S		1
24RRC(2) B-8/4RRC(8)-4S		1
24RRC(2) B-8/4RRC(4)-4S		1
24RRC(1) B-11/8 Cap		1
24RRC(1) B-11/4RRC(8)-4S		1
24RRC(1) B-18/12 Cap		1
24RRC(1) B-18/12RRC(7)-4S		1
	TOTAL	35
Type 304 Welds with Low Carbon Content		
[a]4JP (NZ) A-1 Inconel 182 buttering		1
[a]4JP (NZ) B-1 Inconel 182 buttering		1
[a]4JP (NZ) A-2		1
[a]4JP (NZ) B-2		1
	TOTAL	4

[a] Confirmed by CMTR review safe end material used was type 304 with a carbon content of ≤ 0.025%.

Table 5.2-11

Main Steam Isolation Valves
Material Information

Item	Material Spec	Material Type	Minimum Design Wall Thickness
Body	SA-216	GR WCB	1.58 in.
Bonnet	SA-105	GR II	7.66 in.
Stem disc[a]	SA-105	N/A	1.56 in.
Disc piston[a]	SA-105	N/A	3.24 in.
Stem[a]	SA-564 or A-182	Tp 630 H1100 GR F6A C1 3	
Bonnet studs	SA-540	Class 4	1-5/8 in. diameter
Bonnet nuts	SA-194	GR 7	1-5/8 in. diameter

See Section 5.2.3.3 for fracture toughness response.

Piping connecting the MSIV
Outside diameter 12 in.
Nominal wall thickness = 1.103 in. plus 0.125 in.

[a] Redesign/replacement materials

Table 5.2-12

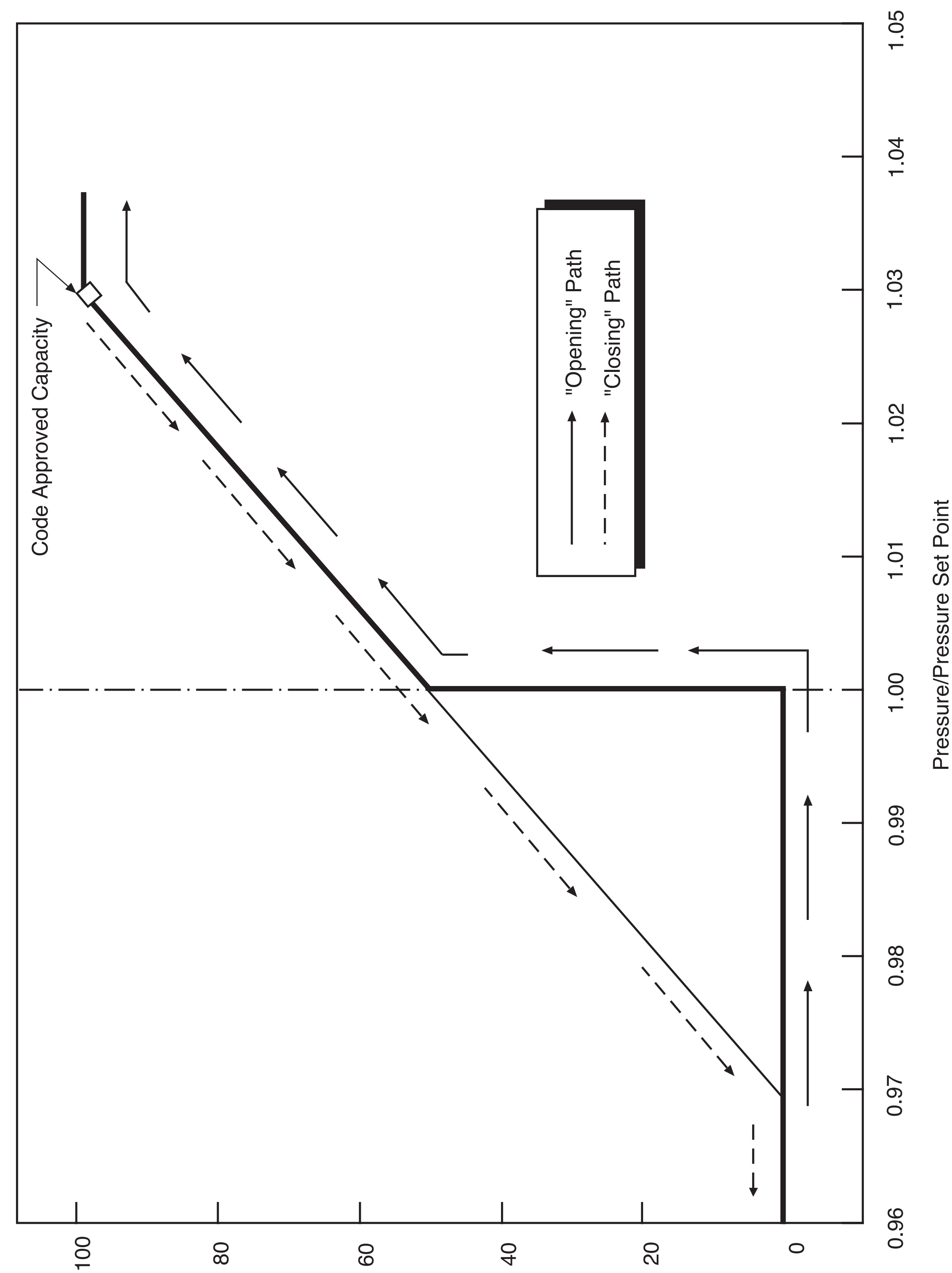
Summary of Isolation/Alarm of System Monitored and the Leak Detection Methods Used

		Variable Monitored												
FUNCTION		A	A	A	A/I	A/I	A	A/I	A/I	A/I	A/I	A/I	A	A
Source of Leakage	Location	High PC °F	PC Sump High Flow Rate	High/Dry-well Cooler Condensate Flow[a]	Equip-ment Area High T and ΔT	Low Steam Line Pressure	RB Sump or Drain High Flow Rate	PC Pressure (High)	High Flow Rate[b]	RCIC Diaphragm High Exhaust Line Pressure	RWCU ΔFlow (High)	Reactor Low Water Level	High Differential Pressure	Fission Products High[a]
Main steam line	PC	X	X	X		X[c]		X	X			X		X
	RB				X	X[c]	X		X			X		
RHR	PC	X	X	X				X				X		X
	RB				X		X		X			X		
RCIC steam	PC	X	X	X		X		X	X					X
	RB				X		X		X	X				
RCIC water	PC		X											
	RB						X							
RWCU water	PC	X	X	X				X[b]			X	X		X
	RB hot				X		X		X		X	X		
	RB cold						X		X		X	X		
Feedwater	PC	X	X	X				X						
	RB				X[d]		X							
ECCS water	PC		X										X	X
	RB						X						X	
Reactor coolant	PC	X	X	X				X				X		X
	RB													

PC - Primary containment
RB - Reactor building
RWCU - Reactor water cleanup
CCW - Closed cooling water

A - Alarm
I - Isolation

NOTE:
[a] All systems within the drywell share a common detection system.
[b] Break downstream of flow element will isolate the system.
[c] In run mode only.
[d] Alarm only (steam tunnel).



Columbia Generating Station
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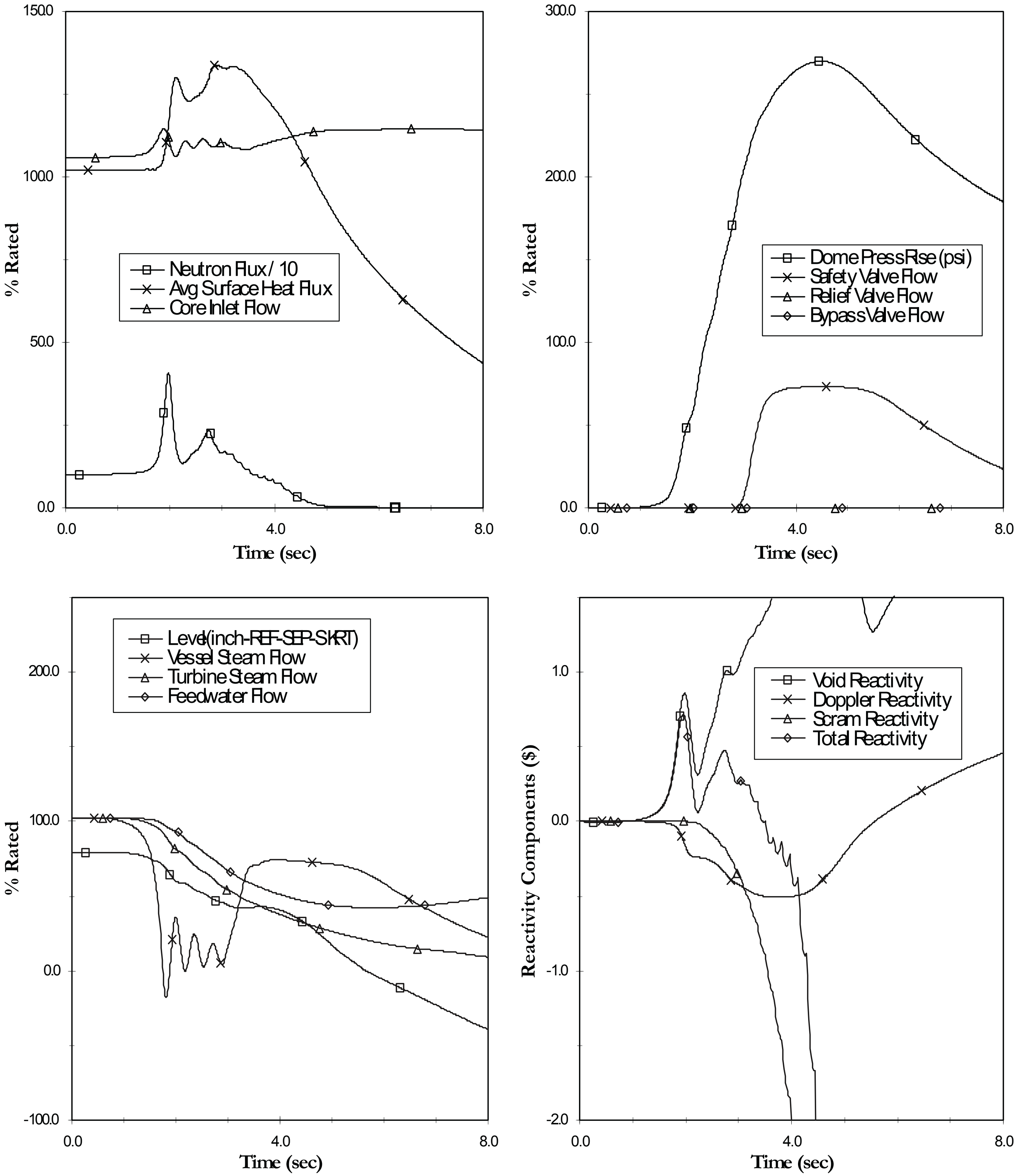
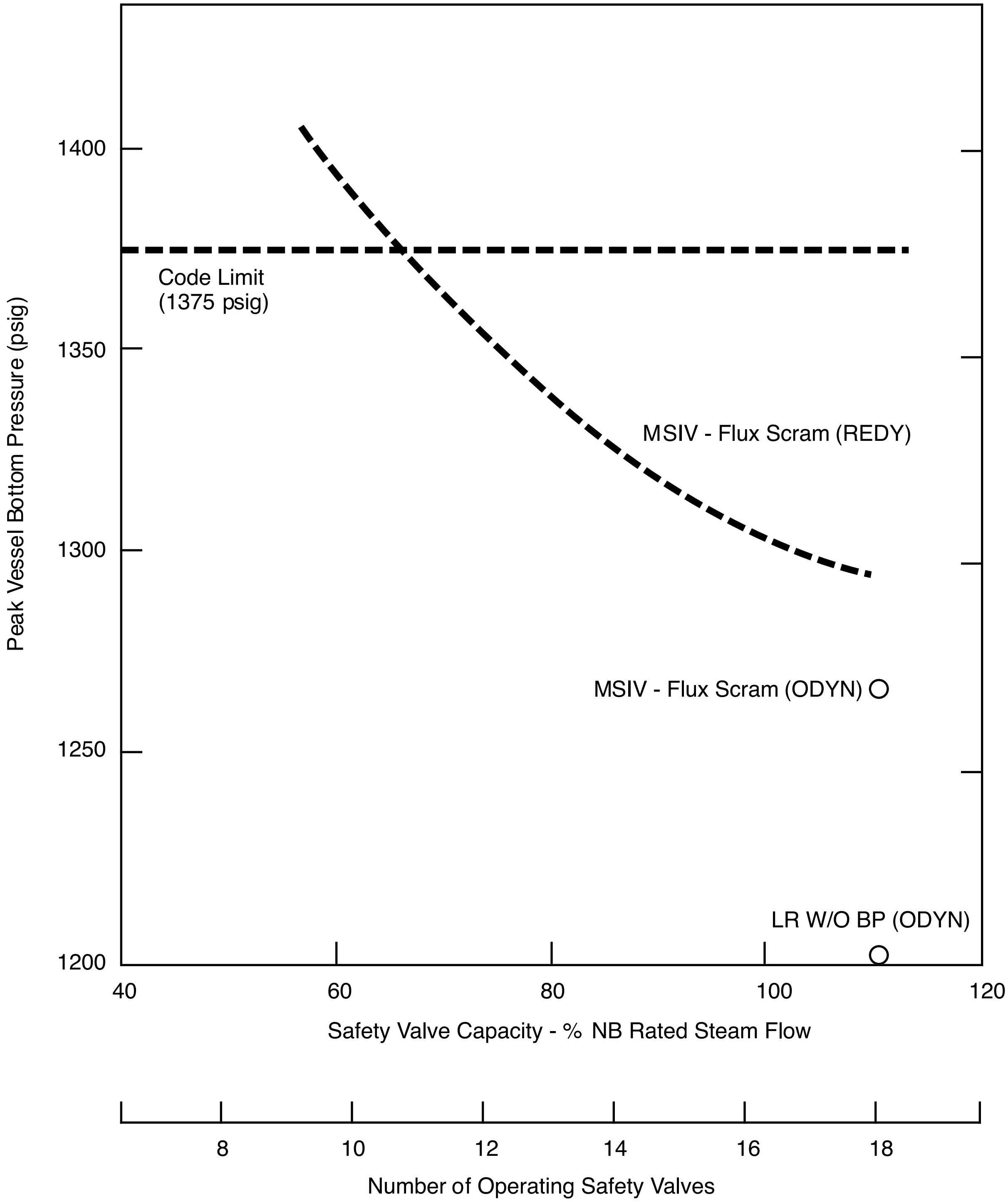
Simulated Safety Relief Valve Spring Mode
Characteristic used for Capacity Sizing Analysis

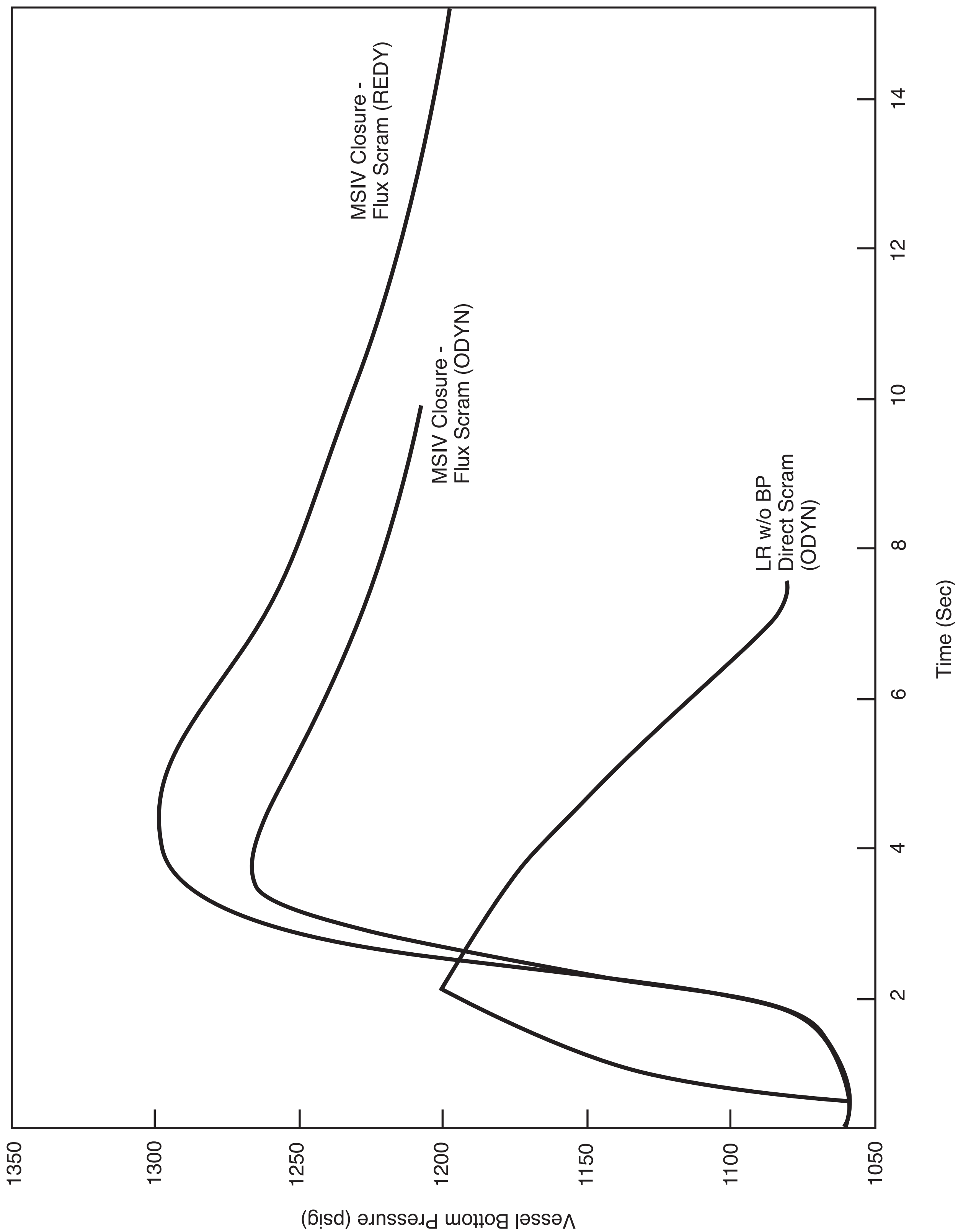
Draw. No. 960690.46 | Rev. | Figure 5.2-1



Columbia Generating Station Final Safety Analysis Report	MSIV Closure with Flux Scram - Nominal Safety Setpoint +3% 6 SRV Out-of-Service		
	Draw. No. 950021.16	Rev.	Figure 5.2-2



Peak Vessel Pressure Versus Safety Valve Capacity



Columbia Generating Station
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Time Response of Pressure Vessel for Pressurization Events

Draw. No. **960690.48**	Rev.	Figure **5.2-4**

MPL NO. B22-1040

TABLE VI: ELEVATION CORRELATION CHART

The right-hand columns are the control room water level indication and trip levels (see Note 3). The Fuel Zone, Wide Range and Narrow Range columns fall under Safeguards, and the last Narrow Range column falls under Feedwater.

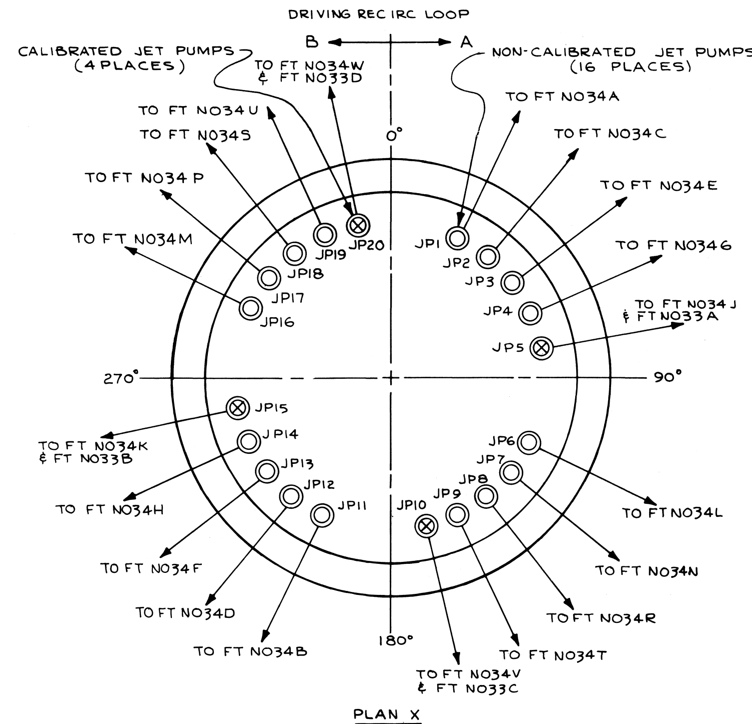
REFERENCE	(COLD VESSEL) INCHES ABOVE VESSEL ZERO	DESCRIPTION OF TRIPS	INSTRUMENT(S) PROVIDING TRIP	REACTOR VESSEL LEVEL IDENTITY SEE REF. 25 OF P&ID	FUEL ZONE LR-R615 LI-R610	WIDE RANGE LR/PR-R623A,B LI-R604	NARROW RANGE C34 LR-R608	NARROW RANGE C34 LI-R606A,B,C	UPSET C34-LR-R608	SHUTDOWN LI-R605
TOP OF HEAD FLANGE	898 MAX									+400"
									+180"	
STEAM LINE NOZZLE N3	648									
INSTRUMENT LINE NOZZLE N14	599					+60"	+60"	+60"		
							+54.5"	+54.5"		
		TRIP RCIC & CLOSE HPCS INJECTION VALVE CLOSE MAIN TURBINE STOP VALVES. TRIP FEED PUMPS.	TABLE IV REF. 2	8						
								+40.5"		
		HIGH LEVEL ALARM	REF. 2	7						
		NORMAL WATER LEVEL	REF. 2	5,6						
		LOW LEVEL ALARM RUN RECIRC FLOW BACK *	REF. 2	4				+31.5"		
		SCRAM & CLOSE RHR SHUTDOWN COOLING ISOLATION VALVES. CONTRIBUTE TO AUTO DEPRESSURIZATION. RUN BACK RECIRC FLOW	TABLE IV REF. 2	3			+13"	+13"		
WATER LEVEL INSTRUMENT ZERO	527.5				0"	0"	0"	0"	0"	0"
BOTTOM OF DRYER SKIRT										
INSTRUMENT LINE NOZZLE N13	517									
		INITIATE RCIC & HPCS. CLOSE PRIMARY SYSTEMS ISOLATION VALVES EXCEPT RHR SHUTDOWN COOLING. MSIVs, MAIN STEAM LINE DRAIN VALVES, TRIP RECIRC PUMPS	TABLE IV	2		-50"				
					-110"					
		INITIATE RHR & CORE SPRAY SYS CONTRIBUTE TO AUTO DEPRESSURIZATION. START STAND-BY DIESEL. CLOSE MSIVs, MAIN STEAM LINE DRAIN VALVES.	TABLE IV	1		-129"				
						-150"				
INSTRUMENT LINE NOZZLE N12	366									
TOP OF ACTIVE FUEL	366.3									
JET PUMP INSTRUMENT NOZZLE N9	152.0				-310"					
JET PUMP DIFFUSER TAP	143.5									

* FUNCTION IS IN FEEDWATER CONTROL SYS (REF 2) FOR LOSS OF ONE FEED PUMP

TABLE I: SAFETY/RELIEF VALVE LOCATION, SUFFIX ASSIGNMENT, ASSOCIATED EQUIPMENT

SAFETY / RELIEF VALVE	F013	A	B	C	D	E	F	G	H	J	K	L	M *	N ⊛	P	R *	S *	U ⊛	V ⊛
ACCUMULATORS	A003 (ADS)												M	N	P	R	S	U	V
	A004	A	B	C	D	E	F	G	H	J	K	L	M	N	P	R	S	U	V
CHECK VALVES	F036	A	B	C	D	E	F	G	H	J	K	L	M	N	P	R	S	U	V
	F040(ADS)												M	N	P	R	S	U	V
VACUUM BREAKER	F037	A	B	C	D	E	F	G	H	J	K	L	M	N	P	R	S	U	V
PRESSURE SWITCH	N039	A	B	C	D	E	F	G	H	J	K	L	M	N	P	R	S	U	V
COMPUTER INPUTS FOR VALVE POS(REF 3)		C1732	C1733	C1734	C1735	C1736	C1737	C1738	C1739	C1740	C1741	C1742	C1743	C1744	C1745	C1746	C1747	C1748	C1749
TEMPERATURE ELEMENT	N004	A	B	C	D	E	F	G	H	J	K	L	M	N	P	R	S	U	V
ADS FUNCTION													YES	YES	YES	YES	YES	YES	YES

ADS - SAFETY/RELIEF VALVE FOR AUTO DEPRESSURIZATION
* - CONTROL PROVIDED IN REMOTE SHUTDOWN SYS (REF 5)
⊛ - CONTROL PROVIDED IN ALTERNATE REMOTE SHUTDOWN SYS (SEE REF 5)
⊛⊛ NS4 EXCEPT MSIVs, MAIN STEAM LINE DRAIN VALVES.
⊛⊛⊛ MSIVs, MAIN STEAM LINE DRAIN VALVES.



PLAN X

NOTES:

1. REF. DOC. 1 SHALL BE USED WITH AND FORM PART OF THIS P&ID DATA, IF THERE ARE ANY CONFLICTS BETWEEN THE P&ID AND THIS P&ID DATA, THE P&ID DATA SHALL GOVERN.
2. MPL NO'S ARE PREFIXED BY B22 UNLESS OTHERWISE NOTED.
3. WATER LEVEL INSTRUMENTS FOR VARIOUS RANGES ARE CALIBRATED AS STATED BELOW. ALL WATER LEVEL SWITCH SET POINTS ARE NOMINAL: IE; THE ANALYSES ARE PERFORMED WITH THE SWITCH TRIP UNCERTAINTY INCLUDED. REACTOR BUILDING TEMPERATURE ASSUMED TO BE 75°F.
   - A. FUEL ZONE: THE INSTRUMENTS ARE CALIBRATED FOR SATURATED WATER STEAM CONDITIONS AT 0 PSIG IN THE VESSEL AND THE DRYWELL WITH NO JET PUMP FLOW. WATER LEVEL SWITCH TRIP UNCERTAINTY IS ± 6" OF WATER LEVEL AT CALIBRATION CONDITIONS.
   - B. WIDE RANGE: THE INSTRUMENTS ARE CALIBRATED FOR 1000 PSIG IN THE VESSEL, 135°F IN THE DRYWELL AND 20 BTU/LB SUBCOOLING BELOW THE MIDDLE WATER LEVEL NOZZLE AND SATURATED CONDITIONS ABOVE THE MIDDLE WATER LEVEL NOZZLE WITH NO JET PUMP FLOW. WATER LEVEL SWITCH TRIP UNCERTAINTY IS ± 6" OF WATER LEVEL AT CALIBRATION CONDITIONS.
   - C. NARROW RANGE: (SAFEGUARDS AND FEEDWATER): THE INSTRUMENTS ARE CALIBRATED FOR SATURATED WATER STEAM CONDITIONS AT 1000 PSIG IN THE VESSEL AND 135°F IN THE DRYWELL. WATER LEVEL SWITCH TRIP UNCERTAINTY IS ± 1.5" OF WATER LEVEL AT CALIBRATION CONDITIONS.
   - D. UPSET RANGE: THE INSTRUMENT IS CALIBRATED FOR SATURATED WATER STEAM CONDITIONS AT 1000 PSIG IN THE VESSEL AND 135°F IN THE DRYWELL.
   - E. SHUTDOWN: THE INSTRUMENT IS CALIBRATED FOR 120°F WATER AT 0 PSIG IN VESSEL AND 80°F IN THE DRYWELL.

761E445

REFERENCE DOCUMENTS:

1. NUCLEAR BOILER SYSTEM P&ID ---------- B22-1010
2. FEEDWATER CONTROL SYSTEM IED ---------- C34-1010
3. NUCLEAR BOILER SYSTEM FCD ---------- B22-1030
4. ELEC EQ SEP SAFEGUARD SYS ---------- A62-4050
5. REMOTE SHUTDOWN SYS IED ---------- C61-1010

TABLE II: JET PUMP INSTRUMENTATION SUFFIX LETTER ASSIGNMENT

LOOP	A										B									
JP NO.	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	20
FT N034 E/SK615	A	C	E	G	J	L	N	R	T	V	B	D	F	H	K	M	P	S	U	W
dPI R608	A	C	E	G	J	L	N	R	T	V	B	D	F	H	K	M	P	S	U	W
√ K600	A	C	E	G	J	L	N	R	T	V	B	D	F	H	K	M	P	S	U	W
Σ	K601A				K601C			K607A			K601B				K601D			K607B		
Σ	K602A										K602B									
⚠	A1751	A1752	A1753	A1754	A1755	A1756	A1757	A1758	A1759	A1760	A1761	A1762	A1763	A1764	A1765	A1766	A1767	A1768	A1769	A1770
FT N033	—	—	—	—	A	—	—	—	—	C	—	—	—	—	B	—	—	—	—	D
√ K605	—	—	—	—	A	—	—	—	—	C	—	—	—	—	B	—	—	—	—	D
FI R609	—	—	—	—	A	—	—	—	—	C	—	—	—	—	B	—	—	—	—	D

THE SUPPLY SYSTEM IS RESPONSIBLE FOR DESIGN CHANGE CONTROL OF THIS DOCUMENT

TABLE IV: WATER LEVEL INSTRUMENT CONTACT UTILIZATION

INSTRUMENT NUMBER	DIV	UPPER RANGE TRIP 2-A	UPPER RANGE TRIP 2-B	LEVEL #	LOWER RANGE TRIP 1-A	LOWER RANGE TRIP 1-B	LEVEL #	TRANS E/S MPL #
LIS-N024A	IA				RPS (NS4)		3	—
LIS-N024B	IB	RCIC (I)			RPS (NS4)		3	—
LIS-N024C	IIA				RPS (NS4)		3	—
LIS-N024D	IIB	RCIC (II)			RPS (NS4)		3	—
LIS-N100A	III	HPCS		8				—
LIS-N100B	III	HPCS		8				—
LIS-N038A	I					ADS (A)	3	—
LIS-N038B	II					ADS (B)	3	—
LIS-N037A	I	RCIC		2	LPCS & RHR(A)	ADS (A)	1	—
LIS-N037B	II	RCIC		2	RHR(B) & RHR(C)	ADS (B)	1	—
LIS-N037C	I	RCIC		2	LPCS & RHR(A)	ADS (A)	1	—
LIS-N037D	II	RCIC		2	RHR(B) & RHR(C)	ADS (B)	1	—
LIS-N031A	III	HPCS		2				—
LIS-N031B	III	HPCS		2				—
LIS-N031C	III	HPCS		2				—
LIS-N031D	III	HPCS		2				—
MS-LS-300A	IA	⊛⊛		2				C72A-K613A
MS-LS-300B	IB	⊛⊛		2				C72A-K613B
MS-LS-300C	IIA	⊛⊛		2				C72A-K613C
MS-LS-300D	IIB	⊛⊛		2				C72A-K613D
LIS-N036A		RECIRC (A)		2				—
LIS-N036B		RECIRC (B)		2				—
LIS-N036C		RECIRC (A)		2				—
LIS-N036D		RECIRC (B)		2				E12 K603A
MS-LIS-200A	IA	⊛⊛⊛		1				C72A-K613A
MS-LIS-200B	IB	⊛⊛⊛		1				C72A-K613B
MS-LIS-200C	IIA	⊛⊛⊛		1				C72A-K613C
MS-LIS-200D	IIB	⊛⊛⊛		1				C72A-K613D

TABLE V: PRESSURE INSTRUMENT CONTACT UTILIZATION

INSTRUMENT NUMBER	DIV	TRIP	TRANS E/S MPL #
PS N020A	IA	RPS	—
PS N020B	IB	RPS	—
PS N020C	IIA	RPS	—
PS N020D	IIB	RPS	—
PS N023A	IA	RPS	—
PS N023B	IB	RPS	—
PS N023C	IIA	RPS	—
PS N023D	IIB	RPS	—
PS N039 *	I	RELIEF VALVE	—
PS N045A		RECIRC (A)	—
PS N045B		RECIRC (B)	—
PS N045C		RECIRC (A)	—
PS N045D		RECIRC (B)	—
PS N047A,C	III	HPCS	—
PS N047B,D	III	HPCS	—
PS N048A,C	I	LPCS RHR(A)	—
PS N048B,D	II	RHR(B), RHR(C)	—
PT N051A	IA	————	K613A
PT N051B	IB	————	K613B
PS N015A	IA	NS4	—
PS N015B	IB	NS4	—
PS N015C	IIA	NS4	—
PS N015D	IIB	NS4	—
VS N056A	IA	NS4	—
VS N056B	IB	NS4	—
VS N056C	IIA	NS4	—
VS N056D	IIB	NS4	—
LT N027	—	————	K615
dPT N032	—	————	K615
LT-N027	—	————	K615
dPT-N032	—	————	K615
PT-N018	—	SRV TEST	—
PS-N413A,C	I	RHR(A), LPCS	—
PS-N413B,D	II	RHR(B) & (C)	—

FSAR FIG.

02-02B22-04	26	22	1
CVI NO	CVI SHT	REV	DWG SHT

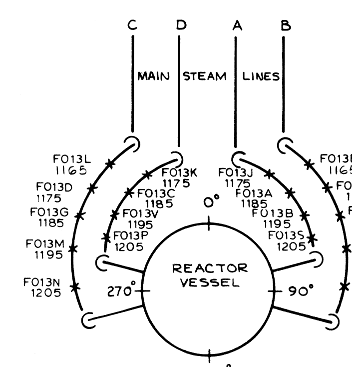


FIGURE I
SAFETY/RELIEF VALVE LOCATION & SPRING SET PRESSURE

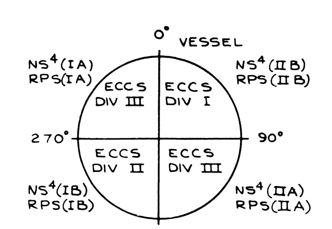
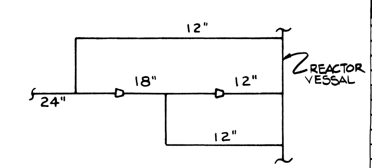


FIGURE II
DIVISION ORIENTATION FOR VESSEL INSTRUMENTATION (SEE REF 4)



FEEDWATER PIPING DETAIL
(TYPICAL)

Columbia Generating Station
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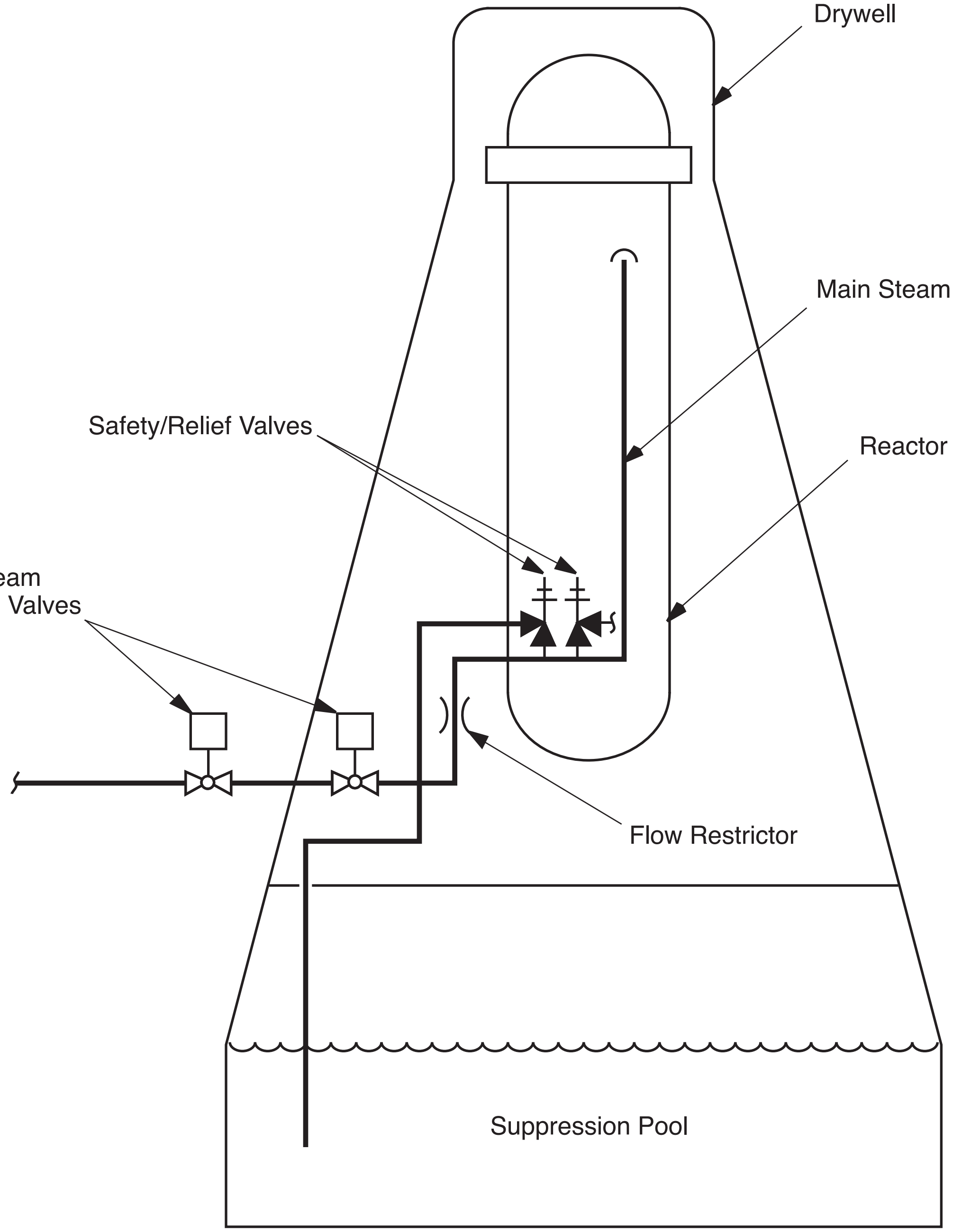
Nuclear Boiler System - P&ID

Draw. No. 02B22-04,26,1 | Rev. 22 | Figure 5.2-5



Columbia Generating Station
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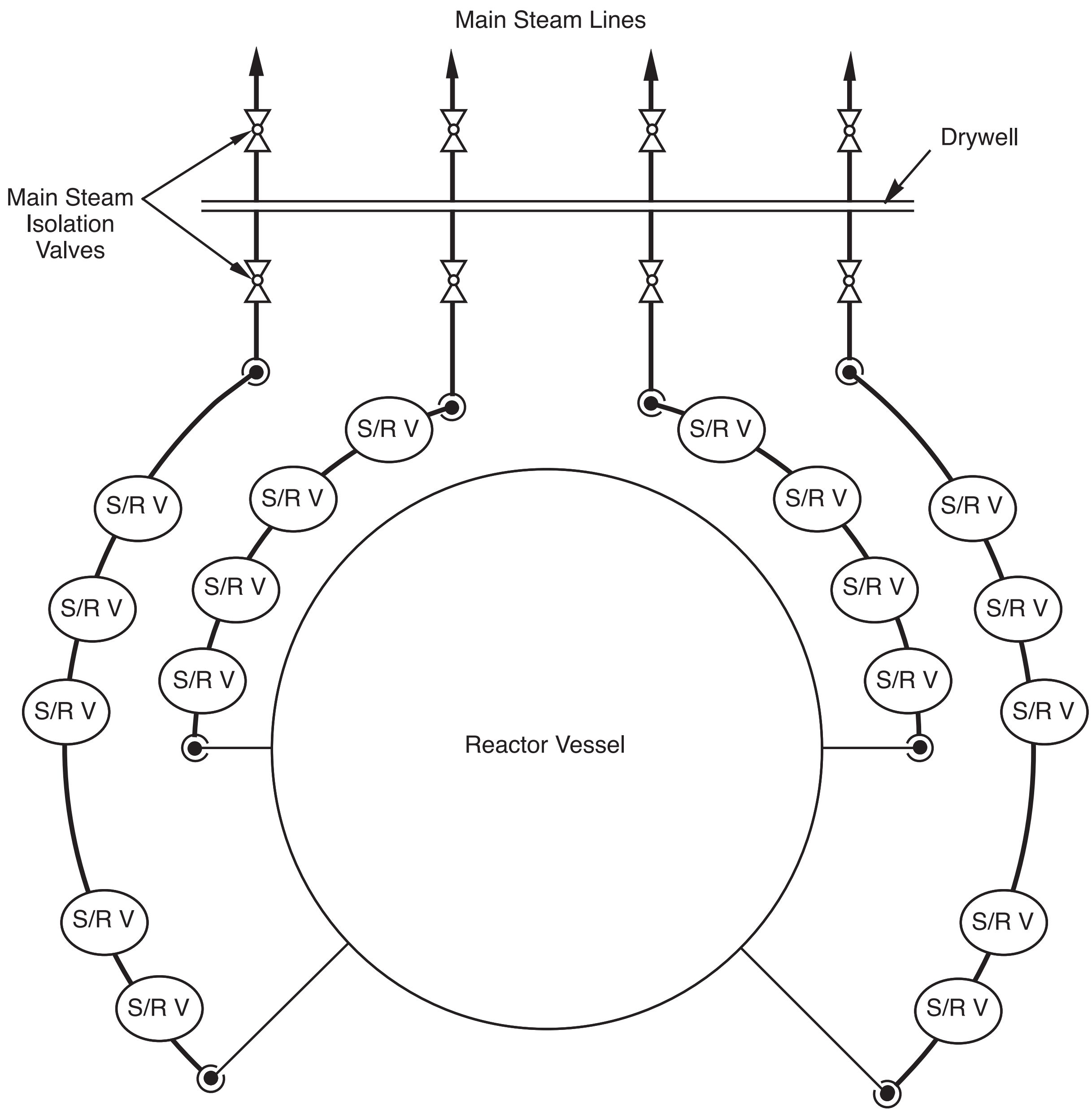
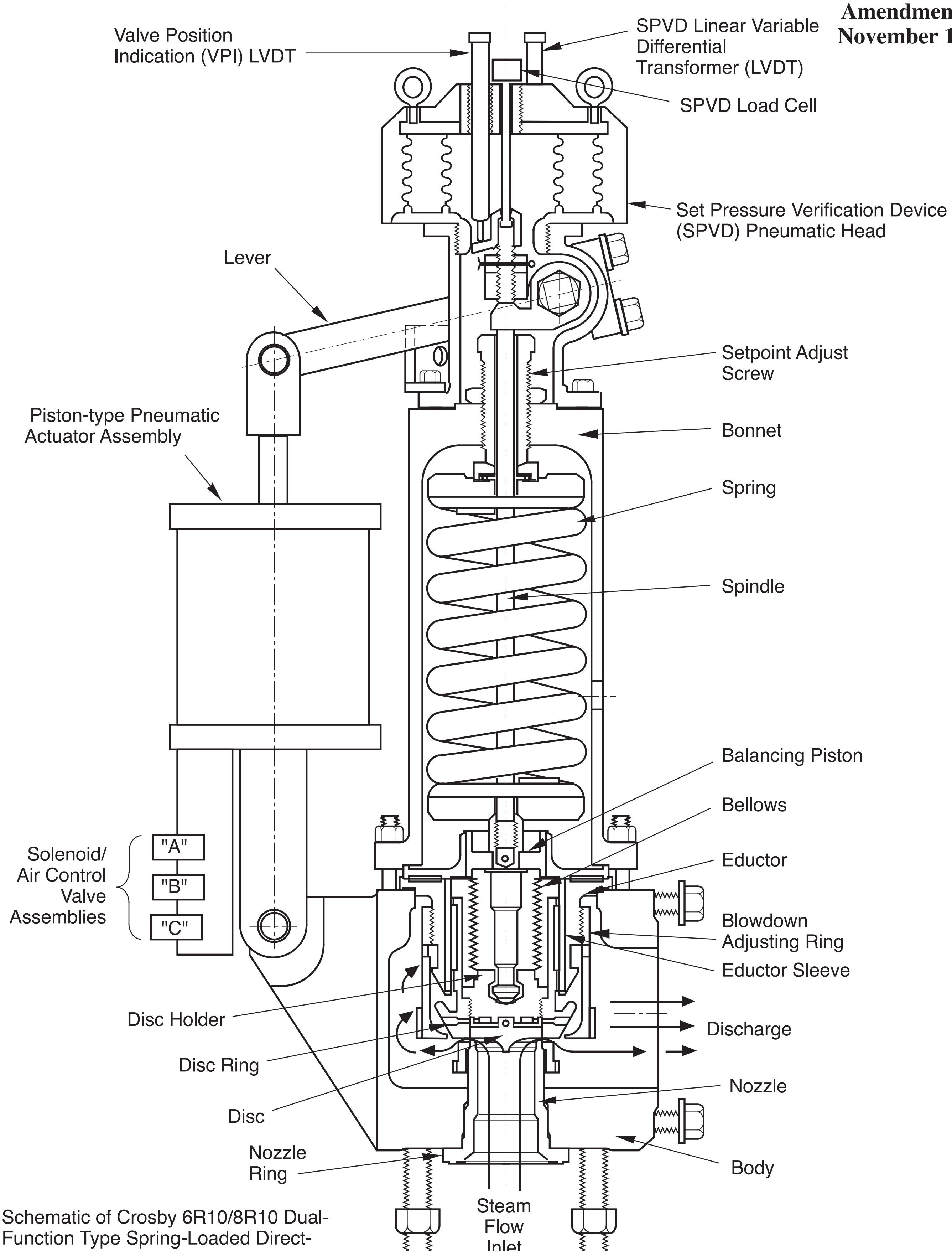
Safety/Relief Valve Schematic Elevation

Draw. No. 960690.49	Rev.	Figure 5.2-6



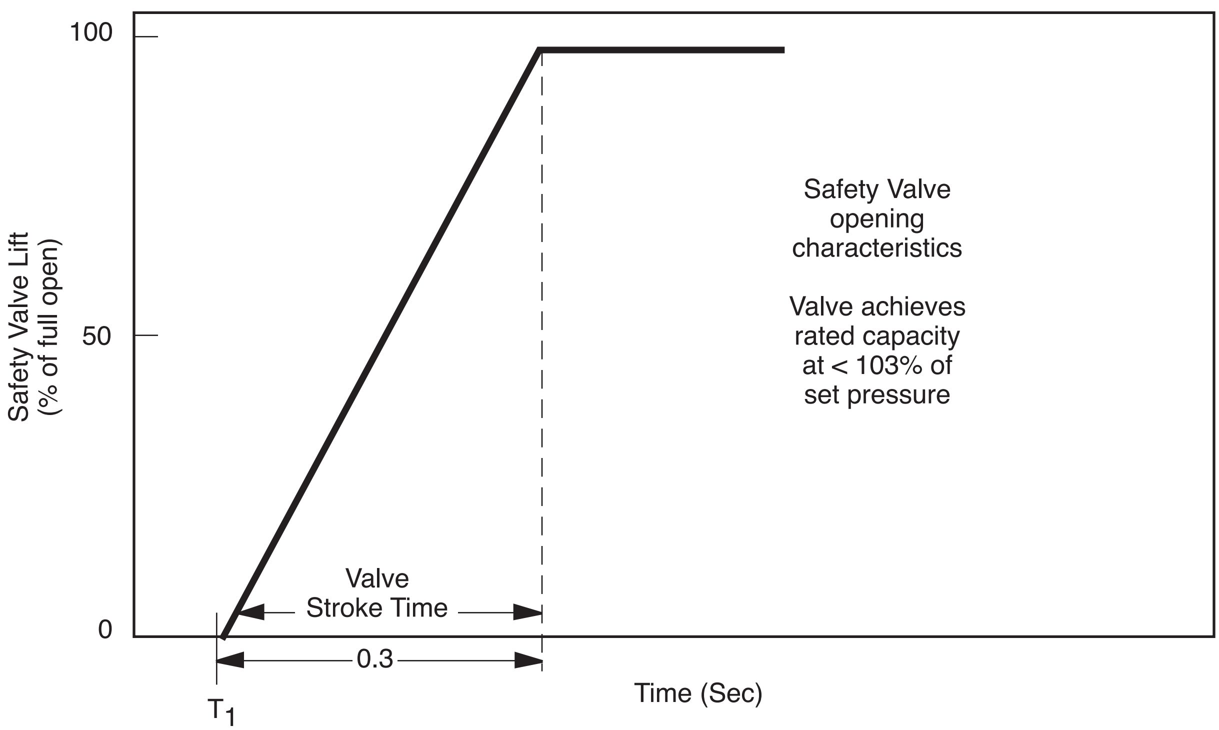
Safety/Relief Valve and Steam Line Schematic

Draw. No. 960690.62 | Rev. | Figure 5.2-7

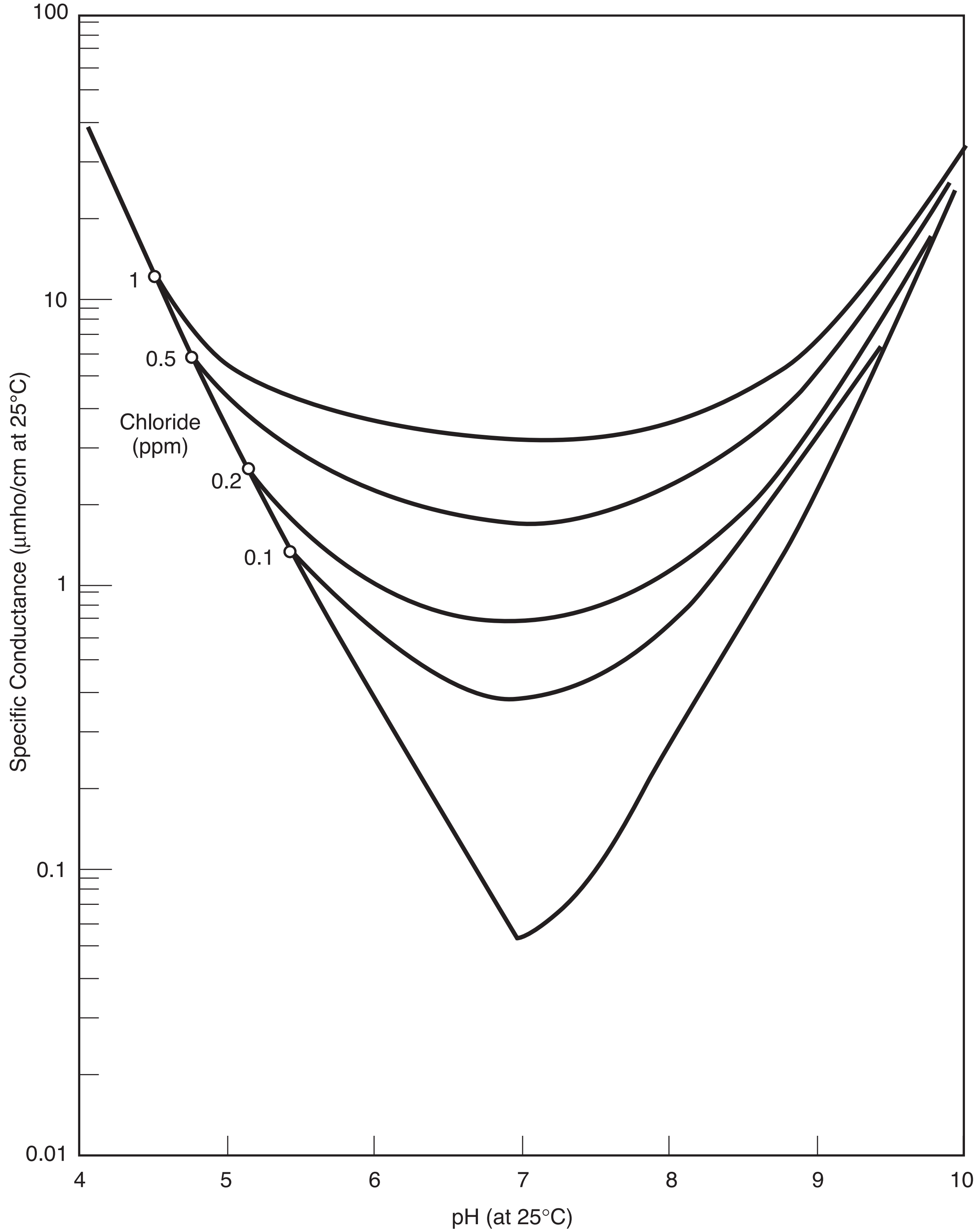


Schematic of Crosby 6R10/8R10 Dual-Function Type Spring-Loaded Direct-Acting Safety/Relief Valve

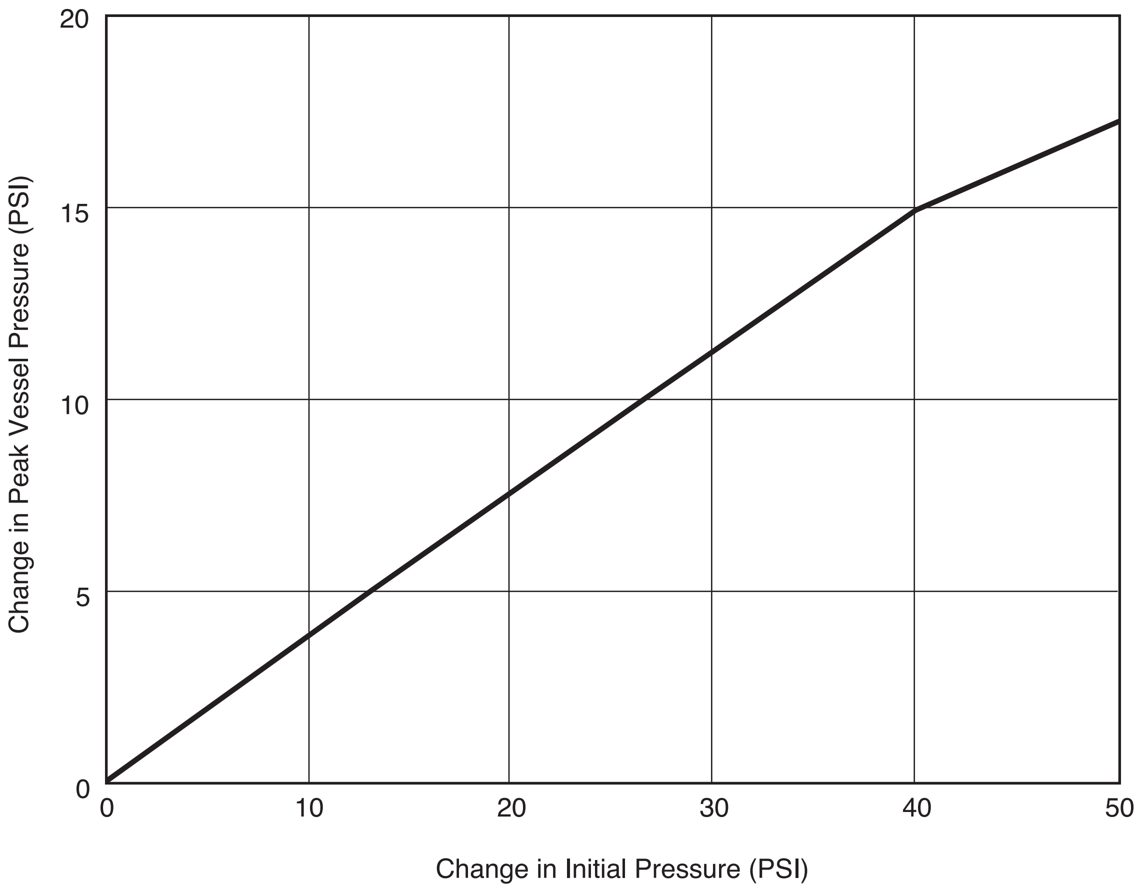
Columbia Generating Station Final Safety Analysis Report	Schematic of Safety Valve with Auxiliary Actuating Device		
	Draw. No. 960690.85	Rev.	Figure 5.2-8



T_1 = Time at which pressure exceeds the valve set pressure



Conductance Versus pH as a Function of Chloride Concentration of Aqueous Solution at 25°C



Typical BWR Characteristic MSIV Closure Flux Scram

5.3 REACTOR VESSEL

5.3.1 REACTOR VESSEL MATERIALS

5.3.1.1 Materials Specifications

The materials used in the reactor pressure vessel and appurtenances are shown in Table 5.2-7 together with the applicable specifications.

5.3.1.2 Special Processes Used for Manufacturing and Fabrication

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

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

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

All fabrication of the reactor pressure vessel is performed in accordance with the General Electric Company (GE) approved drawings, fabrication procedures, and test procedures. The shells and vessel heads are made from formed plates and the flanges and nozzles from forgings. Welding performed to join these vessel components is in accordance with procedures qualified per ASME Section III and IX requirements. Weld test samples are required for each procedure for major vessel full penetration welds. Tensile and impact tests are performed to determine the properties of the base metal, heat-affected zone (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, Subsection NB. Postweld heat treatment at 1100°F minimum is applied to all low alloy steel welds.

Radiographic examination is performed on all pressure containing welds in accordance with requirements of ASME Section III, Subsection NB-5320. In addition, all welds are given a supplemental ultrasonic examination.

The materials, fabrication procedures, and testing methods used in the construction of boiler water reactor (BWR) reactor pressure vessels meet or exceed requirements of ASME Section III, Class 1 vessels.

5.3.1.3 Special Methods for Nondestructive Examination

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

5.3.1.4 Special Controls for Ferritic and Austenitic Stainless Steels

The degree of compliance with Regulatory Guides 1.31, 1.34, 1.43, 1.44, 1.50, 1.71, and 1.99 is described in Section 1.8.

5.3.1.5 Fracture Toughness

5.3.1.5.1 Compliance with Code Requirements

The ferritic pressure boundary material of the reactor pressure vessels was qualified by impact testing in accordance with the 1971 edition of Section III ASME Code and Summer 1971 Addenda. From an operational standpoint, the minimum temperature limits for pressurization defined by the 1998 Edition of Section XI ASME Code and 2000 Addenda, Appendix G, Protection Against Nonductile Failure, are used as the basis for compliance with ASME Code Section III.

5.3.1.5.2 Compliance with 10 CFR 50 Appendix G

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

Ferritic material complying with 10 CFR 50 Appendix G must have both drop-weight tests and Charpy V-notch (CVN) tests with the CVN specimens oriented transverse to the maximum material working direction to establish the RT_{NDT}. The CVN tests must be evaluated against both an absorbed energy and a lateral expansion criteria. The maximum acceptable RT_{NDT} must be determined in accordance with the analytical procedures of ASME Code Section III, Appendix G. Appendix G of 10 CFR 50 requires a minimum of 75 ft-lb upper shelf CVN

energy for beltline material. It also requires at least 45 ft-lb CVN energy and 25 mils lateral expansion for bolting material at the lower of the preload or lowest service temperature.

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

From the previous comparison it can be seen that the fracture toughness testing performed on the CGS reactor vessel material cannot be shown to comply with 10 CFR 50 Appendix G. However, to determine operating limits in accordance with 10 CFR 50 Appendix G, estimates of the beltline material RT_{NDT} and the highest RT_{NDT} of all other material were made and are discussed in Section 5.3.1.5.2.2. The method for developing these operating limits is also described therein.

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

5.3.1.5.2.1 Intent of Proposed Approach. The intent of the proposed special method of compliance with 10 CFR 50 Appendix G for this vessel is to provide operating limitations on pressure and temperature based on fracture toughness. These operating limits ensure that a margin of safety against a nonductile failure of this vessel is very nearly the same as that for a vessel built to the Summer 1972 Addenda.

The specific temperature limits for operation when the core is critical are based on 10 CFR 50 Appendix G, Paragraph IV, A.2.C.

5.3.1.5.2.2 Operating Limits Based on Fracture Toughness. Operating limits which define minimum reactor vessel metal temperatures versus reactor pressure during normal heatup and cooldown and during inservice hydrostatic testing were established using the methods of Appendix G of Section XI of the ASME B&PV Code, 1998 Edition, 2000 Addenda. The results are shown in Figure 5.3-1.

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

For the purpose of setting these operating limits the reference temperature, RT_{NDT}, is determined from the toughness test data taken in accordance with requirements of the code to which this vessel is designed and manufactured. This toughness test data, CVN and/or dropweight NDT, is analyzed to permit compliance with the intent of 10 CFR 50 Appendix G. Because all toughness testing needed for strict compliance with Appendix G was not required at the time of vessel procurement some toughness results are not available. For example, longitudinal CVNs, instead of transverse, were tested, usually at a single test temperature of +10°F or -20°F, for absorbed energy. Also, at the time either CVN or NDT testing was permitted; therefore, in many cases both tests were not performed as is currently required. To substitute for this absence of certain data, toughness property correlations were derived for the vessel materials to operate on the available data to give a conservative estimate of RT_{NDT} compliant with the intent of Appendix G criteria.

These toughness correlations vary, depending upon the specific material analyzed, and were derived from the results of Welding Research Council (WRC) Bulletin 217, "Properties of Heavy Section Nuclear Reactor Steels," and from toughness data from the CGS vessel and other reactors. In the case of vessel plate material (SA-533 Grade 8, Class 1), the predicted limiting toughness property is either NDT or transverse CVN 50 ft-lb temperature minus 60°F. NDT values are available for CGS vessel shell plates. The transverse CVN 50 ft-lb transition temperature is estimated from longitudinal CVN data in the following manner. The lowest longitudinal CVN 50 ft-lb value is adjusted to derive a longitudinal CVN 50 ft-lb transition temperature by adding 2°F per ft-lb to the test temperature. If the actual data equals or exceeds 50 ft-lb, the test temperature is used. Once the longitudinal 50 ft-lb temperature is derived, an additional 30°F is added to account for orientation effects and to estimate the transverse CVN 50 ft-lb temperature minus 60°F, estimated in the preceding manner.

Using the above general approach, an initial RT_{NDT} of 28°F was established for the core beltline region.

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

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

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

Figure 5.3-2 provides a sketch of the reactor vessel, including the basic dimensions, all longitudinal and circumferential welds, and all plates of the beltline region. Tables 5.3-2 through 5.3-7 contain the supporting information for Figure 5.3-2, such as piece mark, heat number, and impact data for the plates and filler material used in the beltline region.

Closure bolting material (SA-540 Grade B24) toughness test requirements for CGS were for 30 ft-lb at 60°F below the boltup temperature. Current code requirements are for 45 ft-lb and 25 mils lateral expansion at the preload or lowest service temperature, including boltup. All CGS closure stud materials meet current requirements at +10°F.

The effect of the main closure flange discontinuity was considered by adding 60°F to the RT_{NDT} to establish the minimum temperature for boltup and pressurization. The minimum boltup temperature of 80°F for CGS, which is shown on Figure 5.3-1, is based on an initial RT_{NDT} of +20°F for the shell plate connecting to the closure flange forgings.

The effect of the feedwater nozzle discontinuities were considered by adjusting the results of a BWR/6 reactor discontinuity analysis to the reactor. The adjustment was made by increasing the minimum temperatures required by the difference between the CGS and BWR/6 feedwater nozzle forging RT_{NDT}. The feedwater nozzle adjustment was based on an RT_{NDT} of 0°F.

The reactor vessel closure studs have a minimum Charpy impact energy of 45 ft-lb and 26 mils lateral expansion at 10°F. The lowest service temperature for the closure studs is 10°F.

Vessel irradiation embrittlement of beltline materials, as measured by adjusted reference temperatures and upper shelf energies due to increased flux, was evaluated against the requirements of 10 CFR 50 Appendix G. For a predicted fluence of 7.41 x 10^{17}n/cm^2, fracture toughness values are acceptable and remain within Appendix G limits.

5.3.1.5.2.3 Temperature Limits for Boltup. A minimum temperature of 10°F is required for the closure studs. A sufficient number of studs may be tensioned at 70°F to seal the closure flange O-rings for the purpose of raising reactor water level above the closure flanges to assist in warming them. The flanges and adjacent shell are required to be warmed to a minimum temperature of 80°F before they are stressed by the full intended bolt preload. The fully preloaded boltup limits are shown in Figure 5.3-1.

5.3.1.5.2.4 Inservice Inspection Hydrostatic or Leak Pressure Tests. Based on 10 CFR 50 Appendix G, and Regulatory Guide 1.99, Revision 2, requirements, pressure/temperature limit curves were established based on an RT_{NDT} of 28°F for the limiting beltline material; see Figure 5.3-1. The fracture toughness analysis for inservice inspection of leak test resulted in

curve A shown in Figure 5.3-1. The predicted shift in the RT_{NDT} temperature was determined using the methodology outlined in Regulatory Guide 1.99, Revision 2.

Technical Specification 3.10.1 allows inservice leak and hydrostatic testing to be performed in Mode 4 when the metallurgical characteristics of the reactor pressure vessel require testing at temperatures greater than 200°F, given specified Mode 3 Limiting Conditions for Operations are met. This exemption is only applicable provided reactor coolant temperature does not exceed 275°F.

5.3.1.5.2.5 Operating Limits During Heatup, Cooldown, and Core Operation. The fracture toughness analysis was done for the normal 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 analysis are operating limits defined by Figure 5.3-1. Curves A, B, and C give the limits for hydrotest, nonnuclear heating, and nuclear heating. The minimum boltup temperature of 80°F is based on an RT_{NDT} at 20°F for a shell plate which connects to the lower flange (Heat and Slab No. C-1307-2); above 80°F the core beltline plate (Heat and Slab No. C-1272-1), which has an initial RT_{NDT} of 28°F, is most limiting for inservice hydrostatic or leak pressure tests (curve A). The feedwater nozzles, which have an RT_{NDT} of 0°F, are more restrictive than the core beltline at lower pressures during nonnuclear and nuclear heating (curves B and C).

5.3.1.5.2.6 Reactor Vessel Annealing. Inplace annealing of the reactor vessel to counteract radiation embrittlement is unnecessary because beltline material adjusted reference temperature of the NDT is well within the 10 CFR 50 Appendix G 200°F screening limit.

5.3.1.6 Material Surveillance

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

The CGS plant-specific RPV materials surveillance program is replaced by the NRC approved BWR Vessel and Internals Project (BWRVIP) Integrated Surveillance Program (ISP), as described in the latest approved revision of BWRVIP-86 (Reference 5.3.4-2). The ISP meets the requirements of 10 CFR 50, Appendix H.

The current surveillance capsule withdrawal schedule for the representative materials for the CGS vessel is based on the latest approved revision of BWRVIP-86 (Reference 5.3.4-2). No capsules from the CGS vessel are included in the ISP. The withdrawal of capsules for the CGS plant-specific surveillance program is permanently deferred by participation in the ISP. Capsules from other plants will be removed and tested in accordance with the ISP

implementation plan. The results from these tests will provide the necessary data to monitor embrittlement for the CGS vessel.

Materials for the plant-specific materials surveillance program were selected to represent materials used in the reactor beltline region. The specimens were manufactured from a plate actually used in the beltline region and a weld typical of those in the beltline region and thus represent base metal, weld metal, and the transition zone between base metal and weld. The plate and weld were heat treated in a manner which simulates the actual heat treatment performed on the core region shell plates of the completed vessel. WPPSS-ENT-089 (Reference 5.3.4-1) provides additional detail and supporting information for the materials surveillance program.

For the extent of compliance to 10 CFR 50 Appendix H, see Table 5.3-8. NEDO-21708 also addressed the requirements of Appendix H to 10 CFR 50 and supports the current application of Regulatory Guide 1.99.

5.3.1.6.1 Positioning of Surveillance Capsules and Method of Attachment for Plant-Specific Surveillance Program

Surveillance specimen capsules are located at three azimuths at a common elevation in the core beltline region. The sealed capsules are not attached to the vessel but are in welded capsule holders. The capsule holders are mechanically restrained by capsule holder brackets as shown in Figure 5.3-4. The capsule holder brackets allow the capsule holder to be removed at any desired time in the life of the plant for specimen testing. A positive spring-loaded locking device is provided to retain the capsules in position throughout any anticipated event during the lifetime of the vessel.

The capsule holder brackets are designed, fabricated, and analyzed to the requirements of the ASME B&PV Code Section III. The surveillance brackets are welded to the clad material which surfaces the pressure vessel walls. As attached, the brackets do not have to comply with specifications of the ASME Code.

5.3.1.6.2 Time and Number of Dosimetry Measurements

General Electric provides a separate neutron dosimeter so that fluence measurements may be made at the vessel ID during the first fuel cycle to verify the predicted fluence at an early date in plant operation. This measurement is made over this short period to avoid saturation of the dosimeters now available. Once the fluence-to-thermal power output is verified, no further dosimetry is considered necessary because of the linear relationship between fluence and power output.

5.3.1.6.3 Neutron Flux and Fluence Calculations

A description of the methods of analysis for neutron flux and fluence calculations is contained in Sections 4.1.4.5 and 4.3.2.8.

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 thread hole in its vessel shell flange. A nut and washer are installed on the upper end of each stud. The proper amount of preload can be applied to the studs by sequential tensioning using hydraulic tensioners. The design and analysis of this area of the vessel is in full compliance with all Section III Class 1 Code requirements. The material for studs, nuts, and washers is SA-540, Grade B23 or B24. The maximum reported ultimate tensile stress for the bolting material was 167,000 psi which is less than the 170,000 psi limitation in Regulatory Guide 1.65. Also the Charpy impact test recommendations of Paragraph IV.A.4 of Appendix G to 10 CFR 50 were not specified in the vessel order since the order was placed prior to issuance of Appendix G to 10 CFR 50. However, impact data from the certified materials report shows that all bolting material has met the Appendix G impact properties. For example, the lowest reported CVN energy was 45 ft-lb at 10°F versus the required 45 ft-lb at 70°F and the lowest reported CVN expansion was 26 mils at 10°F versus the required 25 mils at 70°F.

Hardness tests are performed on all main closure bolting to demonstrate that heat treatment has been properly performed. Studs, nuts, and washers are ultrasonically examined in accordance with Section III, N8-2585 and the following additional requirements:

a. Examination is performed after heat treatment and prior to machining threads.

b. Straight beam examination is performed on 100% of each stud. Reference standard for the radial scan is 0.5-in. diameter flat bottom hole having a depth equal to 10% of the material thickness. For the end scan the reference standard is a 0.5-in. flat bottom hole having a depth of 0.5 in. For additional details of the techniques used to examine the reactor vessel studs, see the response to Regulatory Guide 1.65, Revision 0, October 1973, in Section 1.8.

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

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

5.3.2 PRESSURE-TEMPERATURE LIMITS

5.3.2.1 Limit Curves

Limits on pressure and temperature for inservice leak and hydrostatic tests, normal operation (including heatup and cooldown), and reactor core operation are shown in Figure 5.3-1. The basis used to determine these limits is described in Section 5.3.1.5.

5.3.2.2 Operating Procedures

By comparison of the pressure versus temperature limits in Figure 5.3-1 with intended normal operating procedures for the most severe upset transient, it is shown that the limits will not be exceeded during any foreseeable upset condition. Reactor operating procedures have been established such that actual transients will not be more severe than those for which the vessel design adequacy has been demonstrated. Of the design transients, the upset condition producing the most adverse temperature and pressure condition anywhere in the vessel head and/or shell areas has a minimum fluid temperature of 250°F and a maximum pressure peak of 1180 psig. Scram automatically occurs with initiation of this event, prior to the reduction in fluid temperature, such that the applicable operating limits are bounded by curve A of Figure 5.3-1. Figure 5.3-1 show that at the maximum transient pressure of 1180 psig, the minimum allowable reactor vessel metal temperature conservatively bounds the minimum 250°F reactor fluid temperature.

5.3.3 REACTOR VESSEL INTEGRITY

The reactor vessel was fabricated for GE's Nuclear Energy Division by CBI Nuclear Co., and was subject to the requirements of GE's Quality Assurance program.

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

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

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

5.3.3.1 Design

5.3.3.1.1 Description

5.3.3.1.1.1 Reactor Vessel. The reactor vessel shown in Figure 5.3-5 is a vertical, cylindrical pressure vessel of welded construction. The vessel is designed, fabricated, tested, inspected, and stamped in accordance with the ASME Code Section III, Class 1, including the addenda in effect at the date of order placement. Design of the reactor vessel and its support system meets Seismic Category I equipment requirements. The materials used in the reactor pressure vessel are shown in Table 5.2-7.

The cylindrical shell and bottom head sections of the reactor vessel are fabricated of low alloy steel, the interior of which is clad with stainless steel weld overlay. Nozzle and nozzle weld zones are unclad except for those mating to stainless steel piping systems.

Inplace annealing of the reactor vessel is unnecessary because shifts in transition temperature caused by irradiation during the 40-year life can be accommodated by raising the minimum pressurization temperature. Radiation embrittlement is not a problem outside of the vessel beltline region because the irradiation in those areas is less than 1×10^{18} nvt with neutron energies in excess of 1 MeV. The inside diameter and minimum wall thickness of the reactor vessel beltline is provided in Table 5.3-9.

Quality control methods used during the fabrication and assembly of the reactor vessel and appurtenances ensure that design specifications were met. The vessel top head is secured to the reactor vessel by studs and nuts. These nuts are tightened with a stud tensioner. The vessel flanges are sealed with two concentric metal seal rings designed to permit no detectable leakage through the inner or outer seal at any operating condition, including heating to operating pressure and temperature at a maximum rate of 100°F/hr in any 1-hr period. To detect seal failure, a vent tap is located between the two seal rings. A monitor line is attached to the tap to provide an indication of leakage from the inner seal ring seal.

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

5.3.3.1.1.3 Protection of Closure Studs. The BWR does not use borated water for reactivity control. This section is therefore not applicable.

5.3.3.1.2 Safety Design Bases

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

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

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

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

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

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

5.3.3.1.3 Power Generation Design Basis

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

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

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

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

5.3.3.1.4 Reactor Vessel Design Data

Reactor vessel design data are contained in Tables 5.2-6 and 5.2-7.

5.3.3.1.4.1 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.

5.3.3.1.4.2 Control Rod Drive Housings. The control rod drive (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 CRD tube, a four-lobed fuel support piece, and the four fuel assemblies that rest on the fuel support piece. The housings are fabricated of Type 304 austenitic stainless steel.

5.3.3.1.4.2.1 Control Rod Drive Return Line. To preclude CRD return line cracking on CGS, the return line was deleted and the system modified. The modification consists of adding pressure equalizing valves between the exhaust and cooling water headers and the use of reverse flow through multiple hydraulic control unit (HCU) solenoid valves as the CRD system exhaust flow path. The acceptance of this modification is based on system analyses and performance tests conducted on operating BWRs which have shown satisfactory system operation. The system tests showed that system pressure transients, CRD settling times, and CRD speeds were all unchanged. The tests also showed that all systems functions performed normally.

5.3.3.1.4.3 In-Core Neutron Flux Monitor Housings. Each in-core neutron flux monitor housing is inserted through the in-core penetrations in the bottom head and is welded to the inner surface of the bottom head.

An in-core flux monitor guide tube is welded to the top of each housing and either a source range monitor/intermediate range monitor drive unit or a local power range monitor is bolted to the seal/ring flange at the bottom of the housing.

5.3.3.1.4.4 Reactor Vessel Insulation. The insulation panels for the cylindrical shell of the vessel are self-supporting, with seismic restraints attached to the sacrificial shield wall. The insulation is designed to be removable over those portions of the vessel where required for the purpose of in-service inspection.

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

The vessel top head nozzle is provided with a flange with large groove facing. The drain nozzle is of the full penetration weld design. The recirculation inlet nozzles (located as shown in Figure 5.3-5), feedwater inlet nozzles, core spray inlet nozzles, low-pressure coolant injection (LPCI) nozzles, and the CRD hydraulic system return nozzle all have thermal sleeves. Nozzles connecting to stainless steel piping have safe ends or extensions made of

stainless steel. These safe ends or extensions were welded to the nozzles after the pressure vessel was heat treated to avoid furnace sensitization of the stainless steel. The material used is compatible with the material of the mating pipe.

The nozzle for the standby liquid control (SLC) pipe was designed to minimize thermal shock effects on the reactor vessel in the event of injection of cold SLC solution. However, the SLC injection pipe has been relocated to a nozzle on the high-pressure core spray (HPCS) injection line and no longer uses the old nozzle in the bottom head of the reactor pressure vessel. The old nozzle is still in service as the connection for pressure sensing below the core plate, but there is no flow through the nozzle under any operating condition.

In the past, thermal fatigue cracking of feedwater nozzles and vibrational cracking of sparger arms have been observed at other operating BWRs. The mechanisms which have caused cracking in other operating BWRs are understood. A summary discussion of these problems and the solutions incorporated in the CGS design is presented in the following.

A detailed evaluation of the problems of the feedwater nozzle and sparger is presented in NEDE-21821, "BWR Feedwater Nozzle/ Sparger Final Report," March 1978. The solution of the feedwater nozzle and sparger cracking problems involved several elements, including material selection and processing, nozzle clad elimination, and thermal sleeve and sparger redesign. The following summarizes the problems and solutions that have been implemented in the CGS design.

Problem	Cause	Fix
Sparger arm cracks	Vibration	Eliminated clearance between thermal sleeve and safe end
RPV feedwater thermal fatigue	Thermal	Eliminated clad, eliminated leakage with a welded joint between the sparger and safe end

The sparger vibration has been attributed to a self-excitation caused by instability of leakage flow through the annular clearance between the thermal sleeve and safe end. Tests have shown that the vibration is eliminated if the clearance is reduced sufficiently or sealed. The solution that was selected for CGS uses a welded joint to ensure no leakage. This feature is also an essential part of the solution of the nozzle cracking problem. Freedom from vibration over a range of conditions has been demonstrated by the tests reported in NEDE-23604 (see Figures 5.3-6 and 5.3-7).

The cracking of the feedwater nozzles is a two-part process. The crack initiation mechanism as discussed above is the result of self-initiated thermal cycling. If this were the only mechanism present, the cracks would initiate, grow to a depth of approximately 0.25 in., and arrest. This degree of cracking could be tolerated; however, there is another mechanism which supports crack growth. This mechanism is the system induced transients, primarily the startup/shutdown transients. Because of CGS's welded thermal sleeve arrangement, leakage flow is eliminated and the heat transfer between the feedwater and the nozzle are reduced to the point where the thermal stresses in the nozzle are not high enough to cause a significant crack growth. Analyses presented in NEDE-21821, Section 4.7, demonstrated the benefits of the welded thermal sleeve and of using unclad nozzles. With these demonstrated benefits and inservice surveillance, CGS found it unnecessary to install instrumentation for design verification.

CGS has installed two automatic feedwater low flow control valves, RFW-FCV-10A and 10B. These valves have the capacity to control flow down to 362 gpm, or about 1.25% of total flow. This valve configuration will substantially reduce the temperature differential between the feedwater and the water in the RPV during low power operation, also reducing the thermal stresses in the nozzle.

5.3.3.1.4.6 Materials and Inspection. The reactor vessel was designed and fabricated in accordance with the appropriate ASME B&PV Code as defined in Section 5.2.1.2. Table 5.2-7 defines the materials and specifications. Table 5.3-8 defines the compliance with reactor vessel material surveillance program requirements.

5.3.3.1.4.7 Reactor Vessel Schematic (BWR). The reactor vessel schematic is contained in Figure 5.3-3. Trip system water levels are indicated as shown.

5.3.3.2 Materials of Construction

All materials used in the construction of the reactor pressure vessel conform to the requirements of ASME Code Section II materials. The vessel heads, shells, flanges, and nozzles are fabricated from low alloy steel plate and forgings purchased in accordance with ASME specifications SA533 Grade B Class 1 and SA-508 Class 2. Special requirements for the low alloy steel plate and forgings are discussed in Section 5.3.1.2. Cladding employed on the interior surfaces of the vessel consists of austenitic stainless steel weld overlay.

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

5.3.3.3 Fabrication Methods

The reactor pressure vessel is a vertical cylindrical pressure vessel of welded construction fabricated in accordance with ASME Code Section III, Class 1, requirements. All fabrication of the reactor pressure vessel was performed in accordance with buyer-approved drawings, fabrication procedures, and test procedures. The shells and vessel heads were made from formed low alloy steel plates and the flanges and nozzles from low alloy steel forgings. Welding performed to join these vessel components was in accordance with procedures qualified per ASME Section III and IX requirements. Weld test samples were required for each procedure for major vessel full penetration welds.

Submerged arc 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 NB. Postweld heat treatment of 1100°F minimum was applied to all low alloy steel welds.

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

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

5.3.3.4 Inspection Requirements

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

5.3.3.5 Shipment and Installation

The completed reactor vessel was given a thorough cleaning and examination prior to shipment. The vessel was tightly sealed for shipment to prevent entry of dirt or moisture. Preparations for shipment 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, periodic cleanings were performed, and only approved miscellaneous materials were used during assembly.

5.3.3.6 Operating Conditions

Restrictions on plant operation to hold thermal stresses within acceptable ranges are included in the Technical Specifications. These restrictions on coolant temperature are

a. The average rate of change of reactor coolant temperature during normal heatup and cooldown,

b. Coolant temperature difference between the dome (inferred from P_{sat}) and the bottom head drain, and

c. Idle reactor recirculation loop and average reactor coolant temperature differential.

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

The above operational limits when maintained insure 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 material 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. Reactor vessel material integrity is also maintained during abnormal operating conditions where safety systems or controls provide an automatic response in the reactor vessel. The special and transient events considered in the design of the vessel are discussed or referenced in Section 5.2.2.

5.3.3.7 Inservice Surveillance

Inservice inspection of the reactor pressure vessel is in accordance with the requirements as discussed in Section 5.2.4. The vessel was examined once prior to startup to satisfy the preoperational requirements of IS-232 or the ASME Code, Section XI. Subsequent inservice

inspection will be scheduled and performed in accordance with the requirements of 10 CFR 50.55a subparagraph (g).

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. See Section 5.3.1.6 for description of the materials surveillance program. Operating procedures will be modified in accordance with test results to ensure adequate brittle fracture control.

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

5.3.4 REFERENCES

5.3.4-1 WPPSS-ENT-089, “WNP-2 RPV Surveillance Program,” Current Revision.

5.3.4-2 BWRVIP-86, Revision 1-A, “BWR Vessel and Internals Project, Updated BWR Integrated Surveillance Program (ISP) Implementation Plan,” Final Report, October 2012.

Table 5.3-1

10 CFR 50 Appendix G Matrix

Appendix G Paragraph	Topic	Comply Yes/No or N/A	Alternative Actions or Comments
I, II	Introduction; Definitions	--	
III.A	Compliance with ASME Code, Section NB-2300	Yes	See Section 5.3.1.5.2 for discussion.
III.B.1	Location and Orientation of Impact Test Spec	Yes	See III.A above.
III.B.2	Materials Used to Prepare Test Specimens	No	Compliance except for CVN orientation and CVN upper shelf.
III.B.3	Calibration of Temperature Instruments and Charpy Test Machines	No	Paragraph NB-2360 of the ASME B&PV Code Section III was not in existence at the time of purchase of the CGS reactor pressure vessel. However, the requirements of the 1971 edition of the ASME B&PV Section III code, Summer 1971 addenda, were met. For the discussions of the GE interpretations of compliance and NRC acceptance see References 1 and 2. The temperature instruments and Charpy Test Machines calibration data are retained until the next recalibration. This is in accordance with Regulatory Guide 1.88, Revision 2, GE Alternative Position 1.88, and ANSI N45.2.9-1974. Therefore, the instrument calibration data for CGS would not be currently available.
III.B.4	Qualification of Testing Personnel	No	No written procedures were in existence as required by the regulation; however, the individuals were qualified by on-the-job training and past experience. For the discussion of the GE interpretation of compliance and NRC acceptance see References 1 and 2.
III.B.5	Test Results Recording and Certification	Yes	See References 1 and 2.
III.C.1	Test Conditions	No	See III.A, III.B.2 above.

Table 5.3-1

10 CFR 50 Appendix G Matrix (Continued)

Appendix G Paragraph	Topic	Comply Yes/No or N/A	Alternative Actions or Comments
III.C.2	Materials Used to Prepare Test Specimens for Reactor Vessel Beltline	Yes	Compliance on base metal and weld metal tests. Test weld not made on same heat of base plate necessarily.
IV.A.1	Acceptance Standard of Materials	--	
IV.A.2.a	Calculates Stress Intensity Factor	Yes	
IV.A.2.b	Requirements for Nozzles, Flanges, and Shell Region Near Geometric Discontinuities	No	Plus 60°F was added to the RT_{NDT} for the reactor vessel flanges. For feedwater nozzles the results of the BWR/6 analysis was adjusted to CGS RT_{NDT} conditions.
IV.A.2.c	RPV Metal Temperature Requirement When Core is Critical	Yes	Comply with 10 CFR 50 Appendix G.
IV.A.2.d	Minimum Permissible Temperature During Hydro Test	Yes	
IV.A.3	Materials for Piping, Pumps, and Valves	No	Main steam line piping is in compliance. See 5.2.3.3 for discussions on pumps and valves.
IV.A.4	Materials for Bolting and Other Fasteners	Yes	Current toughness requirements for closure head studs are met at +10°F even though testing was done per the 1971 ASME code.
IV.B	Minimum Upper Shelf Energy for RPV Beltline	No	Weld and longitudinal CVN data were taken at -20°F and +10°F only. An estimate of compliance to requirements should be made from the first surveillance capsule results per MTEB 5-2.

Table 5.3-1

10 CFR 50 Appendix G Matrix (Continued)

Appendix G Paragraph	Topic	Comply Yes/No or N/A	Alternative Actions or Comments
IV.B (continued)			Beltline plates were tested with longitudinal CVNs at +10°F only. The minimum values are for Heat C1272-1 (0.15% Cu; 34, 26, 30, 31, 34, 30 ft-lb; 10 and 40% shear at +10°F) and Heat C1273-1 (0.14% Cu; 33, 33, 30, 30, 34, 35 ft-lb; 10% shear at +10°F). Beltline welds were tested with CVNs at 10°F or -20°F only. Lowest weld values are found for Heat 04P046/Lot D217A27A (0.06% Cu; 34, 36, 37, 39, 40 ft-lb; 20 and 30% shear at -20°F). Heat C3L46C/Lot J020A27A (0.02% Cu; 35, 39, 40 ft-lb; 60% shear at +10°F) and Heat 05P018/Lot D211A27A (0.09% Cu; 29, 30, 31, 36, 38 ft-lb; 30 and 40% shear at -20°F). Because of the preceding relatively low test temperatures and Cu contents, it is anticipated that end-of-life upper shelf CVN values would be in excess of 50 ft-lb.
IV.C	Requirements for Annealing when RT_{ndt} >200	N/A	
V.A	Requirements for Material Surveillance Program	See Table 5.3-8	
V.B	Conditions for Continued Operation	Yes	Requirements for continued operations are covered in Technical Specifications and the Reactor Pressure Vessel Surveillance Program document (WPPSS-ENT-089, Reference 5.3.4-1). See Section 5.3.1.6 for description of the Materials Surveillance Program.
V.C	Alternative if V.B Cannot be Satisfied	N/A	The Surveillance Program demonstrates compliance with Appendix G, Section IV. See Section 5.3.1.6 for description of the Materials Surveillance Program.

Table 5.3-1

10 CFR 50 Appendix G Matrix for (Continued)

Appendix G Paragraph	Topic	Comply Yes/No or N/A	Alternative Actions or Comments
V.D	Requirement for RPV Thermal Annealing if V.C Cannot be Met	N/A	
V.E	Reporting Requirements for V.C and V.D	N/A	

REFERENCES

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

2. Letter from Robert B. Minoque, NRC, to G. G. Sherwood, GE, dated February 14, 1978.

Table 5.3-2

Plate Material Cross Reference

	Heat	Slab
Ring 21		
PCMK 21-1-1	C1272	1
PCMK 21-1-2	C1273	1
PCMK 21-1-3	C1273	2
PCMK 21-1-4	C1272	2
Ring 22		
PCMK 22-1-1	B5301	1
PCMK 22-1-2	C1336	1
PCMK 22-1-3	C1337	1
PCMK 22-1-4	C1337	2

Table 5.3-3

Weld Material Cross Reference

Weld Identification	Type	Heat	Lot
AB - Girthweld	E8018NM	492L4871	A422B27AF
	RAC01NMM	5P6756	0342
	RAC01NMM	3P4955	0342
	E8018NM	04T931	A423B27AG
Ring 21			
BA	E8018NM	04P046	D217A27A
	E8018NM	07L669	K004A27A
	RAC01NMM	3P4966	1214
BB	E8018NM	04P046	D217A27A
	E8018NM	07L669	K004A27A
	E8018NM	C3L46C	J020A27A
	RAC01NMM	3P4966	1214
	E8018NM	08M365	G128A27A
BC	E8018NM	09L853	A111A27A
	E8018NM	C3L46C	J020A27A
	RAC01NMM	3P4966	1214
BD	E8018NM	C3L46C	J020A27A
	RAC01NMM	3P4966	1214
	E8018NM	04P046	D217A27A
	E8018NM	C3L46C	J020A27A
Ring 22			
BE	RAC01NMM	3P4966	1214
BF	E8018NM	04P046	D217A27A
	E8018NM	05P018	D211A27A
	RAC01NM	3P4966	1214
BG	E8018NM	624063	C228A27A
	E8018NM	624039	D224A27A
	RAC01NMM	3P4966	1214
BH	E8018NM	04P096	D217A27A
	E8018NM	624039	D205A27A
	RAC01NMM	3P4966	1214

Table 5.3-4

Plate Material

	Charpy Impact ft-lb @ +10°F	Charpy Expansion MLE	Drop Weight NDT (°F)	RT_{NDT} (°F)
Ring 21				
PCMK 21-1-1 Heat C1272-1	34, 26, 30/31, 34, 30	30, 34, 24/27, 26, 32	–10	28
PCMK 21-1-2 Heat C1273-1	33, 33, 30/30, 34, 35	30, 31, 27/26, 34, 32	–20	20
PCMK 21-1-3 Heat C1273-2	38, 48, 55/66, 61, 71	44, 39, 34/53, 52, 56	–30	4
PCMK 21-1-4 Heat C1272-2	40, 42, 44/51, 55, 50	32, 36, 38/41, 44, 42	–30	0
Ring 22				
PCMK 22-1-1 Heat B5301-1	64, 62, 66/52, 52, 55	56, 56, 56/45, 44, 44	–30	–20
PCMK 22-1-2 Heat C1336-1	70, 72, 71/60, 44, 66	59, 60, 62/56, 41, 51	–30	–8
PCMK 22-1-3 Heat C1337-1	71, 76, 74/70, 72, 55	61, 60, 60/63, 61, 52	–30	–20
PCMK 22-1-4 Heat C1337-2	62, 72, 82/73, 67, 73	51, 61, 66/52, 59, 61	–50	–20

Table 5.3-5

Weld Material

Type/Heat/Lot/Control	Charpy Impact (ft-lb)	Charpy Expansion MLE	Charpy Test Temperature (°F)	RT_{NDT} (°F)
<u>Girth Weld AB</u>				
E8018NM/492L4871 Lot A422B27AF	78, 82, 105, 93, 81	55, 60, 72, 74, 60	–20[a]	–50
RAC01NMM/5P6756[b] Lot 0342	76, 79, 77, 80, 72	64, 72, 55, 69, 60	+10	–50
RAC01NMM/5P6756[c] Lot 0342	76, 79, 77, 80, 72	64, 72, 55, 69, 60	+10	–50
RAC01NMM/3P4955[b] Lot 0342	49, 63, 47, 49, 64	39, 48, 36, 43, 57	+10	–20
RAC01NMM/3P4955[c] Lot 0342	52, 37, 45, 55, 33	44, 30, 43, 50, 32	+10	–16
E8018NM/04T931 Lot A423B27AG	86, 84, 102, 63, 61	69, 58, 60, 57, 70	–20	–50
<u>Ring 21BA</u>				
E8018NM/04P046 Lot D217A27A	34, 36, 37, 39, 40	23, 28, 24, 20, 24	–20[a]	–48
E8018NM/07L669 Lot K004A27A	50, 50, 54	44, 44, 46	+10[a]	–50
RAC01NMM/3P4966[c] Lot 1214/3482	40, 71, 75, 63, 59	41, 63, 68, 58, 53	+10[a]	–30
RAC01NMM/3P4966[b] Lot 1214/3482	65, 70, 67, 69, 49	60, 60, 63, 55, 44	+10[a]	-48
<u>Ring 21BB</u>				
E8018NM/04P046 Lot D217A27A	34, 36, 37, 39, 40	23, 28, 24, 20, 24	–20[a]	–48
E8018NM/07L669 Lot K004A27A	50, 50, 54	44, 44, 46	+10[a]	–50
E8018NM/C3L46C Lot J020827A	35, 39, 40	34, 39, 39	+10[a]	–20

Table 5.3-5

Weld Material (Continued)

Type/Heat/Lot/Control	Charpy Impact (ft-lb)	Charpy Expansion MLE	Charpy Test Temperature (°F)	RT_{NDT} (°F)
RAC01NMM/3P4966[c] Lot 1214/3482	40, 71, 75, 63, 59	41, 63, 68, 58, 53	+10[a]	–30
RAC01NMM/3P4966[b] Lot 1214/3482	65, 70, 67, 69, 49	60, 60, 63, 55, 44	+10[a]	–48
E8018NM/08M365 Lot G128A27A	49, 50, 51	38, 40, 43	+10[a]	–48
Ring 21BC				
E8018NM/09L853 Lot A111A27A	78, 78, 79	60, 62, 62	+10[a]	–50
E8018NM/C3L46C Lot J020A27A	35, 39, 40	34, 39, 39	+10[a]	–20
RAC01NMM/3P4966[c] Lot 1214/3482	40, 71, 75, 63, 59	41, 63, 68, 58, 53	+10[a]	–30
RAC01NMM/3P4966[b] Lot 1214/3482	65, 70, 67, 69, 49	60, 60, 63, 55, 44	+10[a]	–48
Ring 21BD				
E8018NM/C3L46C Lot J020A27A	35, 39, 40	34, 39, 39	+10[a]	–20
RAC01NMM/3P4966[c] Lot 1214/3482	40, 71, 75, 63, 59	41, 63, 68, 58, 53	+10[a]	–30
RAC01NMM/3P4966[b] Lot 1214/3482	65, 70, 67, 69, 49	60, 60, 63, 55, 44	+10[a]	–48
E8018NM/04P046 Lot D217A27A	34, 36, 37, 39, 40	23, 28, 24, 20, 24	–20[a]	–48
Ring 22BE				
RAC01NMM/3P4966[c] Lot 1214/3481	39, 38, 38, 82, 84	68, 64, 63, 81, 72	+10	–20
RAC01NMM/3P4966[b] Lot 1214/3481	28, 84, 63, 75, 78	18, 62, 57, 51, 57	+10	–6

Table 5.3-5

Weld Material (Continued)

Type/Heat/Lot/Control	Charpy Impact (ft-lb)	Charpy Expansion MLE	Charpy Test Temperature (°F)	RT_{NDT} (°F)
Ring 22BF				
E8018NM/04P046 Lot D217A27A	34, 36, 37, 39, 40	23, 28, 24, 20, 24	–20[a]	–48
E8018NM/05P018 Lot D211A27A	29, 30, 31, 36, 38	26, 26, 31, 33, 35	–20[a]	–38
RAC01NMM/3P4966[c] Lot 1214/3481	39, 38, 38, 82, 84	68, 64, 63, 81, 72	+10	–20
RAC01NMM/3P4966[b] Lot 1214/3481	28, 84, 63, 75, 78	18, 62, 57, 51, 57	+10	–6
Ring 22BG				
E8018NM/624063 Lot C228A27A	37, 40, 51, 57, 70	33, 34, 41, 47, 55	–20[a]	–50
E8018NM/624039 Lot D224A27A	28, 33, 34, 36, 42	29, 32, 33, 34, 42	–20[a]	–36
RAC01NMM/3P4966[c] Lot 1214/3481	39, 38, 38, 82, 84	68, 64, 63, 81, 72	+10	–20
RAC01NMM/3P4966[b] Lot 1214/3481	28, 84, 63, 75, 78	18, 62, 57, 51, 57	+10	–6
Ring 22BH				
E8018NM/04P046 Lot D217A27A	34, 36, 37, 39, 40	23, 28, 24, 20, 24	–20[a]	–48
E8018NM/624039 Lot D205A27A	41, 44, 49, 54, 58	32, 36, 40, 41, 45	–20[a]	–50
RAC01NMM/3P4966[c] Lot 1214/3481	39, 38, 38, 82, 84	68, 64, 63, 81, 72	+10	–20
RAC01NMM/3P4966[b] Lot 1214/3481	28, 84, 63, 75, 78	18, 62, 57, 51, 57	+10	–6

[a] Drop weight NDT not applicable.
[b] Tandem wire process.
[c] Single wire process.

Table 5.3-6

Vessel Beltline Plate

Plate	P	Cu	C	Mn	Si	S	Ni	Mo	V
C1272-1	0.013	0.15	0.23	1.31	0.26	0.02	0.60	0.55	--
C1272-2	0.013	0.15	0.23	1.31	0.26	0.02	0.60	0.55	--
C1273-1	0.014	0.14	0.23	1.28	0.23	0.018	0.60	0.57	--
C1273-2	0.014	0.14	0.23	1.28	0.23	0.018	0.60	0.57	--
B5301-1	0.017	0.13	0.20	1.34	0.23	0.014	0.50	0.52	--
C1336-1	0.017	0.13	0.21	1.36	0.22	0.013	0.50	0.49	--
C1337-1	0.018	0.15	0.22	1.32	0.21	0.013	0.51	0.50	--
C1337-2	0.018	0.15	0.22	1.32	0.21	0.013	0.51	0.50	--

Peak I.D. EOL (33.1 EFPY) fluence = 7.41 x 10^{17} n/cm^2.

Table 5.3-7

Vessel Beltline Weld Material Chemistry[a]

Weld Heat/Control	Cu	C	Mn	Si	S	Ni	Mo	V	P
492L4871[b]	0.03	0.07	1.17	0.32	0.02	0.98	0.51	0.02	0.02
5P6756/0342[c]	0.08[f]	0.063	1.27	0.57	0.011	0.936[f]	0.45	0.006	0.01
5P6756/0342[d]	0.08[f]	0.078	1.24	0.53	0.012	0.936[f]	0.46	0.006	0.01
3P4955/0342[d]	0.027[f]	0.035	1.33	0.56	0.011	0.921[f]	0.52	0.006	0.016
3P4955/0342[c]	0.027[f]	0.054	1.28	0.55	0.010	0.921[f]	0.54	0.007	0.016
04T931[b]	0.03	0.05	1.03	0.28	0.024	1.00	0.53	0.01	0.02
04P046[b]	0.06	0.044	1.04	0.40	0.021	0.90	0.58	0.02	0.009
07L996[b]	0.03	0.05	1.24	0.48	0.016	1.02	0.54	--	0.014
3P4966/3481[d]	0.025[f]	0.074	1.38	0.36	0.013	0.913[f]	0.49	0.006	0.010
3P4966/3481[c]	0.025[f]	0.067	1.39	0.38	0.014	0.913[f]	0.53	0.008	0.011
3P4966/3482[c]	0.025[f]	0.059	1.35	0.38	0.013	0.913[f]	0.50	0.005	0.013
3P4966/3482[d]	0.025[f]	0.077	1.42	0.41	0.013	0.913[f]	0.53	0.005	0.014
CL46C[b]	0.02	0.063	0.96	0.32	0.017	0.87	0.53	--	0.019
08M365[b]	0.02	0.057	1.23	0.47	0.023	1.10	0.57	--	0.02
09L853[b]	0.03	0.052	1.23	0.46	0.023	0.86	0.51	--	0.018
05P018[b]	0.09	0.057	1.21	0.44	0.021	0.90	0.53	0.01	0.008
624063[b]	0.03	0.041	1.12	0.41	0.018	1.00	0.54	0.01	0.009
624039[b,e]	0.07	0.060	1.11	0.45	0.025	1.01	0.57	0.02	0.015
624039[b,e]	0.10	0.041	1.12	0.45	0.02	0.92	0.53	0.01	0.01

[a] As deposited.
[b] M = Manual Welding Process
[c] S = Single Wire Process
[d] T = Tandem Wire Process
[e] Different lot numbers
[f] GE Nuclear Energy, "Pressure-Temperature Curves for Energy Northwest Columbia," NEDC-33144-P (CVI CAL 1012-00,3), Table 4-6b.

LDCN-06-000

Table 5.3-8

10 CFR 50 Appendix H Matrix

Appendix H Paragraph	Topic	Comply Yes/No or N/A	Alternative Actions or Comments
I	Introduction	N/A	
II.A	Fluence 10^{17}n/cm^2	Yes	CGS Plant-specific RPV Surveillance Program is replaced by the BWRVIP ISP. See Section 5.3.1.6.
II.B	Standards Requirements (ASTM) for Surveillance	No	Plant-specific Surveillance Program: Noncompliance with ASTM E185-73 in that the surveillance specimens are not necessarily from the limiting beltline material. Specimens are from actual beltline material, however, and can be used to predict behavior of the limiting material. Heat and heat/lot numbers for surveillance specimens were supplied. See Section 5.3.1.6.
II.C.1	Surveillance Specimen Shall be Taken for Locations Alongside the Fracture Test Specimens (Section III.B of Appendix G)	No	Plant-specific Surveillance Program: Noncompliance in that specimens may not have necessarily been taken from alongside specimens required by Section III of Appendix G and transverse CVNs may not be employed. However, representative materials have been used, and RT_{NDT} shift appears to be independent of specimen orientation. See Section 5.3.1.6.
II.C.2	Locations of Surveillance Capsules in RPV	Yes	Code basis is used for attachment of brackets to vessel cladding.
II.C.3.a	Withdrawal Schedule of Capsules, $RT_{NDT} < 100°F$	N/A	See Section 5.3.1.6. Starting RT_{NDT} of limiting material is based on alternative action (see paragraph III.A of Appendix G).
II.C.3.b	Withdrawal Schedule of Capsules, $RT_{NDT} < 200°F$	N/A	
II.C.3.c	Withdrawal Schedule of Capsules, $RT_{NDT} > 200°F$	N/A	

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Table 5.3-8

10 CFR 50 Appendix H Matrix (Continued)

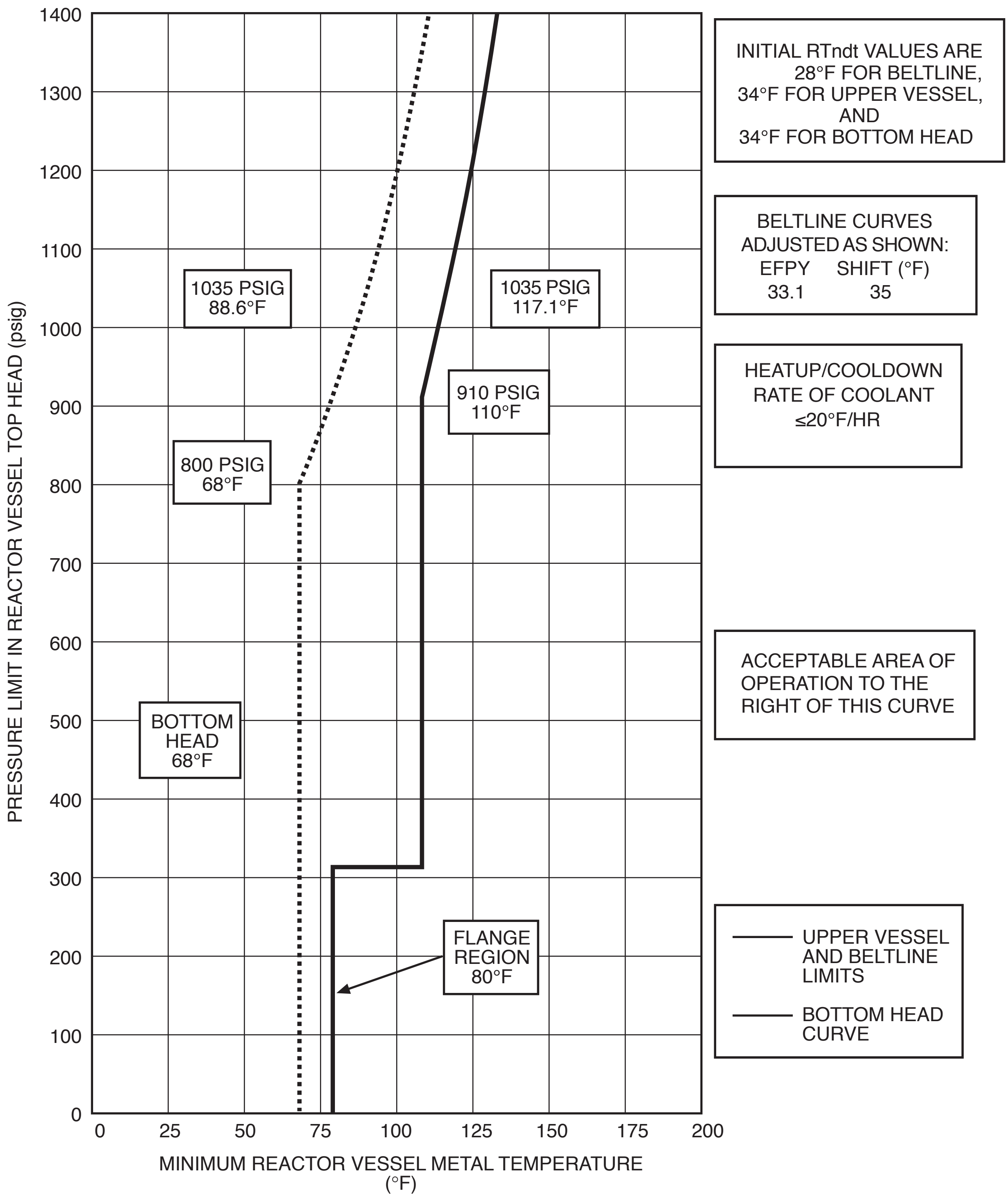
Appendix H Paragraph	Topic	Comply Yes/No or N/A	Alternative Actions or Comments
III.A	Fracture Toughness Testing Requirements of Specimens	Yes	Requirements for postirradiation testing of surveillance material are addressed in the BWRVIP ISP implementation plan (Reference 5.3.4-2).
III.B	Method of Determining Adjusted Reference Temperature for Base Metal, HAZ, and Weld Metal	Yes	Method of determining adjusted reference temperatures found in the BWRVIP ISP implementation plan (Reference 5.3.4-2).
IV.A	Reporting Requirements of Test Results	Yes	Reporting requirements are discussed in the BWRVIP ISP implementation plan (Reference 5.3.4-2).
IV.B	Requirement for Dosimetry Measurement	Yes	Dosimetry requirements are discussed in the BWRVIP ISP implementation plan (Reference 5.3.4-2).
IV.C	Reporting Requirements of Pressure/Temperature Limits	Yes	A discussion of the pressure/temperature limits and reporting requirements is found in the BWRVIP implementation plan (Reference 5.3.4-2).

Table 5.3-9

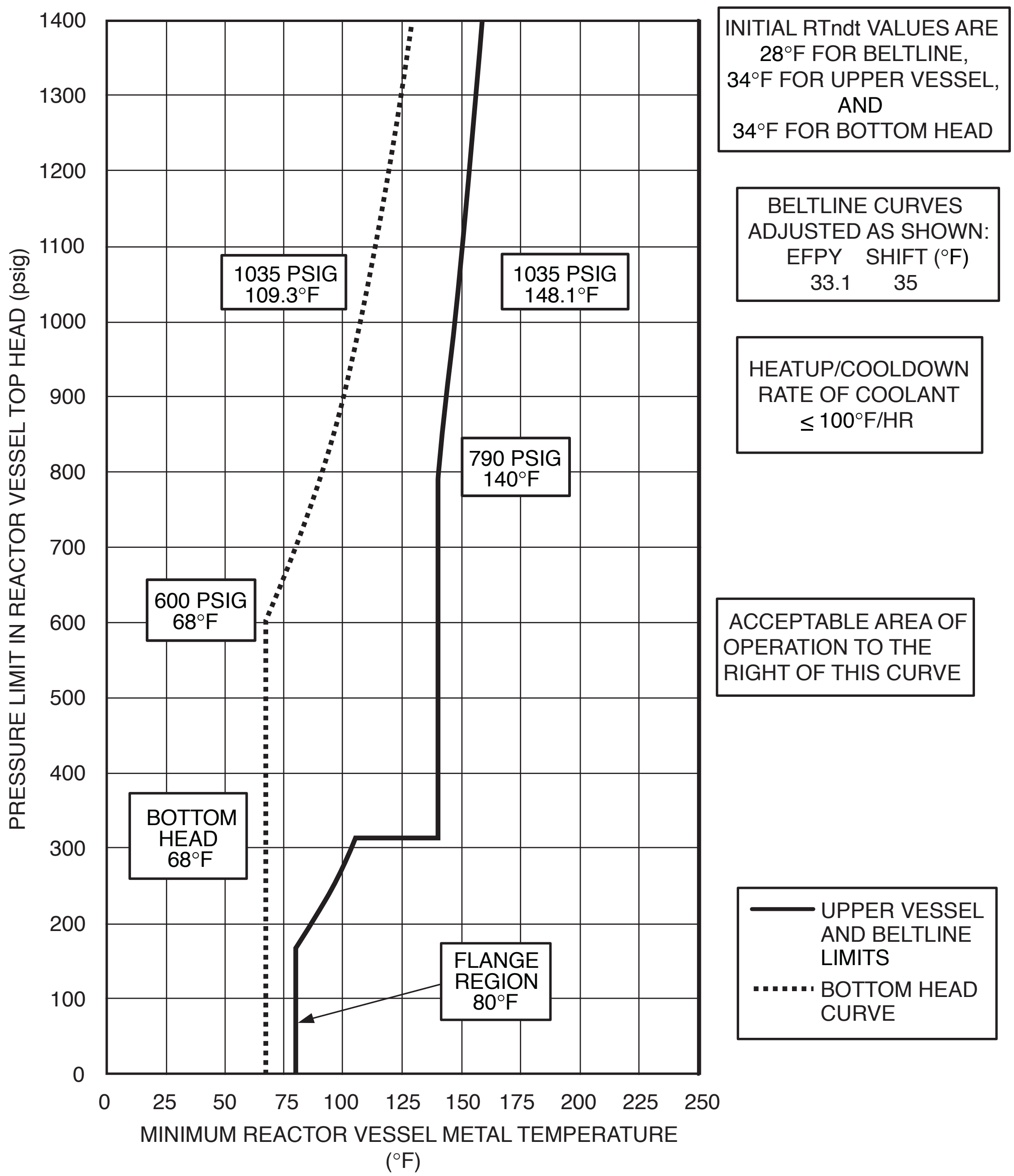
Reactor Vessel Beltline Minimum
Wall Thickness and Diameter

Inside diameter with clad	= 251 in. (minimum)
Wall thickness (ring #22, lower intermediate shell)	= 6.188 in. (minimum)
Wall thickness (ring #21, lower shell)	= 9.5 in. (minimum)
Clad thickness	= 0.1875 in. (nominal)
	= 0.125 in. (minimum)

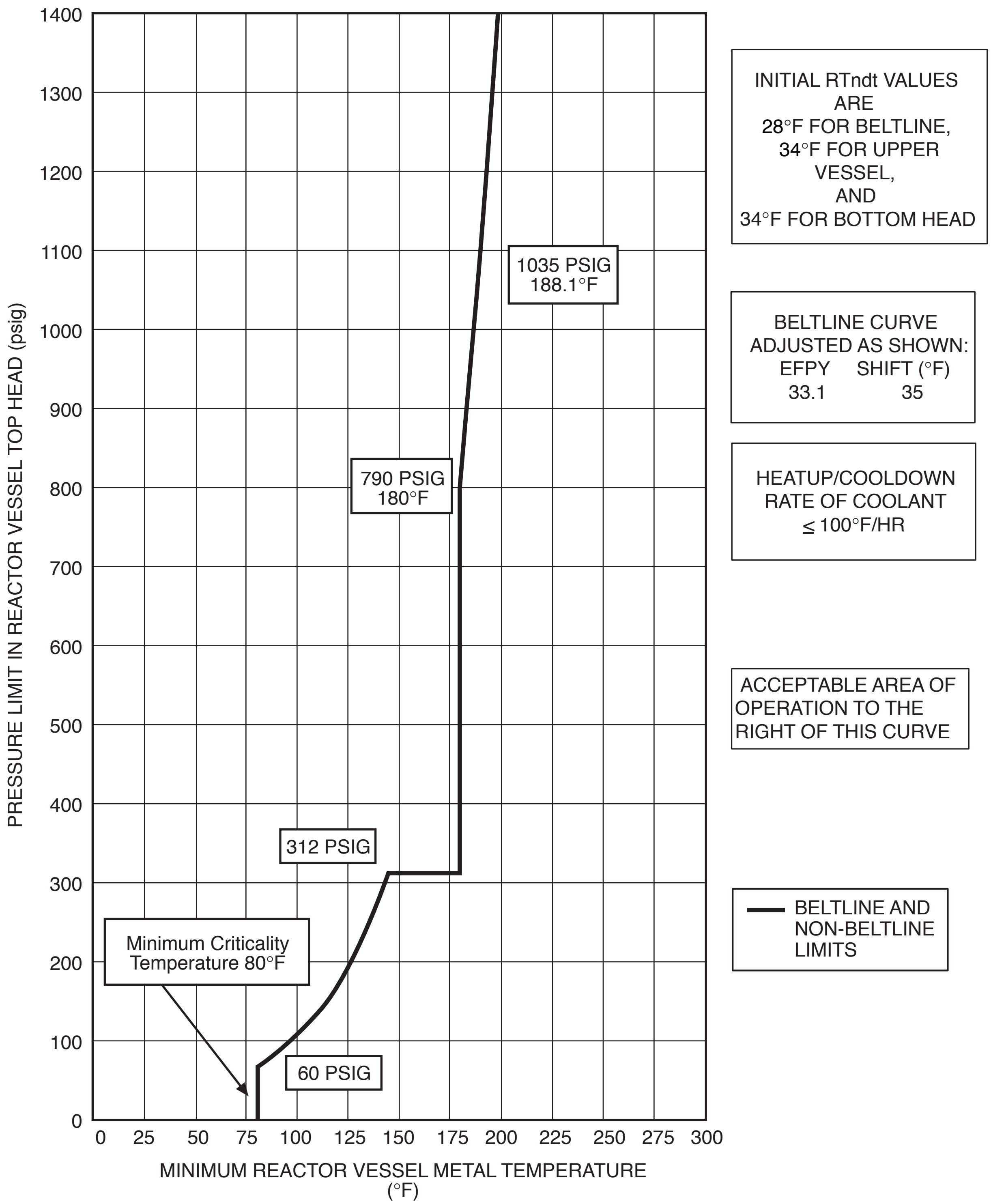
Refer to Figure 5.3-2 and CVI 02B13-06,2 Rev. 8 (VPF #3133-001-9) CBI Nuclear Company Drawing No. 1, Rev. 8, “Vessel Outline.”



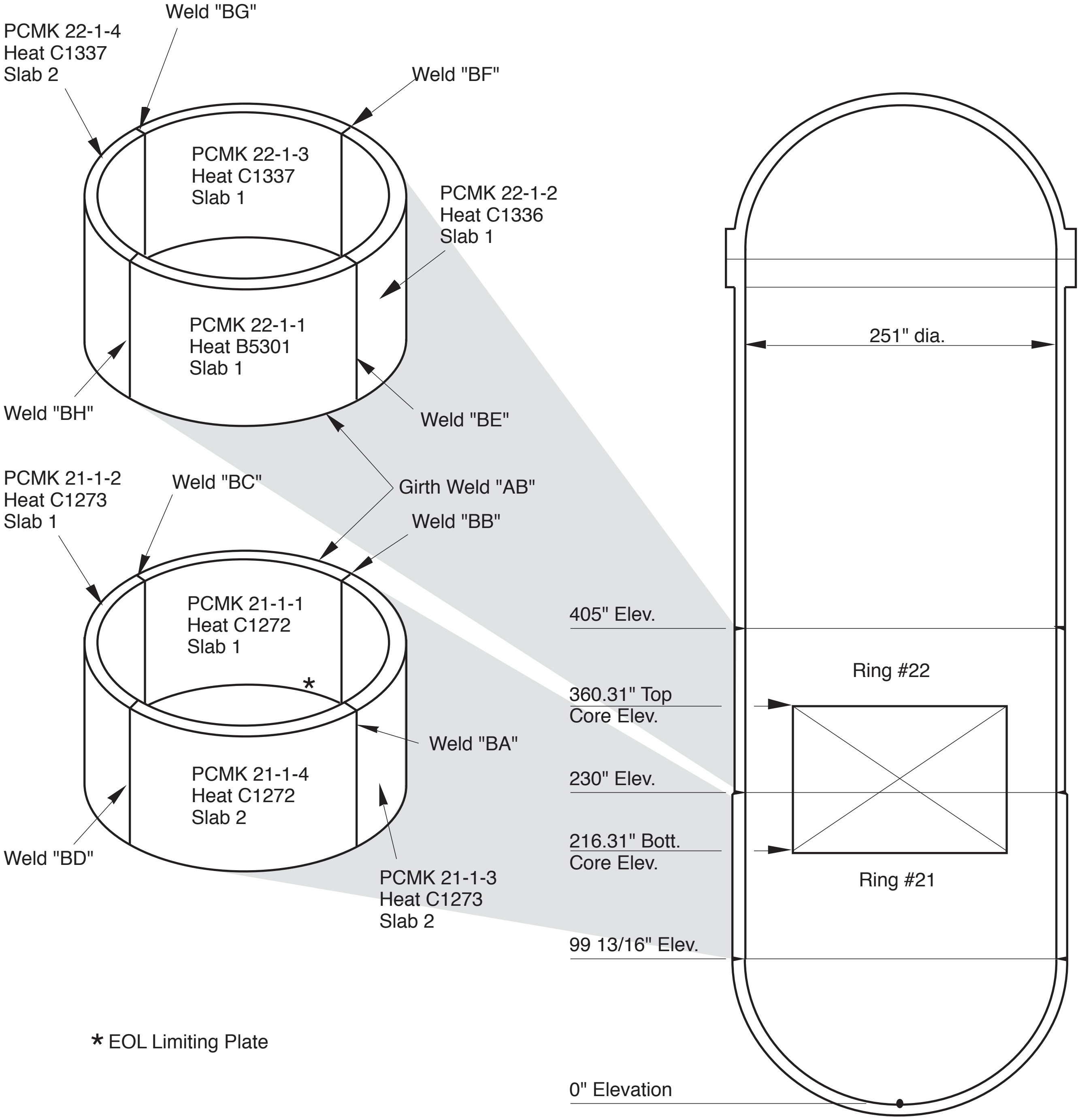
Columbia Generating Station **Final Safety Analysis Report**	**Pressure Temperature Limits** **Testing Curve A** **(Inservice Leak and Hydrostatic Testing Curve)**
	Draw. No. 900547.42 — Rev. — Figure 5.3-1.1



Columbia Generating Station Final Safety Analysis Report	Pressure Temperature Limits Curve B (Non-Nuclear Heating and Cooldown Curve)
	Draw. No. 990578.74 Rev. Figure 5.3-1.2

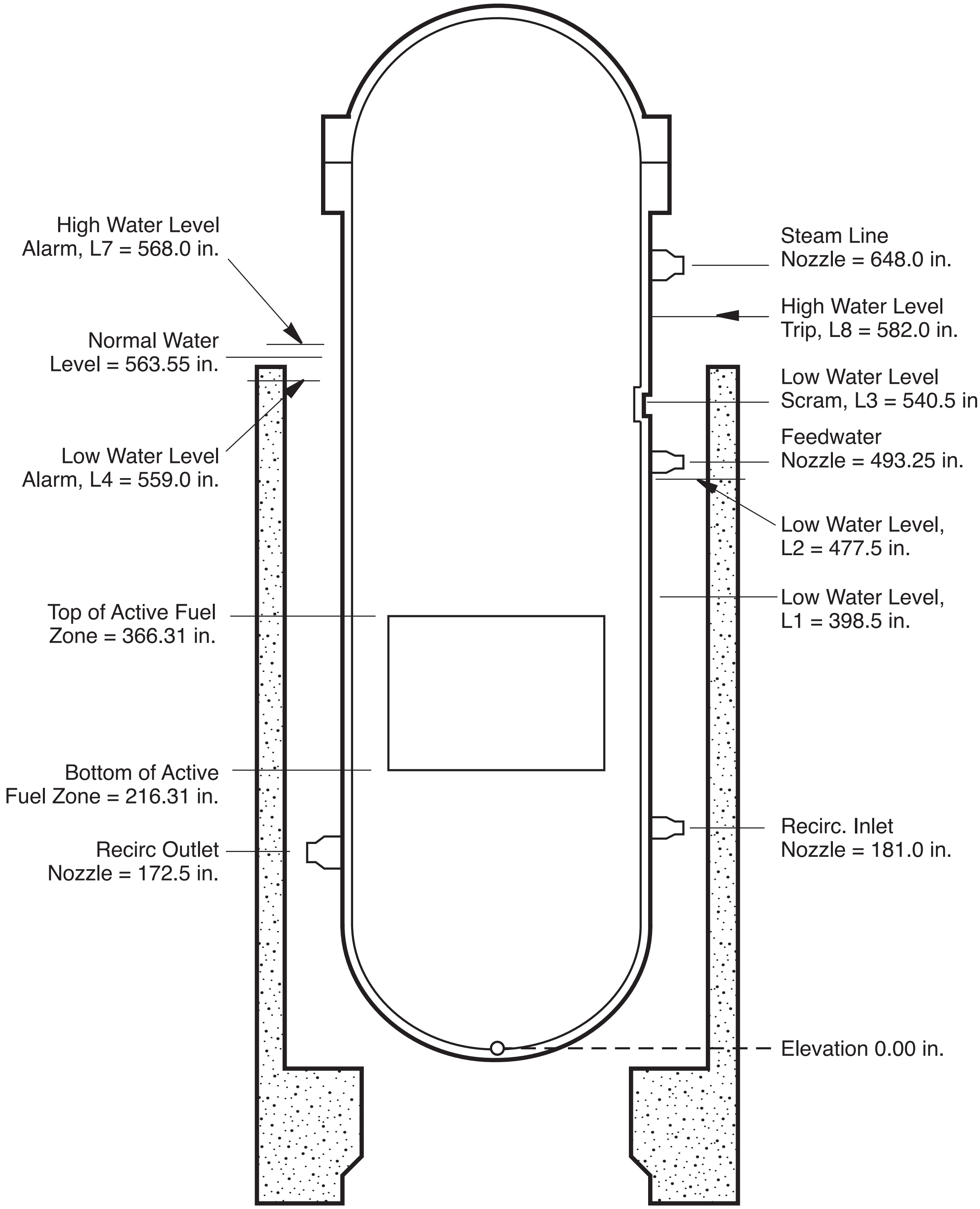


Columbia Generating Station **Final Safety Analysis Report**	**Pressure Temperature Limits** **Curve C** **(Nuclear Heating and Cooldown Curve)**
	Draw. No. 900547.43 · Rev. · Figure 5.3-1.3

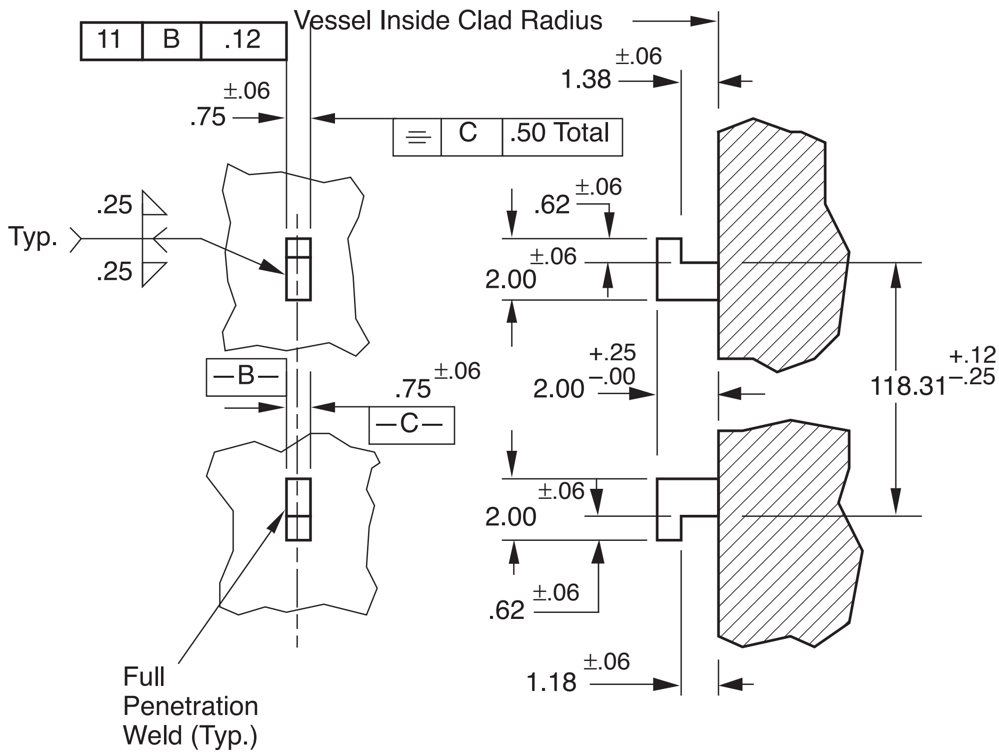


Columbia Generating Station **Final Safety Analysis Report**	**Vessel Beltline Plate and Weld Seam Identification**
	Draw. No. 910402.30 · Rev. · Figure 5.3-2

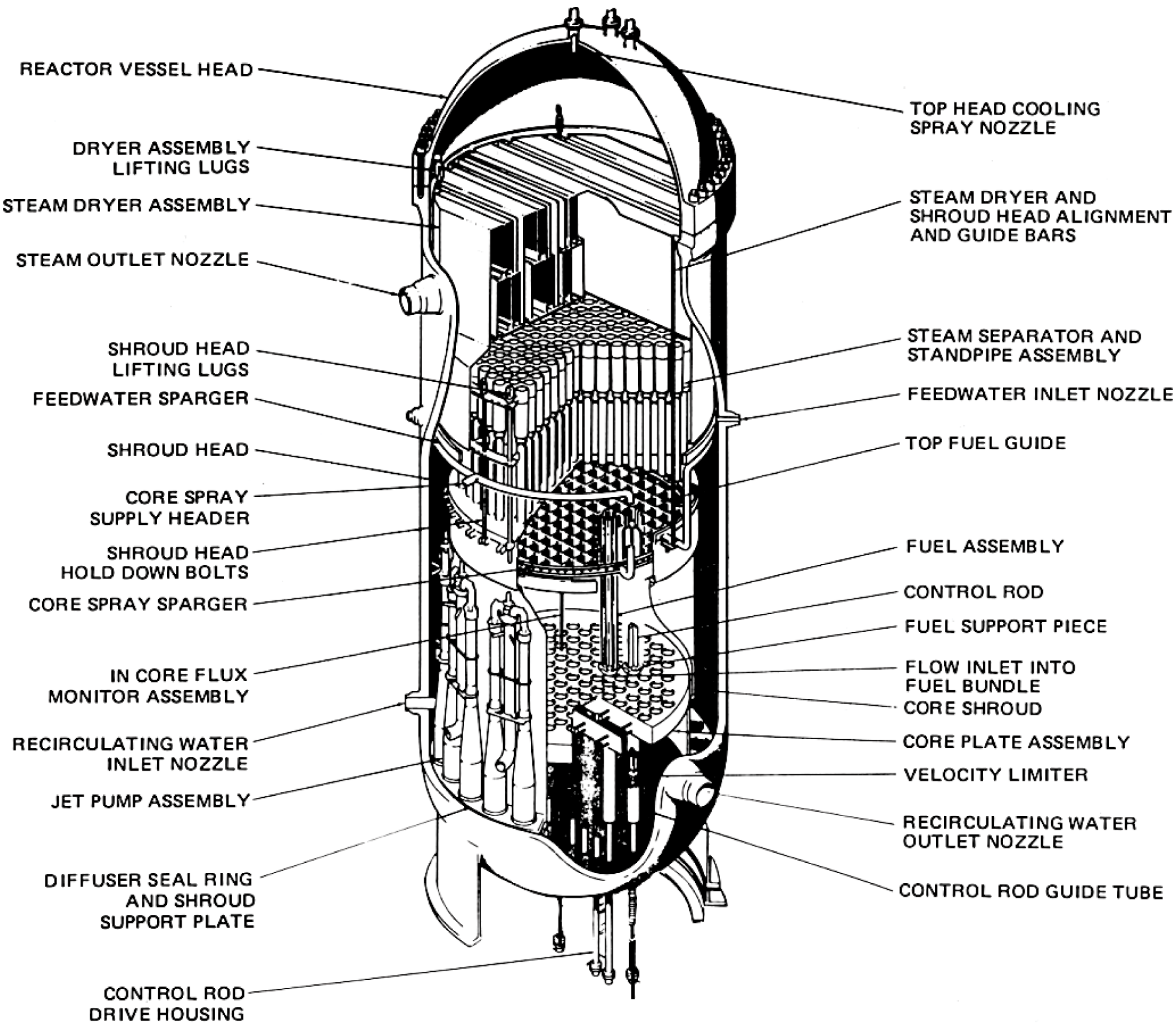
Form No. 960690
LDCN-02-064



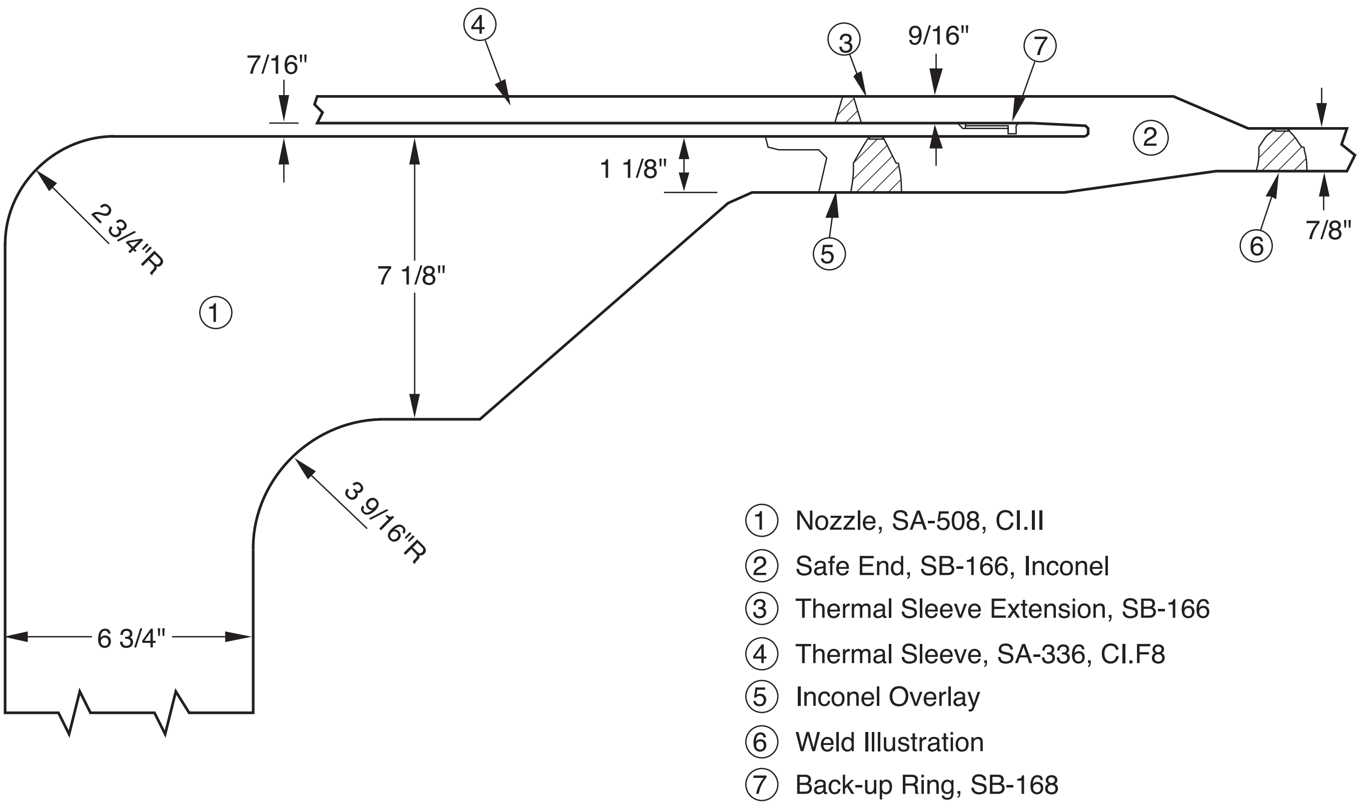
Nominal Reactor Vessel Water Level Trip and Alarm Elevation Settings



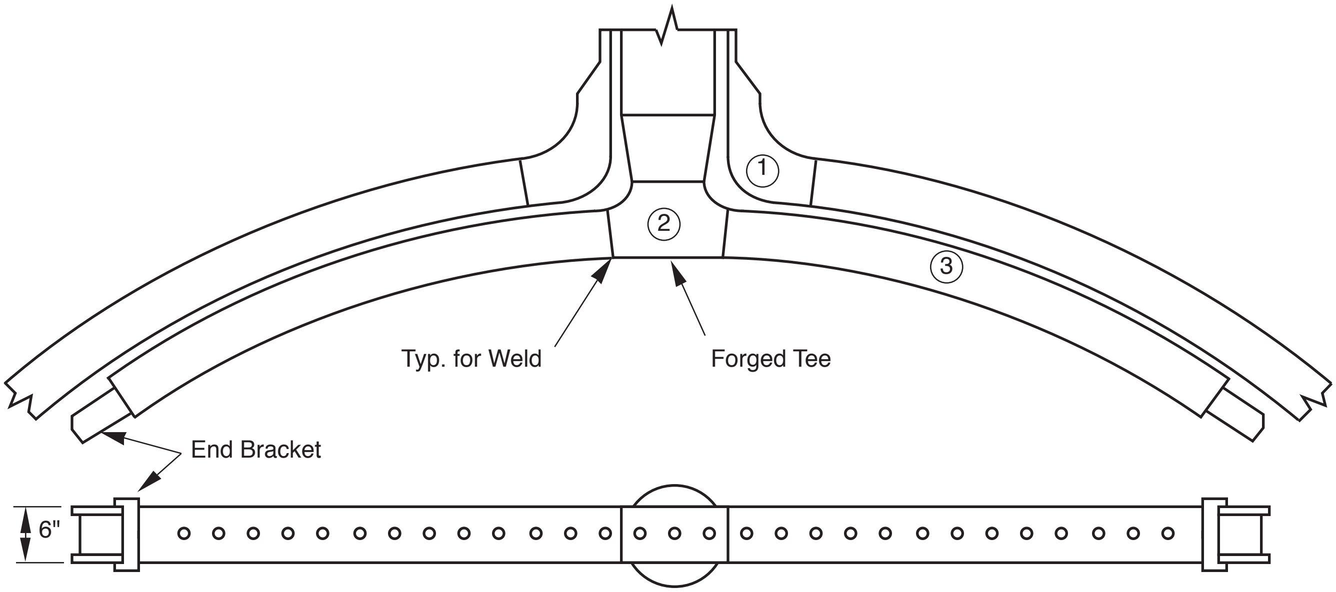
Bracket for Holding Surveillance Capsule



Columbia Generating Station Final Safety Analysis Report	Reactor Vessel		
	Draw. No. 020002.44	Rev.	Figure 5.3-5



Feedwater Nozzle



1 Nozzle, SA-508, Cl.II
2 Forged Tee, 304S.S
3 Sparger Header, 304S.S

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Feedwater Sparger

Draw. No. **960690.57** | Rev. | Figure **5.3-7**

5.4 COMPONENT AND SUBSYSTEM DESIGN

Pumps and valves within the reactor coolant pressure boundary (RCPB) are described in Table 5.4-1.

5.4.1 REACTOR RECIRCULATION PUMPS

5.4.1.1 Safety Design Bases

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

a. An adequate fuel barrier thermal margin shall be ensured during postulated transients,

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

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

5.4.1.2 Power Generation Design Bases

The RRC meets the following power generation design bases:

a. The system shall provide sufficient flow to remove heat from the fuel, and

b. System design shall minimize maintenance situations that would require core disassembly and fuel removal.

5.4.1.3 Description

The RRC consists of the two recirculation pump loops external to the reactor vessel. These loops provide the piping path for the driving flow of water to the reactor vessel jet pumps (see Figure 5.4-1). Each external loop contains one high-capacity variable-speed motor-driven recirculation pump. The motor is powered by an adjustable speed drive (ASD). The external loop also contains 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 reactor vessel internals. Their location and mechanical design are discussed in Section 3.9.5. The important design and performance characteristics of the RRC is shown in Table 5.4-2.

The head, flow, torque, net positive suction head (NPSH), BHP, and efficiency curves are shown in Figures 5.4-2, 5.4-3, and 5.4-4. Instrumentation and control description is provided in Sections 7.6 and 7.7.

The recirculation system piping and normally flooded section of the reactor vessel is periodically coated with a microscopic layer of noble metals. This coating serves to create a catalytic layering of the noble metal platinum to reduce the hydrogen addition injection rate required to achieve a low electrochemical corrosion potential (ECP). The low ECP achieves intergranular stress corrosion cracking (IGSCC) and irradiation assisted stress corrosion cracking (IASCC) protection while minimizing the effects of high dose rates attributed to regular hydrogen injection rates.

The recirculated coolant consists of saturated water from the steam separators and dryers that have 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 provides the driven flow for the jet pumps. This flow enters the jet pump at suction inlets and is accelerated by the driving flow. The flows, both driving and driven, are mixed in the jet pump throat section and result in partial pressure recovery. The balance of recovery is obtained in the jet pump diffusing suction (see Figure 5.4-5). The adequacy of the total flow to the core is discussed in Section 4.4.

The allowable heatup rate for the recirculation pump casing is the same as the reactor vessel. If one loop is shut down, the idle loop can be kept hot by leaving the loop valves open; this permits the reactor pressure plus the active jet pump head to cause reverse flow in the idle loop. When starting the pump in an idle recirculation loop with the other loop in operation, the operating loop flow will be verified to be less than 50% of rated loop flow within 15 minutes prior to pump start.

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

The 20-in. motor-operated gate valves provide pump and flow control valve (FCV) isolation during maintenance. The suction valve is capable of closing with up to 50 psi differential, while the discharge valve can close with up to 400 psi differential. Both valves are remote manually operated.

The FCV is blocked open (seized in the full open position). This condition does not affect the pressure integrity or impact the transient duty cycle of the valve or allow the ball to break away from the shaft.

The required NPSH for the recirculation pumps and jet pumps is supplied by the subcooling provided by the feedwater flow. Accurate temperature detectors are provided in the recirculation lines. Steam dome temperature is provided through pressure conversion. The difference between these two readings is a direct measurement of the subcooling. If the subcooling falls below the time-delayed setpoint 10.7°F, the ASD system is reduced to minimum frequency 15 Hz (25% pump speed) on both of the RRC loops. Each loop has independent instrumentation for cavitation protection.

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

Connections to the piping on the suction and discharge sides of the pumps provide a means to flush and decontaminate the pump and adjacent piping. The piping low point drain, designed for the connection of temporary piping, is used during flushing or decontamination.

Each recirculation pump is driven by an adjustable speed motor and is equipped with a two-stage mechanical seal cartridge. Each of the two seals in the package is subject to one-half the total pressure being sealed. Each seal is structurally capable of sealing full pressure for limited periods of operation. The two seals can be replaced without removing the motor from the pump. 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 temperature and pressure drop across each individual seal can be monitored.

Each recirculation pump motor is a 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 ASD-supplied power to the drive motors so that they are adequately cooled during the transient. This inertia requirement is met without a flywheel.

The ASD can vary the discharge flow of the pump proportionally to a reactor operator remote manually adjusted demand signal. The RRC GE-FANUC digital control scheme is described in Sections 7.6 and 7.7. The recirculation loop flow rate can be varied, within the expected flow range, in response to changes to system demand.

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

The pump drive motor, impeller, and wear rings are designed for as long a life as is practical. Pump mechanical seal parts and the valve packing have life expectancies which afford convenient replacement during the refueling outages.

The ASD system selected to drive the recirculation pump induction motor is a dual channel system. Two ASDs are provided, capable of 11,200 hp at 66 Hz per RRC loop. If one channel fails, the RRC loop flow capability must be reduced to the capability of a single channel ASD. The dual channel ASD system provides for high availability of the ASD system. The ASD system is a solid-state frequency converter with overall high availability. Sections 7.6 and 7.7 provide more detail of the system design.

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

The RRC pressure boundary equipment is designed as Seismic Category I equipment. The pump is assumed to be filled with water for the analysis. Vibration snubbers located at the top of the motor and at the bottom of the pump casing are designed to resist the horizontal reactions.

The recirculation piping, valves, and pumps are supported by hangers to avoid the use of piping expansion loops that would be required if the pumps were anchored. In addition, the recirculation loops are provided with a system of restraints designed so that reaction forces associated with any split or circumferential break do not jeopardize drywell integrity. This restraint system provides adequate clearance for normal thermal expansion movement of the loop. The criteria for the protection against the dynamic effects associated with a loss-of-coolant accident (LOCA) are contained in Section 3.6.

The recirculation system piping, valves, and pump casings are covered with thermal insulation having a total maximum heat transfer rate of 65 Btu/hr-ft^2 with the system at rated operating conditions. This heat loss includes losses through joints, laps, and other openings that may occur in normal application.

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

The residual heat removal (RHR) system can use the recirculation loop jet pumps to provide circulation through the reactor core. Operating restrictions limit RHR operation to regions where jet pump cavitation does not occur.

5.4.1.3.1 Recirculation System Cavitation Consideration

Cavitation Coefficients

The recirculation pump, jet pump, and FCV were tested to determine their cavitation coefficients so that prolonged operation in cavitating regimes can be avoided.

Equipment Damage Provisions

Cavitation interlocks are provided for the recirculation pump and jet pumps; since cavitation produces material damage after long-term operation and the damage potential decreases with an increase in water temperature, short periods of cavitation during a transient or accident are not a concern. However, long-term operation that might occur is of sufficient concern to call for inspections during the next refueling outage. Consequently, to avoid the need for such inspections, automatic interlocks are installed. Class 1E equipment is not necessary for power generation design requirements, so the automatic interlocks are non-Class 1E.

The consequences of cavitation would require inspection of the affected component and repair or replacement if the inspection showed unacceptable damage. Consequently, cavitation could call for increased scheduled outage time for inspection/repair affecting plant availability power generation design goals.

The ASD and its GE-FANUC digital control system is a non-safety-related system. The ASD and control system have alarm and protective systems and are provided with on-line video diagnostic displays at the main control room operating benchboard.

5.4.1.4 Safety Evaluation

Reactor recirculation system malfunctions that pose threats of damage to the fuel barrier are described and evaluated in Section 15.3. It is shown in Section 15.3 that none of the malfunctions result in significant fuel damage. The RRC has sufficient flow coastdown characteristics to maintain fuel thermal margins during abnormal operational transients.

The core flooding capability of a jet pump design plant is discussed in detail in the emergency core cooling system (ECCS) document submitted to the NRC (Reference 5.4-1). The ability to reflood the boiling water reactor (BWR) core to the top of the jet pumps is shown schematically in Figure 5.4-6 and is discussed in Reference 5.4-1.

Piping and pump design pressures for the RRC are based on peak steam pressure in the reactor dome, appropriate pump head allowances, and the elevation head above the lowest point in the recirculation loop. Piping and related equipment pressure parts are chosen in accordance with applicable codes. Use of the listed code design criteria ensures 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 shall not be less than 130% of operating speed. Calculation submittal was required and approved.

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 motor bearings are required to be such that dynamic load capability at rated operating conditions is not exceeded during the safe shutdown earthquake (SSE). Calculation submittal was required of the vendor and has been received and approved by GE.

Pump overspeed occurs during the course of a LOCA due to blowdown through the broken loop's pump. Design studies determined that the overspeed was not sufficient to cause destruction of the motor; consequently no pump overspeed protection provision was made.

A failure modes effects analysis (FMEA) was performed on the block valves. In addition, an analysis was made to determine the effect of block valve closure on recirculation pump coastdown. The analysis postulates that coincident with a recirculation pump trip, the block valves begin to close. It was concluded that any closure time greater than 1 minute will have no effect on coastdown times. The consequences of an inadvertent closure without a coincident pump trip is covered in the FMEA.

5.4.1.5 Inspection and Testing

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

During the preoperational test program, the RRC was hydrostatically tested at 125% reactor vessel design pressure. Preoperational tests on the RRC also included 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 RRC piping and equipment were observed as described in Section 5.4.14.

5.4.2 STEAM GENERATORS (Pressurized Water Reactor)

This is not applicable to BWR plants.

5.4.3 REACTOR COOLANT PIPING

The RCPB piping is discussed in Sections 3.9.3.1 and 5.4.1. The recirculation loops are shown in Figures 5.4-1 and 5.4-7. The design characteristics are presented in Table 5.4-2. Avoidance of stress corrosion cracking is discussed in Section 5.2.3.

5.4.4 MAIN STEAM LINE FLOW RESTRICTORS

5.4.4.1 Safety Design Bases

The main steam line flow restrictors were designed to

a. Limit the rate of vessel blowdown to 200 percent of the normal rated flow in the event of a steam line break outside containment. This limits the reactor depressurization rate to a value which will ensure that the steam dryer and other reactor internal structures remain in place.

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

c. Limit the amount of radiological release outside of the drywell prior to main steam isolation valve (MSIV) closure, and

d. Provide trip signals for MSIV closure.

5.4.4.2 Description

A main steam line flow restrictor (see Figure 5.4-8) is provided for each of the four main steam lines. The restrictor is a complete assembly welded into the main steam line. It is located between the last main steam line safety/relief valve (SRV) and the inboard MSIV.

The restrictor limits the coolant blowdown rate from the reactor vessel in the event a main steam line break occurs outside the containment. The restrictor assembly consists of a venturi-type nozzle insert welded, in accordance with applicable code requirements, into the main steam line. The flow restrictor is designed and fabricated in accordance with the ASME "Fluid Meters," 6th edition, 1977.

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

The ratio of venturi throat diameter to steam line inside diameter of approximately 0.55 results in a maximum pressure differential (unrecovered pressure) of about 10 psig at 100% of rated

flow. This design limits the steam flow in a severed line to less than 200% rated flow, yet it results in negligible increase in steam moisture content during normal operation. The restrictor is also used to measure steam flow to initiate closure of the MSIVs when the steam flow exceeds preselected operational limits.

5.4.4.3 Safety Evaluation

A postulated guillotine break of one of the four main steam lines outside the containment results in mass loss from both ends of the break. The flow from the upstream side is initially limited by the flow restrictor upstream of the inboard isolation valve. Flow from the downstream side is initially limited by the total area of the flow restrictors in the three unbroken lines. Subsequent closure of the MSIVs further limits the flow when the valve area becomes less than the limiter area and finally terminates the mass loss when full closure is reached.

Analysis of the main steam break accident outside containment demonstrates that the radioactive materials released to the environs results in calculated doses that are in compliance with 10 CFR 50.67 and Regulatory Guide 1.183 dose 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 0.10% to 0.20% moisture flowing at velocities approximately 150 ft/sec (steam piping I.D.) to 600 ft/sec (steam restrictor throat). The cast austenitic stainless steel (ASME SA351, or ASTM A351, Type CF8) was selected for the steam flow restrictor material because it has excellent resistance to erosion-corrosion in a high velocity steam atmosphere. The excellent performance of stainless steel in high velocity steam appears to be due to its resistance to corrosion. A protective surface film forms on the stainless steel which prevents any surface attack and this film is not removed by the steam.

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

Surface finish has a minor effect on erosion-corrosion. Experience shows that a machined or a ground surface is sufficiently smooth and that no detrimental erosion will occur.

5.4.4.4 Inspection and Testing

Because the flow restrictor forms a permanent part of the main steam line piping and has no moving components, no testing program is planned. Only very slow erosion will occur with

time, and such a slight enlargement will have no safety significance. Stainless steel resistance to corrosion has been substantiated by turbine inspections at the Dresden Unit 1 facility, which have revealed no noticeable effects from erosion on the stainless steel nozzle partitions. The Dresden inlet velocities are about 300 ft/sec and the exit velocities are 600 to 900 ft/sec. However, calculations show that, even if the erosion rates are as high as 0.004 in. per year, after 40 years of operation the increase in restrictor choked flow rate would not exceed 5%. The impact on calculated accident radiological releases would be minimal.

5.4.5 MAIN STEAM LINE ISOLATION SYSTEM

The MSIV leakage control system has been deactivated.

5.4.5.1 Safety Design Bases

The MSIVs, individually or collectively, shall

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

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

c. Close the main steam line when required despite single failure in either valve or in the associated controls, to provide a high level of reliability for the safety function,

d. Use separate energy sources as the motive force to close independently the redundant isolation valves in the individual steam lines,

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

f. Have capability to close the steam lines, either during or after seismic loadings, to ensure isolation if the nuclear system is breached, and

g. Have capability for testing during normal operating conditions to demonstrate that the valves will function.

5.4.5.2 Description

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

Figure 5.4-9 shows an MSIV. Each is a 26-in. Y-pattern, globe valve. Rated steam flow rate through each valve is 3.85×10^6 lb/hr. The main disc or poppet is attached to the lower end of the stem. Normal steam flow tends to close the valve, and higher inlet pressure tends to hold the valve closed. The bottom end of the valve stem closes a small pressure balancing hole in the poppet. When the hole is open, it acts as a pilot valve to relieve differential pressure forces on the poppet. Valve stem travel is sufficient to give flow areas past the wide open poppet approximately equal to the seat port area. The poppet travels approximately 90% of the valve stem travel to close the main disc and approximately the last 10% of travel to close the pilot hole. The air cylinder can open the poppet with a maximum differential pressure of 200 psi across the isolation valve in a direction that tends to hold the valve closed.

A 45-degree angle permits the inlet and outlet passages to be streamlined; this minimizes pressure drop during normal steam flow and helps prevent debris blockage. The pressure drop at 105% of rated flow is 7 psi maximum. The valve stem penetrates the valve bonnet through a stuffing box that has Grafoil 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 maintain the valve in the closed position if air pressure is not available.

The valve is operated by pneumatic pressure and by the action of compressed springs. The control unit is attached to the air cylinder. This unit contains three types of control valves that open and close the main valve and exercise it at slow speed. Remote manual switches in the control room enable the operator to operate the valves.

Operating air is supplied to the outboard valves from the plant air system and to the inboard valves from the containment instrument system (nitrogen). An air accumulator between the control valve and a check valve provides backup operating air. The outboard MSIVs will close on spring force or air cylinder pressure; the inboard valves require spring force and air pressure to close.

Each valve is designed to accommodate saturated steam at plant operating conditions, with a moisture content of approximately 0.25%, an oxygen content of 30 ppm, and a hydrogen content of 4 ppm. The valves are furnished in conformance with a design pressure and temperature rating in excess of plant operating conditions to accommodate plant overpressure conditions.

In the worst case if the main steam line should rupture downstream of the valve, steam flow would quickly increase to 200% of rated flow. Further increase is prevented by the venturi flow restrictor inside the containment.

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

The design objective for the valve is a minimum of 40-years service at the specified operating conditions. Operating cycles (excluding routine exercise cycles) are estimated to be 100 cycles per year during the first year and 50 cycles per year thereafter.

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

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

The MSIVs are designed to close under accident environmental conditions of 340°F for 1 hr at drywell design pressure. In addition, they are designed to remain closed under the following postaccident environment conditions:

a. 340°F for an additional 2 hr at drywell design pressure of 45 psig maximum,
b. 320°F for an additional 3 hr at 45 psig maximum,
c. 250°F for an additional 24 hr at 25 psig maximum, and
d. 200°F during the next 100 days at 20 psig maximum.

To resist sufficiently the response motion from the SSE, the main steam line valve installations are designed as Seismic Category I equipment. The valve assembly is manufactured to withstand the SSE forces applied at the mass center of the extended mass of the valve operator, assuming the cylinder/spring operator is cantilevered from the valve body and the valve is located in a horizontal run of pipe. The stresses caused by horizontal and vertical seismic forces are assumed to act simultaneously. The stresses in the actuator supports caused by

seismic loads are combined with the stresses caused by other live and dead loads, including the operating loads. The allowable stress for this combination of loads is based on the allowable stress set forth in applicable codes. The parts of the MSIVs that constitute a process fluid pressure boundary are designed, fabricated, inspected, and tested as required by the ASME Code Section III.

5.4.5.3 Safety Evaluation

The analysis of a complete, sudden steam line break outside the containment is described in Chapter 15, "Accident Analyses." 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 pipe lines included, and reactor power level) are exceeded (see Section 7.2.1.1).

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 (Reference 5.4-2).

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

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

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

* Response time for full closure is set prior to plant operation for 3 sec minimum, 5 sec maximum.

c. Each valve was 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 were conducted. Tests included radiographic, liquid penetrant, or magnetic particle examinations of castings, forgings, welds, hardfacings, and bolts; and

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

After the valves were installed in the nuclear system, each valve was tested as discussed in Chapter 14.

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

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

5.4.5.4 Inspection and Testing

The MSIVs can be functionally tested for operability during plant operation and refueling outage. The test provisions are listed below. During refueling outage the MSIVs can be functionally tested, leak tested, and visually inspected.

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

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

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

The leak rate through the pipe line valve seats (pilot and poppet seats) can be measured accurately during shutdown by the procedure described in the following:

a. With the reactor at approximately 125°F and normal water level and decay heat being removed by the RHR system in the shutdown cooling mode, all MSIVs are closed utilizing both spring force and air pressure on the operating cylinder;

b. Air from the instrument air system is introduced between the isolation valves at 25 to 26 psig. A pressure decay test or an air makeup test is used to determine combined inboard and outboard isolation valve seat leakage;

c. If combined inboard and outboard isolation valve seat leakage is above the allowed leakage for a single isolation valve, the outboard isolation valve is then tested for seat leakage;

d. To leak-test the outboard isolation valves, the reactor vessel side of the inboard valves is pressurized to approximately the same pressure as the test pressure between the inboard and outboard valves using nitrogen gas or a hydrostatic head. A pressure decay or makeup leak test is then performed on the area between the isolation valves. This ensures essentially zero leakage through the inboard valves with test results indicating outboard valve seat leakage. The volume between the closed valves is accurately known. Corrections for temperature variation during the test period are made to obtain the required accuracy; and

e. At each refueling outage, the MSIVs are slow closed to verify the stem packing is not too tight. Also, the inboard MSIV containment instrument air (CIA) supply pressure boundary from the accumulator check valve to the actuator is verified to not exceed the allowable leak rate.

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

During prestartup tests following an extensive shutdown, the valves will receive the same pressure boundary leakage or hydro tests (approximately 1000 psi) that are imposed in the primary system.

5.4.6 REACTOR CORE ISOLATION COOLING SYSTEM

5.4.6.1 Design Bases

The reactor core isolation cooling (RCIC) system initiates the discharge of a specified constant flow into the reactor vessel over a specified pressure range within a 30-sec time interval. The RCIC water discharge into the reactor vessel varies between a temperature of 40°F up to and

including a temperature of 140°F. The mixture of the RCIC water and the hot steam does the following:

a. Quenches the steam,

b. Removes reactor residual heat by reducing the heat level (enthalpy) due to the temperature differential between the steam and water, and

c. Replenishes reactor vessel inventory.

The RCIC system uses an electrical power source of high reliability, which permits operation with either onsite power or offsite power.

The steam supply to the RCIC turbine is automatically isolated on detection of abnormal conditions in the RCIC system or in RCIC equipment areas. See Section 7.4.1.1.2.

The RCIC system is not an ECCS nor an engineered safety feature (ESF) system and no credit (simulation) is taken in the accident analysis of Chapter 6 or 15 for its operation. However, the system is designed to initiate during plant transients that cause low reactor water level. The design bases with respect to General Design Criteria 34, 55, 56, and 57 are provided in Chapter 3. Reactor core isolation cooling containment isolation valve arrangements are described in Section 6.2.

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

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

The RCIC system is physically located in a different quadrant of the reactor building and uses different divisional power (and separate electrical routings) than the HPCS system. The system operates for the time intervals and the environmental conditions specified in Section 3.11.

5.4.6.2 System Design

5.4.6.2.1 General

5.4.6.2.1.1 Description. The RCIC system consists of a turbine, pump, piping, valves, accessories, and instrumentation designed to ensure that sufficient reactor water inventory is maintained in the reactor vessel to permit adequate core cooling. This prevents reactor fuel overheating should the vessel be isolated and accompanied by loss-of-coolant flow from the reactor feedwater system.

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

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

On reaching a predetermined low level, the RCIC system is initiated automatically. The RCIC turbine is driven with a portion of the decay heat steam from the reactor and exhausts to the suppression pool. The turbine-driven pump takes suction from the condensate storage tank (CST) during normal modes of operation and injects into the reactor vessel. Condensate storage tank freeze protection is discussed in Section 9.2.6. Since the CST is a covered tank, the water supply is not affected by dust storms. If the water supply from the CST becomes exhausted there is an automatic switchover to the suppression pool as the water source for the RCIC pump. This automatic switchover feature for RCIC consists of two Class 1E level switches mounted on a standpipe in the pump suction line. This standpipe is located on the condensate supply line inside the reactor building at the reactor building/service building interface.

The standpipe is open ended and is used to indicate either a low water level condition in the CST or a loss-of-suction supply from the CST. The standpipe is designed, fabricated, and installed to Seismic Category I, Quality Class I, and ASME Section III, Class 2 standards.

The piping from the reactor building/service building interface to the RCIC system is Seismic Category I; each circumferential buttweld has been radiographically examined per ASME Section III, NC-5230, and a chemical analysis has been performed on all piping materials and as-deposited weld materials.

The inline suction reserve from the CST has sufficient volume to maintain the minimum required NPSH for the RCIC pump plus an approximate four-ft margin while the switchover

occurs, thus ensuring a water supply for continuous operation of the RCIC system. The CST switchover level of 448 ft 3 in. provides an additional submergence of 2 ft (above the top of the CST outlet pipe), which is more than adequate to preclude vortex formation in the CST since less than 6 in. of additional submergence for vortex prevention is required for RCIC.

The available NPSH for worst-case operating conditions (i.e., 625 gpm rated flow, maximum water temperature) was calculated for the RCIC pump suction from the suppression pool and the CST. Using the conservative water temperature of 140°F, the NPSH available from the suppression pool is approximately 60 ft. For the CST, using 100°F water, the NPSH available is 48 ft. In both cases, the NPSH available is greater than the required NPSH of 20 ft indicated in Figure 5.4-10 for the RCIC turbine high speed setpoint of 4500 rpm.

The RCIC suction line from the suppression pool has also been evaluated for vortex formation. The RCIC system has adequate NPSH and will not vortex under the conditions it would be expected to operate.

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

The RCIC turbine discharges into a 10-in. exhaust pipe (see Figure 5.4-11), which has been installed as a sparger to prevent flow-induced oscillations due to steam bubble formation and collapse in the suppression pool. Also, a vacuum breaker system has been installed close to the RCIC turbine exhaust line suppression pool penetration to avoid siphoning water from the suppression pool into the exhaust line as steam in the line condenses during and after turbine operation. The vacuum breaker line runs from the suppression pool air volume to the RCIC exhaust line through two normally open motor-operated gate valves and two swing check valves arranged to allow air flow into the exhaust line and to preclude steam flow to the suppression pool air volume. Condensate buildup in the turbine exhaust line is removed by a drain pot in the low point of the line near the turbine exhaust connection. The condensate collected in the drain pot drains to the barometric condenser.

5.4.6.2.1.2 Diagrams. The following diagrams are included for the RCIC systems:

a. A schematic "Piping and Instrumentation Diagram" (Figures 5.4-11) shows all components, piping, points where interface system and subsystems tie together and instrumentation and controls associated with subsystem and component actuation,

b. A schematic "Process Diagram" (Figure 5.4-12) shows temperature, pressures, and flows for RCIC operation and system process data hydraulic requirements, and

c. RCIC turbine and pump performance curves; Constant Pump Flow Figure 5.4-10 and Constant Pump Speed Figure 5.4-13.

5.4.6.2.1.3 Interlocks. The following defines the various electrical interlocks:

a. There are four key-locked valves, RCIC-V-63 (F063), RCIC-V-8 (F008), RCIC-V-68 (F068), and RCIC-V-69 (F069), and two key-locked resets, the "isolation resets;"

b. RCIC-V-31 (F031) limit switch activates when fully open and closes RCIC-V-10 (F010), RCIC-V-22 (F022), and RCIC-V-59 (F059);

c. RCIC-V-68 (F068) limit switch activates when fully open and clears RCIC-V-45 (F045) permissive so RCIC-V-45 (F045) can open;

d. RCIC-V-45 (F045) limit switch activates when RCIC-V-45 (F045) is not fully closed and energizes 15-sec time delay for low pump suction pressure trip and also initiates startup ramp function. This ramp resets each time RCIC-V-45 (F045) is closed;

e. RCIC-V-45 (F045) limits switch activates when fully closed and permits RCIC-V-4 (F004), RCIC-V-5 (F005), RCIC-V-25 (F025), and RCIC-V-26 (F026) to open and closes RCIC-V-13 (F013), RCIC-V-46 (F046) and RCIC-V-19 (F019). RCIC-V-13 (F013) and RCIC-V-46 (F046) auto open on initiation signal if RCIC-V-45 (F045) and RCIC-V-1 (F001) are open;

f. The turbine trip throttle valve RCIC-V-1 limit switch activates when fully closed and closes RCIC-V-13 (F013), RCIC-V-46 (F046) and RCIC-V-19 (F019);

g. The combined pressure switches at reactor low pressure and high drywell pressure when activated closes RCIC-V-110 and 113 (F080 and F086);

h. RCIC high turbine exhaust pressure, low pump suction pressure, low discharge header pressure, or an isolation signal actuates and closes the turbine trip throttle valve. When signal is cleared, the trip throttle valve must be reset from control room;

i. 125% overspeed trips both the mechanical trip at the turbine and the trip throttle valve. The former is reset at the turbine and then the later is reset in the control room;

j. Valves RCIC-V-8 (F008), RCIC-V-63 (F063), and RCIC-V-76 (F076) automatically isolate on low reactor pressure, high turbine exhaust line pressure, high ambient temperature in RCIC equipment areas (leak detection) and high turbine steam supply flow rate (>300% - break detection). A setpoint of 300% for break isolation provides sufficient operating margin to prevent inadvertent isolations due to startup transients and yet is low enough to detect large pipe breaks. Small breaks are detected by the leak detection system. Steam condensing supply valve RCIC-V-64 (F064) has been lock closed as a part of the steam condensing mode deactivation. Note, the key-locked switches for RCIC-V-8 (F008) and RCIC-V-63 (F063) do not prevent automatic isolation of these valves. The key-locked switches are provided to prevent inadvertent manual isolation of the RCIC steam supply during system operation;

k. An initiation signal opens RCIC-V-10 (F010) if closed, RCIC-V-45 (F045), and RCIC-V-46 (F046) if RCIC-V-1 and RCIC-V-45 (F045) are not closed. The initiation signal also starts barometric condenser vacuum pump; and closes RCIC-V-22 (F022) and RCIC-V-59 (F059) if open;

l. The combined signal of low flow plus high discharge pressure opens and with increased flow closes RCIC-V-19 (F019). Also see items e and f above;

m. The signal of in-line reserve tank low water level opens valve RCIC-V-31 (F031);

n. High reactor water level closes RCIC-V-45 (F045); and

o. Main turbine trips if RCIC-V-13 and RCIC-V-45 are open.

5.4.6.2.2 Equipment and Component Description

5.4.6.2.2.1 Design Conditions. Operating parameters for the components of the RCIC systems defined in the following are shown in Figure 5.4-12.

a. One 100% capacity turbine and accessories,

b. One 100% capacity pump assembly and accessories, and

c. Piping, valves, and instrumentation for

  1. Steam supply to the turbine,

  2. Turbine exhaust to the suppression pool,

3. Makeup supply from the CST to the pump suction,

4. Makeup supply from the suppression pool to the pump suction, and

5. 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 accessory equipment.

5.4.6.2.2.2 Design Parameters. Design parameter for the RCIC system components are listed below. See Figure 5.4-11 for cross reference of component numbers listed below:

a. RCIC pump operation RCIC-P-1 (C001) (Reference to Figures 5.4-11 and 5.4-13)

Flow rate	Injection flow - 600 gpm Lube oil cooling water flow - 16-25 gpm Total pump discharge - 625 gpm (includes no margin for pump wear)
Water temperature range	40°F to 140°F
NPSH	21 ft minimum
Developed head	3016 ft @ 1225 psia reactor pressure 610 ft @ 165 psia reactor pressure
BHP, not to exceed	761 HP @ 3016 ft developed head 130 HP @ 610 ft developed head
Design pressure	1500 psia
Design temperature	40°F to 140°F

b.

RCIC turbine operation RCIC-DT-1 (C002)	HP condition	LP condition
Reactor pressure (saturation temperature)	1225 psia	165 psia
Steam inlet pressure	1210 psia	150 psia
Turbine exhaust press	15 to 25 psia	15 to 25 psia
Design inlet pressure	1265 psia + saturated temperature	
Design exhaust pressure	165 psia + saturated temperature	

c. RCIC orifice sizing

Coolant loop orifice RCIC-RO-9 (D009)	Sized with piping arrangement to ensure maximum pressure of 75 psia at the lube oil cooler inlet and a minimum pressure of 45 psia at the spray nozzles at the barometric condenser
Minimum flow orifice RCIC-RO-5 (D005)	Sized with piping arrangement to ensure minimum flow of 100 gpm with RCIC-V-19 (MO-F019) fully open
Test return orifice RCIC-RO-6 (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
Leak-off orifices RCIC-RO-8 and RCIC-RO-10 (D008 and D010)	Sized for 1/8-in. diameter minimum, 3/16-in. diameter maximum
Minimum flow orifice RCIC-RO-11 (D011)	Sized to maintain a minimum flow of 60 gpm through the RCIC water leg pump (RCIC-P-3) while maintaining a positive pressure in the RCIC system at the highest elevation

d. Valve operation requirements

NOTE: Differential pressures listed in the following were obtained from the RCIC system design specification data sheet and are listed for information. Detailed differential pressure requirements are contained in engineering calculations.

Steam supply valve RCIC-V-45 (F045)	Open and/or close against full steam pressure
Pump discharge valve RCIC-V-13 (F013)	Open and/or close against full pump discharge pressure and open in thermal over-pressure conditions in the RCIC discharge header

Pump minimum flow bypass valve RCIC-V-19 (F019)	Open and/or close against full pump discharge pressure
Steam supply isolation valves RCIC-V-08/RCIC-V-63 (F008)	Open and/or close against full differential pressure of 1210 psi
Turbine lube oil/cooling water pressure control valve RCIC-PCV-15 (F015)	Capable of maintaining constant downstream pressure of 75 psia through lube oil cooler
Pump discharge header relief valve (RCIC-RV-3)	1500 psig relief setting; less than 1 gpm required capacity; the maximum allowable discharge is less than 20 gpm
Pump suction relief valve RCIC-RV-17 (F017)	122 psig relief setting; 20 gpm required capacity
Cooling water relief valve (RCIC-RV-19T)	Sized to prevent overpressurization of piping valves and equipment in the turbine lube oil coolant loop in the event of failure of pressure control valve RCIC-PCV-15 (F015). Set pressure is 99 psig; required flow is 33.1 gpm
Pump test return valve RCIC-V-22 (F022)	Qualified to open, close, and throttle against full pump discharge pressure
Pump test return valve RCIC-V-59 (F059)	Qualified to close (not open) against full pump discharge pressure
Relief valve barometric condenser vacuum tank RCIC-RV-33 (F033)	Relief valve is capable of retaining 10 in. of mercury vacuum at 140°F ambient, with a set pressure of 6 psig; required flow is 20 gpm
Pump suction valve suppression pool RCIC-V-31 (F031)	Located as close as practical to the primary containment
Pump suction valve condensate storage tank RCIC-V-10 (F010)	Open and/or close against full suction head from the condensate storage tank

Main pump discharge check valve RCIC-V-65/RCIC-V-66 (F065/F066)	System test mode bypasses this valve. Its functional capability is demonstrated separately
Warm-up line isolation valve RCIC-V-76 (F076)	Valve will open and/or close against full steam pressure
Vacuum breaker isolation valves RCIC-V-110 (F080) and RCIC-V-113 (F086)	Valves will open and/or close against turbine exhaust pressure

e. Rupture disc

Assemblies RCIC-RD-1/RCIC-RD-2 (D001/D002)	Utilized for turbine casing protection, includes a mated vacuum support to prevent rupture disc reversing under vacuum conditions Rupture pressure 150 psig ± 10 psig Flow capacity 60,000 lb/hr @ 165 psig

f. Condensate storage requirements

Total reserve storage for reactor pressure valve makeup is 135,000 gal.

g. 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 design is based on 170°F.

h. Turbine exhaust vertical reaction force

Unbalanced pressure due to opening and discharge under the suppression pool water level is 20 psi.

i. Ambient conditions

	Temperature	Relative Humidity
Normal plant operations	60°F to 100°F	95%
Isolation conditions	148°F	100%

j. Water leg pump

Design pressure	150 psig
Design temperature	212°F
Capacity	25 gpm @ 200 ft total head

k. Barometric condenser

Design pressure	50 psig
Design temperature	650°F

l. Vacuum tank

Design pressure	15 psig
Design temperature	212°F

m. Condensate pump

Design pressure	50 psig
Design temperature	650°F
Capacity	23 gpm @ 10 in. Hg vac., 70°F 50 psig discharge

n. Turbine and steam supply drain pots

Design pressure	1250 psig
Design temperature	575°F

o. Turbine governing and trip throttle valves

Design pressure	1250 psig
Design temperature	575°F

p. Pump suction strainers in the suppression pool

The suction strainers have been procured to the following specifications:

Primary service rating: ANSI 1501-1

Quality Class I

Seismic Category I

Cleanliness Class B

Applicable Code:	Strainer materials and fabrication meets ASME Section III, Class 2 requirements. The "N" stamp is not be applied since the strainers cannot be hydrostatically tested.
Materials:	Strainer body is stainless steel 304 or 316, or engineer approved equal, suitable for submergence in high quality water during a 40-year lifetime.
Quantity:	2
Diameter:	13.5 in.
Length:	5.25 in.
Rated flow:	300 gpm (per strainer)

The strainers are cylindrical, as shown in Figure 5.4-14. Strainer hole diameter is 0.09375 in. Strainers are attached to ANSI 150# RF Flanges.

Head loss is limited to 4 ft of water assuming the strainers are 50% clogged and the water temperature is 220°F.

5.4.6.2.2.3 Overpressure Protection. Referring to Figure 5.4-11, four RCIC pipe lines have a low design pressure and, therefore, require relief devices or some other basis for addressing overpressure protection.

The design pressure of the other major pipe lines is equal to the vessel design pressure and subject to the normal overpressure protection system. In addition, the RCIC discharge header

has a relief valve, RCIC-RV-3, to protect against thermal overpressurization when the system is in standby mode, isolated from the reactor.

Below are the overpressure protection bases for the low pressure piping lines.

a. RCIC pump suction line

A relief valve [RCIC-RV-17 (F017)] is located on the pump suction line in Figure 5.4-11 to accommodate any potential leakage through the isolation valves [RCIC-V-13 (F013) and RCIC-V-66 (F066)]. A high pump suction pressure alarm is provided in the control room.

b. RCIC turbine exhaust line

This line is normally vented to the suppression pool and is not subject to reactor pressure during normal operation. Rupture discs RCIC-RD-1 (D001) and RCIC-RD-2 (D002), as shown in Figure 5.4-11, are installed on this line to prevent exceeding piping design pressure should the exhaust line isolation valve RCIC-V-68 (F068) be closed when the RCIC turbine is operating. The RCIC system will automatically isolate if the rupture discs were to blow open.

c. Portions of the RCIC minimum flow line downstream of RCIC-V-19 (F019)

This line is normally vented to the suppression pool and is separated from reactor pressure by the pump discharge isolation valves [RCIC-V-13, RCIC-V-65, and RCIC-V-66 (F013, F065, and F066)], pump discharge check valve RCIC-V-90, and one additional normally closed isolation valve in the minimum flow line [RCIC-V-19 (F019)] as shown in Figure 5.4-11.

d. Portions of the RCIC cooling water line downstream of RCIC-PCV-15 (F015)

In the standby condition this line is separated from reactor pressure by the pump discharge valves [RCIC-V-13, RCIC-V-65, and RCIC-V-66 (F013, F065 and F066)], pump discharge check valve RCIC-V-90, and one additional normally closed shut-off valve in the cooling water line [RCIC-V-46 (F046)] as shown in Figure 5.4-11. During system operation a relief valve [RCIC-RV-19T (F018)] is provided to prevent overpressurizing piping, valves, and equipment in the coolant loop in the event of failure of pressure control valve RCIC-PCV-15 (F015) as shown in Figure 5.4-11.

5.4.6.2.3 Applicable Codes and Classifications

The RCIC system components within the drywell up to and including the outer isolation valve are designed in accordance with ASME Code Section III, Class 1, Nuclear Power Plant Components. Safety-related portions of the RCIC system are Seismic Category 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 ensure that the RCIC will operate when necessary, the power supply for the system is taken from immediately available energy sources of high reliability. Added assurance is given by 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.

To ensure RCIC availability for the operational events noted previously, the following are considered in the system design.

a. The RCIC and HPCS are located in different quadrants of the reactor building. Piping runs are separated and the water delivered from each system enters the reactor vessel via different nozzles.

b. Prime mover independence is achieved by using a steam turbine to drive the RCIC and an electric motor-driven pump for the HPCS system.

c. The RCIC and HPCS control independence is secured by using different battery systems to provide control power to each system for system operation. Separate detection initiation logic is used for each system.

d. Both systems are designed to meet appropriate safety and quality class requirements. Environment in the equipment rooms is maintained by separate auxiliary systems.

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

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

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

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

f. Periodic inspections and maintenance of the turbine-pump unit are conducted in accordance with manufacturer's instructions. Valve position indication and instrumentation alarms are displayed in the control room.

g. Specific operating procedures relieve the possibility of thermal shock or water hammer to the steam line, valve seals, and discs. Key lock switches are provided for positive administrative control of valve position. Operating procedures require throttling open the outboard isolation valve RCIC-V-8 to remove any condensate trapped between the isolation valves, warming up the steam line by throttling open the warmup valve RCIC-V-76 located on a pipe line bypassing the inboard isolation valve, and then opening the inboard isolation valve RCIC-V-63. All the condensate is removed from the steam supply line by a drain pot located at the lowest point. An alarm sounds when any of these valves leaves the fully open position.

h. Emergency procedures address the operation of RCIC during a station blackout (SBO) event. The RCIC keepfill pump, RCIC-P-3, is powered by a Class 1E ac source, and will be unavailable during an SBO. Upon loss of ac power, the operator manually initiates RCIC. RCIC may be used during an SBO event by maintaining the RCIC discharge header continuously pressurized. The system can be operated in this manner without its keepfill function.

5.4.6.2.5 System Operation

5.4.6.2.5.1 Automatic Operation. Automatic startup or restart (after level 8 shutdown) of the RCIC system due to an initiation signal from reactor low water level requires no operator action. To permit this automatic operation, Technical Specifications operability requirements ensure that all necessary components are available to perform their required functions. In addition, the following are periodically verified:

a. The flow controller has the correct flow setpoint and is in automatic mode;

b. Each RCIC manual, power-operated, and automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, is in the correct position; and

c. The RCIC system piping is filled with water from the pump discharge valve to the injection valve.

The turbine is equipped with a mechanical overspeed trip. The mechanical overspeed trip must be reset out of the control room at the turbine itself. Once the mechanical overspeed trip is reset, the trip throttle valve can be reset.

RCIC System Operation and Shutdown:

During extended periods of operation and when the normal water level is again reached, the HPCS system may be manually tripped and the RCIC system flow controller may be adjusted and switched to manual operation. This prevents unnecessary cycling of the two systems. The RCIC flow to the vessel is controlled by adjusting flow to the amount necessary to maintain vessel level. Subsequent starts of RCIC will occur automatically if the water level decreases to the low level initiation point (Level 2) following a high level shutdown (Level 8). Should RCIC flow be inadequate, HPCS flow will automatically initiate.

RCIC flow may be directed away from the vessel by diverting the pump discharge to the CST. This is accomplished by closing injection valve RCIC-V-13 and opening the test return valves (RCIC-V-22 and 59). The system is returned to injection mode by closing RCIC-V-59 and then opening RCIC-V-13. This mode of operation will not be used during events where an unacceptable source term is identified in primary containment. Diverting RCIC flow to the CST is not a safety-related function nor does it affect the ability of RCIC to initiate during plant transients. The system automatically switches to injection mode if the water level decreases to the low level initiation point (Level 2).

When RCIC operation is no longer required, the RCIC system is manually tripped and returned to standby conditions.

5.4.6.2.5.2 Test Loop Operation. This operating mode (described in Section 5.4.6.2.4) is conducted by manual operation of the system.

5.4.6.2.5.3 Steam Condensing (Hot Standby) Operation. The steam condensing mode of RHR for Columbia Generating Station has been deactivated. However, the major pieces of equipment are installed with the exception of the steam supply relief valves and are shown on the RCIC and RHR piping and instrumentation diagrams (P&IDs) (Figures 5.4-11 and 5.4-15, respectively). Deletion of this mode of operation for RCIC and RHR will not adversely affect either system's capability to bring the reactor to cold shutdown.

5.4.6.2.5.4 Manual Actions. The RCIC system will automatically initiate and inject into the reactor when the reactor water level drops to a low level (L2, -50 in.). No manual actions are required to operate the system. However, control room operators can manually initiate the system prior to reaching the low level.

5.4.6.2.5.5 Reactor Core Isolation Cooling Discharge Line Fill System. See Section 6.3.2.2.5. The description in this section is also applicable to the RCIC line fill system.

5.4.6.3 Performance Evaluation

The RCIC system makeup capacity is sufficient to avoid the need for ECCS for normal shutdowns and shutdowns resulting from anticipated operational occurrences.

5.4.6.4 Preoperational Testing

The preoperational and initial startup test program for the RCIC system is presented in Chapter 14. Regulatory Guide 1.68 compliance is described in Section 1.8.

5.4.6.5 Safety Interfaces

The balance-of-plant/GE nuclear steam supply system safety interfaces for the RCIC system are (a) preferred water supply from the CST, (b) all associated wire, cable, piping, sensors, and valves that lie outside the nuclear steam supply system scope of supply, and (c) air supply for testable check and solenoid-actuated valve(s).

5.4.7 RESIDUAL HEAT REMOVAL SYSTEM

5.4.7.1 Design Bases

The RHR system is comprised of three independent loops. Each loop contains its own motor-driven pump, piping, valves, instrumentation, and controls. Each loop has a suction source from the suppression pool and is capable of discharging water to the reactor vessel via a separate nozzle, or back to the suppression pool via a separate suppression pool return line. In addition, the A and B loops have heat exchangers which are cooled by standby service water. Loops A and B can also take suction from the RRC suction and can discharge into the reactor recirculation discharge or to the suppression pool and drywell spray spargers. Spool-piece interties are available to permit the RHR heat exchangers to be used to supplement the cooling capacity of the fuel pool cooling (FPC) system (see Section 9.1.3 for details). A spool piece intertie was also used to provide a preoperational flushing path for the low-pressure core spray (LPCS). The A and B loops also have connections to the RCIC steam line. However, these are not used because the steam condensing mode has been eliminated.

5.4.7.1.1 Functional Design Basis

The RHR system is designed to restore and maintain the coolant inventory in the reactor vessel and to provide primary system decay heat removal following reactor shutdown for both normal and postaccident conditions. The primary design operating modes associated with performing these functions are briefly described as follows:

a. Low-pressure coolant injection (LPCI) mode - The RHR system automatically initiates into this mode and pumps suppression pool water into separate lines and core flooder nozzles for injection into the core region of the reactor vessel following a LOCA. The system's LPCI mode operates in conjunction with the other ECCS to provide adequate core cooling for all design basis LOCA conditions.

   The functional design bases for the LPCI mode is to pump a total of 7450 gpm of water per loop using the separate pump loops from the suppression pool into the core region of the vessel when there is a 26 psi differential between reactor pressure and the pressure of the suppression pool air volume. Injection flow commences at 225 psid vessel pressure above drywell pressure.

   The initiating signals are vessel level 1, 32 in. above the active core or drywell pressure equal to 2.0 psig. The pumps will attain rated speed in 27 sec and injection valves fully open in 46 sec.

   These original LPCI mode performance capabilities bound the power uprate conditions and ensure adequate core cooling can be provided following a LOCA at uprated power conditions;

b. Suppression pool cooling (SPC) and containment spray cooling (CSC) modes - The RHR system's SPC and CSC modes provide heat removal from the suppression pool and containment by pumping suppression pool water through the system's heat exchangers and discharging the water either directly back to the suppression pool (i.e., in the SPC mode) or discharging the water to the wetwell and drywell spray spargers (i.e., in the CSC mode) where the water is then returned, by drainage, back to the suppression pool. These modes of operation are designed to provide cooling to maintain containment and suppression pool temperatures and pressures following major transients. Suppression pool cooling is manually initiated by the operator; however, at least one RHR loop is placed in the SPC mode to maintain suppression pool temperature $\leq 110°F$. The drywell spray function removes radioactive fission products from the containment atmosphere during a LOCA and is manually initiated within 15 minutes after the event occurs;

c. Shutdown cooling mode - The RHR system's normal shutdown cooling mode removes reactor core decay and sensible heat from the primary reactor system to permit refueling and servicing. This heat removal function is initiated manually after the reactor pressure has been reduced to less than 48 psig (295°F) by discharge of steam to the main condenser. This mode of operation provides the capability to cool down the reactor under controlled conditions with minimal availability impact. Refer to Section 5.4.7.3.1 for shutdown cooling time to reach 212°F;

d. Alternate shutdown cooling mode - The RHR system's alternate shutdown cooling mode is utilized during normal plant operation and design basis events when the normal shutdown cooling mode is not available to remove reactor core decay and sensible heat. This heat removal function is safety related, initiated manually and pumps suppression pool water into the core and allows the water to return to the suppression pool through the SRVs. The design objective of this mode (as established by Regulatory Guide 1.139) is to reach cold shutdown within 36 hrs and to meet the requirements of GDC 34;

e. Fuel pool cooling mode - During normal plant shutdown, when the reactor vessel head has been removed, the RHR system is designed to be capable of being aligned to assist the FPC and cleanup system in maintaining the fuel pool temperature within acceptable limits. In this mode the system is designed to cool water drawn from the fuel pool by passing it through an RHR system heat exchanger and then discharge the water back to the fuel pool;

f. Minimum flow bypass mode - The RHR system minimum flow bypass mode is designed to provide cooling for the RHR pumps during a small break LOCA that does not result in rapid reactor vessel depressurization to below the RHR system shutoff discharge pressure. This mode cools the pumps by providing a pump flow return line to the suppression pool that allows sufficient pump cooling flow to return to the pool until flow in the main discharge line is sufficient to provide adequate pump cooling. When flow in the main discharge lines is sufficient for cooling of the pumps, motor-operated valves in the minimum flow bypass line to the suppression pool automatically close so that all of the system's flow is directed into the main discharge lines;

g. Standby mode - During normal power operation the RHR system is required to be available for the LPCI mode in the event a LOCA occurs. The system is normally maintained in the standby mode. In this mode the system is aligned with the pumps' suction from the suppression pool and all other valves aligned so that only the injection valves are required to open and the RHR pumps started for LPCI flow to be delivered to the reactor following depressurization.

Until adequate flow is established, the RHR pumps are cooled automatically by flow through the minimum flow valves;

h. Reactor steam condensing mode - The reactor steam condensing mode has been deactivated and will no longer be utilized for CGS. No credit has been taken for the steam condensing mode in any safety analysis; and

i. The potential for exceeding the 100°F/hr cooldown limit during the cooldown mode is minimized by precautions and limitations in the appropriate operating procedures.

5.4.7.1.2 Design Basis for Isolation of Residual Heat Removal System from Reactor Coolant System

Interlocks are provided to inhibit shutdown cooling mode alignment whenever reactor pressure is above the design pressure of the low pressure portions of the RHR system (approximately 135 psig).

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. The minimum pressure above which LPCI protection is required is below the design pressure of the low pressure portions of the RHR system. These interlocks also provide protection of the low pressure portions of the RHR system. These interlocks can be reset when pressure has been reduced to approximately 135 psig. The LPCI injection valves are interlocked to prevent opening when reactor pressure is above approximately 460 psig, which also provides protection for the low pressure portions of the RHR system. In addition, automatic isolation may occur for reasons of vessel water inventory retention which is unrelated to piping pressure ratings. See Section 5.2.5 for an explanation of the leak detection system and the isolation signals.

The RHR pumps are protected against damage from a closed discharge valve by means of automatic minimum flow valves, which open when the main line flow is low and close when the main line flow is greater than the setpoint specified in the Technical Specifications.

5.4.7.1.3 Design Basis for Pressure Relief Capacity

The relief valves in the RHR system are sized for one or both of the following bases:

a. Thermal relief,
b. Valve bypass leakage

Relief valves are set to ensure that the design pressure plus 10% accumulation is not exceeded anywhere in the system being protected. A check valve, RHR-V-209, is installed across RHR-V-9 to prevent thermal overpressurization between RHR-V-8 and RHR-V-9.

The relief valves protecting the RHR system are listed below (see Figure 5.4-15):

Relief Valve	Nominal Setpoint (psig)	Required Capacity (gpm)	Piping Location	Design Pressure (psig)
RHR-RV-88A	205	1	RHR pump suction from suppression pool	220 (loop A)
RHR-RV-88B	205	1		220 (loop B)
RHR-RV-88C	110	1		125 (loop C)
RHR-RV-5	183	1	RHR pump suction from recirculation pipe	220
RHR-RV-25A	487	1	RHR discharge	500
RHR-RV-25B	488	1	RHR discharge	500
RHR-RV-25C	493	1	RHR discharge	500
RHR-RV-30	103	1	RHR flush line to radwaste	125
RHR-RV-36*				

All RHR relief valves are purchased to ASME Section III, Class 2, requirements to match the requirements of the piping they are protecting. As such, the setpoint tolerance is plus or minus 3% for setpoints above 70 psi per ASME Section III, Paragraph NC-7600. Pressure buildups in isolated lines will be slow and discharges from relief valves on these lines will be small. Water hammer and other hydrodynamic loads are not considered a potential problem in RHR relief valve piping.

Redundant interlocks prevent opening valves to the low-pressure suction piping when the reactor pressure is above the shutdown range. These same interlocks initiate valve closure on increasing reactor pressure.

A pressure interlock prevents connecting the discharge piping to the primary system whenever the primary pressure is greater than the design value. In addition a high-pressure check valve will close to prevent reverse flow if the pressure should increase. Relief valves in the discharge piping are sized to account for leakage past the check valve.

The RHR cooling system is connected to higher pressure piping at (a) shutdown cooling suction, (b) shutdown cooling return, (c) LPCI injection, and (d) head spray. The vulnerability to overpressurization of each location is discussed in the following paragraphs:

* RHR-RV-36 has been permanently removed from Columbia Generating Station. It has been replaced with a blind-flanged “Testable Pipe Spool Assembly,” RHR-TPSA-1.

The shutdown cooling suction piping has two gate valves (RHR-V-8 and RHR-V-9) in series which have independent pressure interlocks to prevent opening at high reactor pressure. No single active failure or operator error will result in overpressurization of the lower pressure piping. With the RHR pumps normally lined up to the suppression pool (RHR-V-6A and RHR-V-6B closed), the shutdown cooling suction line is protected from thermal expansion or from leakage past RHR-V-8 by RHR-RV-5. A bypass around RHR-V-6A may also be used to route leakage past RHR-V-8 and RHR-V-9 to the suppression pool. With all the RHR suction valves closed, the suction piping is protected from thermal expansion or leakage past the discharge check valves by RHR-RV-88A, RHR-RV-88B, and RHR-RV-88C. When the bypass around RHR-V-6A is not in service, it will be isolated using a single valve. This will allow the installed relief valves discussed above to protect the bypass piping.

The shutdown cooling return line has swing check valves (RHR-V-50A and RHR-V-50B) to protect it from higher vessel pressures. Additionally, a gate valve (RHR-V-53A and RHR-V-53B) is located in series and has a pressure interlock to prevent opening at high reactor pressures. No single active failure or operator error will result in overpressurization of the lower pressure piping.

Each LPCI injection line has a swing check valve (RHR-V-41A, RHR-V-41B, and RHR-V-41C) to protect it from higher vessel pressures. Additionally, a gate valve (RHR-V-42A, RHR-V-42B, and RHR-V-42C) is located in series and has pressure interlocks to prevent opening at high reactor vessel pressure. No single active failure or operator error will result in overpressurization of the lower pressure piping.

The head spray piping has three swing check valves in series [two belonging to the RCIC system and one (RHR-V-19) belonging to the RHR system], to protect it from higher vessel pressures. Two of the swing check valves have air operators but are only capable of opening the testable check valve if the differential pressure is less than 5.0 psid. Additionally, a globe valve (RHR-V-23) is located in series and has a pressure interlock to prevent opening at high reactor pressures. No single active failure or operator error will result in the overpressurization of the lower pressure piping.

Overpressurization protection of the RHR discharge piping for thermal expansion or from leakage past the head spray, shutdown injection, and LPCI isolation valves is provided by RHR-RV-25A, RHR-RV-25B, and RHR-RV-25C.

The RHR drain system to radwaste is protected from thermal expansion or from leakage past the isolation valves RHR-V-71A, RHR-V-71B, RHR-V-71C, RHR-V-72A, RHR-V-72B, and RHR-V-72C by RHR-RV-30.

5.4.7.1.4 Design Basis With Respect to General Design Criterion 5

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

5.4.7.1.5 Design Basis for Reliability and Operability

The design basis for the shutdown cooling modes of the RHR system is that these modes are controlled by the operator from the control room. The operations performed outside of the control room using the normal shutdown is manual operation of a local flushing water admission valve, which is the means of ensuring that the suction line of the shutdown portions of the RHR system is filled and vented. In addition, the 0.75-in. bypass around RHR-V-6A would be isolated if necessary.

Two modes of operation provide the shutdown cooling function for the RHR system. One mode, the normal Shutdown Cooling Mode, is the preferred operational mode. Although preferred, this mode of RHR does not meet the redundancy and single failure requirements of IEEE 279 and 10 CFR 50 Appendix A, GDC 34. As a result, a second shutdown cooling mode, the Alternate Shutdown Cooling Mode, is provided and is the shutdown cooling mode credited to meet the requirements of IEEE 279 and GDC 34. This mode is safety related, Quality Class 1, Seismic Category 1, redundant and single failure proof. Since the normal Shutdown Cooling Mode of RHR is preferred for CGS, the components required for the operation of this mode are maintained as safety related, Quality Class 1.

For the normal shutdown cooling mode, two separate shutdown cooling loops are provided. The reactor coolant temperature can be brought to 212°F in less than 36 hr with only one loop in operation. With the exception of the shutdown suction including the reactor recirculation loop suction and discharge valves, and shutdown return, the entire RHR system is safety grade and redundant, is part of the ECCS and containment cooling systems, and is designed with the flooding protection, piping protection, power separation, etc., required of such systems. See Section 6.3 for an explanation of the design bases for ECCS systems. Shutdown cooling suction and discharge valves are required to be powered from both offsite and standby emergency power for purposes of isolation and shutdown following a loss of offsite power. In the event that the outboard shutdown cooling suction supply valve (RHR-V-8) fails to open from the control room, an operator may be sent to open the valve by hand.

If the attempt to open the outboard valve proves unsuccessful, or the inboard shutdown cooling suction supply valve (RHR-V-9) fails to open, the operator will establish the alternate shutdown cooling mode path as described in the notes to Figure 15.2-10, Activity C1 or C2.

For the alternate shutdown cooling mode, if vessel depressurization were to be achieved by manual actuation of relief valves, three valves would need to be actuated to pass sufficient steam flow to depressurize the vessel.

Low-pressure liquid flow test results are presented in NEDE-24988-P. This test program adequately demonstrated the ability to use SRVs in the alternate shutdown cooling mode.

Following reactor depressurization (i.e., 100°F/hr), an alternate shutdown coolant flow rate of 2600 gpm would be required to bring the reactor to a shutdown condition. This flow capacity can be achieved by using one ADS valve. However, three valves are always available.

Calculations demonstrate that in the alternate shutdown cooling mode, with one RHR pump in operation, the total system resistance head is 550 ft using one SRV valve. At this calculated head, the pump capacity is 4000 gpm and the reactor pressure is 160 psig.

The air supply for the ADS valves is discussed in Sections 5.2.2, 6.2.2, and 7.3.1.

5.4.7.1.6 Design Basis for Protection from Physical Damage

The RHR system is designed to the requirements of Table 3.2-1. With the exception of the common shutdown cooling line, redundant components of the RHR system are physically located in different quadrants of the reactor building, and are supplied from independent and redundant electrical divisions. Further discussion on protection from physical damage is provided in Section 6.3.

5.4.7.2 Systems Design

5.4.7.2.1 System Diagrams

All of the components of the RHR system are shown in Figure 5.4-15. A description of the controls and instrumentation is presented in Section 7.3.1.1.1.

A process diagram and process data are shown in Figures 5.4-16 and 5.4-17. All of the sizing modes of the system are shown in the process data. The functional control diagram for the RHR system is shown in Figure 7.3-10.

Interlocks are provided (a) to prevent draining vessel water to the suppression pool, (b) to prevent opening vessel suction valves above the suction line design pressure, or above the discharge line design pressure with the pump operating at shutoff head, (c) to prevent inadvertent opening of drywell spray valves, and (d) to prevent pump start when suction valve(s) are not open. This interlock is defeated for the RHR FPC assist mode (see Section 9.1.3).

The RHR system may be used to supplement the cooling capacity of the FPC system. This mode requires the installation of spool pieces and the opening of normally locked closed valves (see Section 9.1.3.2 for details).

The normal shutdown cooling mode of RHR loop B can be aligned to return a portion of the cooling flow back into the reactor vessel via the RCIC head spray nozzle.

The LPCS system may be cross tied with the RHR system to provide a flow path from the CST to the LPCS system via RHR. This preoperational alignment provided clean water to the LPCS system during flushing and provided a flowpath to the vessel for the core spray sparger test. This spoolpiece is not expected to be used again during the lifetime of the plant.

The administrative controls used for these spoolpieces, interlocks, and valves are procedurally regulated to ensure proper system function.

5.4.7.2.2 Equipment and Component Description

a. System main pumps

The RHR main system pumps are motor-driven deepwell pumps with mechanical seals. The pumps are sized on the basis of the LPCI mode (modes A1 and A2, see Figure 5.4-17). Design pressure for the pump suction structure is 220 psig with a temperature range from 40°F 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 housing is carbon steel and the shaft is stainless steel. System configuration (elevation, piping design, etc.) ensures that minimum pump NPSH requirements are met with margin.

Figures 5.4-18 through 5.4-20 are the actual pump performance curves.

The RHR pumps are designed for the life of the plant (40 years) and tested for operability assurance and performance as follows:

1. In-shop tests, including: (a) hydrostatic tests of pressure retaining parts at 1.5 times the design pressure, (b) performance tests to determine the total developed head at zero flow and design flow, and (c) NPSH requirements.

2. After the pumps were installed in the plant, they underwent (a) the system hydro test, (b) functional tests, (c) periodic testing to verify operability in accordance with the Inservice Testing (IST) Program Plan, and (d) about 1 month of operation each year for a refueling shutdown (shutdown operation time has been reduced coincident with reduced outage times).

3. In addition, the pumps are designed for a postulated single operation of 3 to 6 months for one accident during the 40 year life of the plant.

A listing of GE operating experience of Ingersoll-Rand RHR pumps is provided in Tables 5.4-3 and 5.4-4.

b. Heat exchangers

The RHR system heat exchangers are sized on the basis of the duty for the shutdown cooling mode (mode E of the Process Data). All other uses of these exchangers require less cooling surface.

Flow rates are 7450 gpm (rated) on the shell side and 7400 gpm (rated) on the tube side (service water side). Rated inlet temperature is 95°F tube side. Design temperature range of both shell and tube sides are 40°F to 480°F. The tube side water temperature may be as low as 32°F. The low temperature condition is acceptable, based on compliance with the ASME III, Class 2, code. Design pressure is 500 psig on both sides. Fouling allowances are 0.0005 shell side and 0.002 tube side. The construction materials are carbon steel for the pressure vessel with stainless steel tubes and stainless steel clad tube sheet.

c. Valves

All of the directional valves in the system are conventional gate, globe, and check valves designed for nuclear service. The injection valves, reactor coolant isolation valves, and pump minimum flow valves are high speed valves, as operation for LPCI injection or vessel isolation requires. Valve pressure ratings are specified as necessary to provide the control or isolation function: i.e., all vessel isolation valves are rated as Class 1 nuclear valves rated at the same pressure as the primary system.

The pump minimum flow valves (RHR-FCV-64) open automatically at main line flows less than approximately 800 gpm. This allows flow to return to the suppression pool through the minimum flow bypass line, which branches off the main line upstream of the flow element. The minimum flow valve closes at main line flows greater than approximately 900 gpm and forces the entire pump discharge flow through the main line. The minimum flow valve controls meet IEEE-279 requirements.

To prevent loss of vessel inventory to the suppression pool when operating shutdown cooling or RHR/FPC assist mode, the minimum flow valve is not

permitted to open. Administrative controls ensure that the valve is returned to normal status following the conclusion of shutdown cooling.

d. Restricting orifices

The metering orifices in the discharge lines do not serve as restricting orifices. The piping for each mode of RHR operation has been investigated to ensure that the resistance is low enough to allow the rated flows given in Figure 5.4-17 yet high enough to prevent pump runout. Restricting orifices are necessary in the system test lines to prevent excessive runout during SPC and test modes and in the main discharge line to prevent excessive runout for LPCI A & C systems. In addition, restriction orifices are installed ahead of the RHR-V-53A and RHR-V-53B valves to prevent excessive pump runout or valve cavitation during the shutdown cooling mode. Figure 5.4-15 indicates the location of restricting orifices.

Additionally, two orifices are installed in the FPC system to minimize cavitation and limit flow when RHR is used to assist FPC.

e. ECCS portions of the RHR system

The ECCS portions of the RHR system include those sections described in Figure 5.4-16.

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

The SPC components include pool suction strainers, suction piping, pumps, heat exchangers, and pool return lines.

Containment spray components are the same as SPC except that the spray headers replace the pool return lines.

5.4.7.2.3 Controls and Instrumentation

Controls and instrumentation for the RHR system are described in Section 7.3. The RHR system relief valve capacities and settings are listed in Section 5.4.7.1.3.

5.4.7.2.4 Applicable Codes and Classifications

See Section 3.2.

5.4.7.2.5 Reliability Considerations

The RHR system has included the redundancy requirements of Section 5.4.7.1.5. Two redundant loops have been provided to remove residual heat. With the exception of the common shutdown cooling line and the shutdown return valves (RHR-V-53A and RHR-V-53B) which are powered from the same division power source, all mechanical and electrical components are separate. Either loop is capable of cooling down the reactor within a reasonable length of time.

5.4.7.2.6 Manual Action

RHR (shutdown cooling mode)

In the shutdown cooling mode of operation, when reactor vessel pressure is 48 psig or less, a service water pump is started and cooling water flow established through a heat exchanger. The RHR pump suction valve RHR-V-4A and/or RHR-V-4B is then closed and shutdown cooling isolation valves RHR-V-9 and RHR-V-8 opened. RHR pump suction valve RHR-V-6A and/or RHR-V-6B is then opened. Pump suction piping is prewarmed and provided a nominal flush by opening valves to radwaste. These effluent valves to radwaste are then closed and the RHR pump is started. The cooldown rate is controlled by adjusting the heat exchanger outlet valve and heat exchanger bypass valve to achieve the desired temperature of the water returning to the reactor vessel while maintaining the total flow at approximately 7450 gpm.

If prewarming valves were accidentally left open following initiation of shutdown cooling, reactor pressure vessel (RPV) coolant inventory would drain to radwaste. If loss of inventory remained undetected and makeup did not occur, isolation valves RHR-V-8 and RHR-V-9 would automatically close at the RPV scram level specified in the Technical Specifications; depressurization or loss of water from the RHR system causes a low pressure alarm in the RHR discharge piping.

If the bypass around RHR-V-6A were inadvertently left open following the initiation of shutdown cooling using RHR loop B, the RPV coolant inventory would drain to the suppression pool at a flow rate of 1 gpm or less. If this loss of inventory remained undetected and makeup did not occur, RHR-V-8 and RHR-V-9 would automatically close at the RPV scram level.

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

5.4.7.3 Performance Evaluation

Thermal performance of the RHR heat exchangers is based on the capability to remove enough sensible and decay heat from the reactor system to reduce the bulk reactor coolant temperature

to 125°F within 25 hours after control rod insertion, with two RHR loops in operation. Because cooldown is usually a controlled operation, maximum service water temperature less 10°F is used as the service water inlet temperature. These are nominal design conditions; if the service water temperature is higher, the exchanger capabilities are reduced and the cooldown time may be longer or vice versa.

5.4.7.3.1 Shutdown Cooling With All Components Available

No typical curve is included here to show vessel cooldown temperatures versus time due to the infinite variety of such curves due to (a) clean steam systems that use the main condenser as the heat sink when nuclear steam pressure is insufficient to maintain steam air ejector performance, (b) the fouling of the heat exchangers, (c) operator use of one or two cooling loops, (d) coolant water temperature, and (e) 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 used at high vessel temperatures. Total flow mix temperature is controlled to avoid exceeding 100°F/hr cooldown rate. See Figure 5.4-21 for minimum shutdown cooling time to reach 212°F.

5.4.7.3.2 Shutdown Cooling With Most Limiting Failure

Shutdown cooling under conditions of the most limiting failure is discussed in Section 5.4.7.1.5. The capability of the heat exchanger for any time period is balanced against residual heat, pump heat, and sensible heat. The excess over residual heat and pump heat is used to reduce the sensible heat.

5.4.7.4 Preoperational Testing

The preoperational test program and startup test program were used to generate data to verify the operational capabilities of equipment in the system, such as each instrument, setpoint, logic element, pump, heat exchanger, valve, and limit switch. In addition these programs verified the capabilities of the system to provide the flows, pressures, 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 were tested electrically; valves, pumps, controllers, relief valves were tested mechanically; finally the system was tested for total system performance against the design requirements as specified above using both the offsite power and standby emergency power. Preliminary heat exchanger performance was evaluated by operating in the pool cooling mode, but a vessel cooldown was used for the final check due to the small temperature differences available with pool cooling (see Section 14.2).

5.4.8 REACTOR WATER CLEANUP SYSTEM

The reactor water cleanup (RWCU) system is an auxiliary system, a small part of which is part of the RCPB up to and including the outermost containment isolation valve. The other portions of the system are not part of the RCPB and are isolated from the reactor.

5.4.8.1 Design Bases

5.4.8.1.1 Safety Design Bases

The RCPB portion of the RWCU system meets the requirements of Regulatory Guides 1.26 and 1.29 to

a. Prevent excessive loss of reactor coolant,

b. Prevent the release of radioactive material from the reactor,

c. Isolate the cleanup system from the RCPB, and

d. Prevents loss of liquid reactivity control material from the reactor vessel during standby liquid control (SLC) system operation.

5.4.8.1.2 Power Generation Design Bases

The RWCU system

a. Removes solid and dissolved impurities from reactor coolant such that the water purity meets Regulatory Guide 1.56,

b. Discharges excess reactor water during startup, shutdown, and hot standby conditions,

c. Minimizes temperature gradients in the recirculation piping and vessel during periods when the main recirculation pumps are unavailable,

d. Minimizes cleanup system heat loss, and

e. Enables the major portion of the RWCU system to be serviced during reactor operation.

5.4.8.2 System Description

The RWCU system (see Figures 5.4-22 and 5.4-23) continuously purifies reactor water during all modes of reactor operation. The system takes suction from the inlet of each reactor main recirculation pump and from the reactor pressure vessel bottom head. Processed water is returned to the reactor pressure vessel, to the main condenser, or radwaste.

The cleanup system can be operated at any time during planned operations, or it may be shut down. The cleanup system is classified as a primary power generation system. The cleanup system is not an engineered safety system.

Major equipment of the RWCU system is located in the reactor building. This equipment includes the pumps and the regenerative and nonregenerative heat exchangers. Filter-demineralizers and supporting equipment are located in the radwaste building. The entire system is connected by associated valves and piping; controls and instrumentation provide proper system operation. Design data for the major pieces of equipment are presented in Table 5.4-5.

Reactor water is cooled in the regenerative and nonregenerative heat exchangers, filtered, demineralized, and returned to the reactor pressure vessel through the shell side of the regenerative heat exchanger.

The system pump is capable of producing a nominal flow of 181,300 lbm/hr. Two filter demineralizer units are used to process this quantity of water. The system can operate at reduced flow rates with one filter demineralizer unit.

The temperature of water processed through the filter-demineralizers is limited by the resin operating temperature. Therefore, the reactor water must be cooled before being processed in the filter-demineralizers. The regenerative heat exchanger transfers heat from the tube side (hot process) to the shell side (cold process). The shell side flow returns to the reactor. The nonregenerative heat exchanger cools the process further by transferring heat to the reactor building closed cooling water system.

The filter-demineralizers (see Figure 5.4-24) are pressure precoat type filters using ion exchange resins. Spent resins are not regenerable and are sluiced from the filter-demineralizers to a backwash receiving tank from which they are transferred to the radwaste system for processing and disposal. To prevent resins from entering the RRC in the event of complete failure of a filter-demineralizer resin septum, a strainer is installed on each filter-demineralizer. Each strainer and filter-demineralizer vessel has a control room alarm that is energized by high differential pressure. Further increase in differential pressure will isolate the filter-demineralizer. The backwash and precoat cycle for a filter-demineralizer is automatic to prevent operational errors such as inadvertent openings of valves that would initiate a backwash or contaminate reactor water with resins. The filter-demineralizer piping

configuration is arranged to ensure that transfers are complete and crud traps are avoided. A bypass line is provided around the filter-demineralizers.

On low flow or loss of flow in the system, flow is maintained through each filter-demineralizer by its own holding pump. Sample points are provided in the common influent header and in each effluent line of the filter-demineralizers for continuous indication and recording of system conductivity. High conductivity is annunciated in the control room. The influent sample point is also used as the normal source of reactor coolant grab samples. Sample analysis also indicates the effectiveness of the filter-demineralizers.

The suction line of the RCPB portion of the RWCU system contains two motor-operated isolation valves that automatically close in response to signals from the RPV low water level and the leak detection system. The outboard isolation valve, RWCU-V-4, automatically closes in response to signals from actuation of the SLC system and high nonregenerative heat exchanger outlet water temperature. These actions prevent (a) loss of reactor coolant, (b) release of radioactive material from the reactor, (c) removal of liquid reactivity control material, and (d) thermal damage to ion-exchange resins. The RCPB isolation valves may be remote manually operated to isolate the system equipment for maintenance or servicing.

A remote manual-operated gate valve on the return line to the reactor provides long-term leakage control. Instantaneous reverse flow isolation is provided by check valves in the RWCU piping.

Operation of the RWCU system is controlled from the main control room. Resin-changing operations, which include backwashing and precoating, are controlled from the radwaste control room in the radwaste building.

A functional control diagram is provided in Figure 7.3-1.

5.4.8.3 System Evaluation

The RWCU system in conjunction with the condensate treatment system and FPC and cleanup system maintains reactor water quality during all reactor operating modes (normal, standby, startup, shutdown, and refueling). The RWCU components provide a system with the capability to support reactor operations at power levels up to 3629 MWt.

The component pressure and temperature design conditions are shown in Table 5.4-5. The process containing components (piping, valves, vessels, heat exchangers, pumps) are designed to the requirements of Section 3.2. The control requirements for the RCPB isolation valves are designed to the requirements of Table 7.3-5. The nonregenerative heat exchanger is sized to maintain the process temperature required for the cleanup demineralizer resin when the cooling capacity of the regenerative heat exchanger is reduced at times when flow is partially bypassed to the main condenser or radwaste.

5.4.8.4 Demineralizer Resins

Regulatory Guide 1.56 compliance is described in Section 1.8.

5.4.8.5 Reactor Water Cleanup Water Chemistry

5.4.8.5.1 Analytical Methods

Chemical analyses methods used for determination of conductivity, pH, and chloride content of primary coolant are as follows:

Conductivity	measured in accordance to ASTM-D-1125
pH	measured in accordance to ASTM-D-1293
Chloride	determined by ion chromatography in accordance with the vendor's operating manual

5.4.8.5.2 Relationship of Filter-Demineralizer Condition to Water Chemistry

The filter-demineralizer condition during normal power operation is related to inlet conductivity and water volume processed through the unit. The inlet conductivity is related to impurity concentration through the equivalent conductance of the constituents of the process fluid. System flow rates are measured and recorded to determine quantity of water processed.

Periodically, an On-Line NobleChem™ application will be performed, which injects platinum into the reactor coolant, resulting in a microscopic layer of the noble metal to be deposited onto the reactor internals.

Conductivity instrumentation is calibrated against laboratory flow cells in accordance with ASTM-D-1125. The alarm setpoints for the conductivity instrumentation at the inlet and outlet of the filter-demineralizers are set to indicate marginal performance or breakthrough of the filter-demineralizers.

The quantity of the principle ion(s) likely to cause demineralizer breakthrough are not calculated using conductivity as discussed in position 4.C of Regulatory Guide 1.56. Instead, actual ion sample data is taken and used to determine ion levels at the outlet of the filter-demineralizer. When sample data indicates resin breakthrough or the allowable pressure drop is exceeded, the filter-demineralizer is regenerated.

5.4.9 MAIN STEAM LINES AND FEEDWATER PIPING

5.4.9.1 Safety Design Bases

To satisfy the safety bases, the main steam and feedwater lines have been designed

a. To accommodate operational stresses, such as internal pressures and SSE loads, without a failure that could lead to the release of radioactivity in excess of the guideline values in published regulations, and

b. With suitable accesses to permit IST and inspections.

5.4.9.2 Power Generation Design Bases

To satisfy the design bases

a. The main steam lines have been designed to conduct steam from the reactor vessel over the full range of reactor power operation, and

b. The feedwater lines have been designed to conduct water to the reactor vessel over the full range of reactor power operation.

5.4.9.3 Description

The main steam piping is described in Section 10.3. The main steam and feedwater piping is shown in Figure 10.3-2.

The feedwater piping consists of two 24-in. O.D. lines which penetrate the containment and drywell and branch into three 12-in. lines each, which connect to the reactor vessel. Each 24-in. line includes three containment isolation valves consisting of one check valve inside the drywell and one motor-operated gate valve and one check valve outside the containment. The design pressure and temperature of the feedwater piping between the reactor and maintenance valve is 1300 psig and 575°F. The Seismic Category I design requirements are placed on the feedwater piping from the reactor through the outboard isolation valve and connected piping up to and including the first isolation valve in the connected piping.

The materials used in the piping are in accordance with the applicable design code and supplementary requirements described in Section 3.2.

The feedwater system is further described in Sections 7.7.1, 7.7.2, and 10.4.7.

5.4.9.4 Safety Evaluation

Differential pressure on reactor internals under the assumed accident condition of a ruptured steam line is limited by the use of flow restrictors and by the use of four main steam lines. All main steam and feedwater piping is designed in accordance with the requirements defined in Section 3.2.

5.4.9.5 Inspection and Testing

Inspection and testing of the main steam lines and feedwater piping is performed in accordance with the ISI Program Plan to ensure compliance with applicable codes.

5.4.10 PRESSURIZER

Not Applicable to BWRs.

5.4.11 PRESSURIZER RELIEF DISCHARGE SYSTEM

Not Applicable to BWRs.

5.4.12 VALVES

5.4.12.1 Safety Design Bases

Line valves such as gate, globe, and check valves are located in the fluid systems to perform a mechanical function. Valves are components of the system pressure boundary and, having moving parts, are designed to operate efficiently to maintain the integrity of this boundary.

The valves operate under the internal pressure/temperature loading as well as the external loading experienced during the various system transient operating conditions. The design criteria, the design loading, and acceptability criteria are as required in Section 3.9.3 for ASME Class 1, 2, and 3 valves. Compliances with ASME Codes are discussed in Section 5.2.1.

5.4.12.2 Description

Line valves furnished are manufactured standard types, designed and constructed in accordance with the requirements of ASME Section III for Class 1, 2, and 3 valves. All materials, exclusive of seals, packing and wearing components, are designed to endure the 40-year plant life under the environmental conditions applicable to the particular system when appropriate maintenance is periodically performed.

Power operators have been sized to operate successfully under the maximum differential pressure determined in the design specification or design basis calculations.

5.4.12.3 Safety Evaluation

Line valves are shop tested by the manufacturer for performability. Pressure retaining parts are subject to the testing and examination requirements of Section III of the ASME Code. To minimize internal and external leakage past seating surfaces, maximum allowable leakage rates are stated in the design specifications for both back seat as well as the main seat for gate and globe valves.

Valve construction materials are compatible with the maximum anticipated radiation dosage for the service life of the valves.

5.4.12.4 Inspection and Testing

Valves serving as containment isolation valves and which must remain closed or open during normal plant operation may be partially exercised during this period to assure their operability at the time of an emergency or faulted conditions. Other valves, serving as a system block or throttling valves, may be exercised when appropriate.

Motors used with valve actuators are furnished in accordance with applicable industry standards. Each motor actuator has been assembled, factory tested or tested in-situ, and adjusted on the valve for proper operation, position and torque switch setting, position transmitter function (where applicable), and speed requirements. A selected set of motor-operated valves with active safety functions (Generic Letter 89-10 Program and Generic Letter 96-05 Program) have additionally been tested to demonstrate adequate stem thrust (or torque) capability to open (or close) the valve within the specified time at specified maximum expected differential pressure. Modifications have been made to several gate valves to eliminate the possibility for internal pressure locking forces which could prevent the actuator from unseating the valve (Generic Letter 95-07 Program).

Tests verified no mechanical damage to valve components during full stroking of the valve. Suppliers were required to furnish assurance of acceptability of the equipment for the intended service based on any combination of

a. Test stand data,
b. Prior field performance,
c. Prototype testing, and
d. Engineering analysis.

Preoperational and operational testing performed on the installed valves consists of total circuit check out and performance tests to verify design basis capability including speed requirements at specified differential pressure.

5.4.13 SAFETY AND RELIEF VALVES

A listing of the safety and relief valves is provided in Table 5.4-6.

5.4.13.1 Safety Design Bases

Overpressure protection is provided at isolatable portions of systems in accordance with the rules set forth in the ASME Code, Section III for Class 1, 2, and 3 components.

5.4.13.2 Description

Pressure relief valves are designed and constructed in accordance with the same code class as that of the line valves in the system.

The design criteria, design loading, and design procedure are described in Section 3.9.3.

5.4.13.3 Safety Evaluation

The use of pressure relieving devices will ensure that overpressure will not exceed 10% above the design pressure of the system. The number of relieving devices on a system or portion of a system have been determined on an individual component basis.

5.4.13.4 Inspection and Testing

The valves are inspected and tested in accordance with ASME Section XI, if required.

Other than the main steam relief valves, no provisions are to be made for inline testing of pressure relief valves, other than set pressure and leakage. 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. For subsequent set pressure changes, the valve body will be stamped or a stamped tag will be attached indicating the new pressure.

5.4.14 COMPONENT AND PIPING SUPPORTS

Support elements are provided for those components included in the RCPB and the connected systems.

5.4.14.1 Safety Design Bases

Design loading combinations, design procedures, and acceptability criteria are as described in Section 3.9.3. Flexibility calculations and seismic analysis for Class 1, 2, and 3 component and piping supports within the ASME boundary of jurisdiction conform with the appropriate requirements of ASME Section III, Subsection NF. Outside the ASME boundary steel structures conform to the AISC manual of Steel Construction.

Spacing and size of pipe support elements were based on the piping analysis performed in accordance with ASME Section III and further described in Section 3.7. Standard manufacturer hanger types were used and fabricated of materials per ASME Section III, Subsection NF.

5.4.14.2 Description

The use and location or rigid-type supports, variable or constant spring-type supports, and anchors or guides are determined from the results of static and dynamic analyses of the associated piping systems. The normal and transient (including seismic) support point loads generated by the piping analyses are combined as prescribed by Sections 3.9.3 and 3.7, and then utilized as the design basis loadings for each affected pipe support.

Typically, components support elements are manufacturers' standard items which are purchased with certified load capacity data reports. Nonstandard support structures and pressure boundary attachments are qualified by detailed structural analyses in compliance with applicable load combinations and governing design codes.

As described by Sections 5.4.14.1 and 5.4.14.2, each component support system has been rigorously evaluated with all due consideration for extreme loading conditions and satisfaction of conservative design allowable stresses. This demonstration of structural adequacy combined with a comprehensive testing and inspection program (see Section 5.4.14.3) constitutes the safety evaluation basis for these passive support elements.

5.4.14.3 Inspection and Testing

After completion of the installation and balancing of a support system, all hanger elements were visually examined to ensure that they were in correct adjustment to their cold setting position. On initial hot startup operations, thermal growth was observed and it was confirmed that all spring-type hangers and snubbers were functioning properly between their hot and cold setting positions. In addition, during power ascension testing critical systems were instrumented and monitored for vibration response under normal and plant transient conditions. The results of these tests showed all systems to be functioning as predicted by design analyses and thus all systems were accepted as operable and in compliance with the governing ASME Code.

5.4.15 HIGH-PRESSURE CORE SPRAY SYSTEM

See Section 6.3 for a description of the HPCS system.

5.4.16 LOW-PRESSURE CORE SPRAY SYSTEM

See Section 6.3 for a description of the LPCS system.

5.4.17 STANDBY LIQUID CONTROL SYSTEM

See Section 9.3.5 for a description of the SLC system.

5.4.18 REFERENCES

5.4-1 Ianni, P. W., "Effectiveness of Core Standby Cooling Systems for General Electric Boiling Water Reactors," APED-5458, March 1968.

5.4-2 "Design and Performance of General Electric Boiling Water Reactor Main Steam Line Isolation Valves," APED-5750, General Electric Co., Atomic Power Equipment Department, March 1969.

5.4-3 "Power Uprate with Extended Load Line Limit Safety Analysis for WNP-2," NEDC-32141P, General Electric Company.

5.4-4 "Generic Evaluations of General Electric Boiling Water Reactor Power Uprate - Volume I," NEDC-31984P, General Electric Company.

5.4-5 "Reactor Core Isolation Cooling System (RCIC)," Design Basis Document, Section 315.

Table 5.4-1

Reactor Coolant Pressure Boundary Pump and Valve Description[a]

Location	Active/Inactive	Valve	Reference Figure
Valve Description			
RHR vessel in	Active	RHR-V-41A	5.4-15
	Active	RHR-V-41B	5.4-15
	Active	RHR-V-41C	5.4-15
		(E12F041A, B, C)	
	Active	RHR-V-42A	5.4-15
	Active	RHR-V-42B	5.4-15
	Active	RHR-V-42C	5.4-15
		(E12F042A, B, C)	
	Inactive	RHR-V-111A	5.4-15
	Inactive	RHR-V-111B	5.4-15
	Inactive	RHR-V-111C	5.4-15
		(E12F111A, B, C)	
RHR/recirculation line in	Active	RHR-V-50A	5.4-15
	Active	RHR-V-50B	5.4-15
		(E12F050A, B)	
	Active	RHR-V-53A	5.4-15
	Active	RHR-V-53B	5.4-15
		(E12F053A, B)	
	Inactive	RHR-V-112A	5.4-15
	Inactive	RHR-V-112B	5.4-15
		(E12F112A,B)	
	Inactive	RHR-V-123A	5.4-15
	Inactive	RHR-V-123B	5.4-15
		(E12F099A, B)	
Head spray	Active	RHR-V-19 (E12F019)	5.4-15
	Active	RHR-V-23 (E12F023)	5.4-15
RHR shutdown cooling suction	Active	RHR-V-8 (E12F008)	5.4-15
	Active	RHR-V-9 (E12F009)	5.4-15
	Inactive	RHR-V-113 (E12F113)	5.4-15
RCIC vessel out	Active	RCIC-V-8 (E51F008)	5.4-11
	Active	RCIC-V-63 (E51F063)	5.4-11
	Active	RCIC-V-64 (E51F064)	5.4-11
	Active	RCIC-V-76 (E51F0076)	5.4-11

Table 5.4-1

Reactor Coolant Pressure Boundary Pump and Valve Description[a] (Continued)

Location	Active/Inactive	Valve	Reference Figure
(Nuclear boiler)			
Reactor vessel head	Inactive	MS-V-1 (B22F001)	10.3-2
	Inactive	MS-V-2 (B22F002)	10.3-2
Feedwater in	Active	RFW-V-10A	10.3-2
	Active	RFW-V-10B (B22F010A, B)	10.3-2
	Inactive	RFW-V-11A	10.3-2
	Inactive	RFW-V-11B (B22F011A, B)	10.3-2
	Active	RFW-V-32A	10.3-2
	Active	RFW-V-32B (B22F032A, B)	10.3-2
	Active	RFW-V-65A	10.3-2
	Active	RFW-V-65B (B22F065A, B)	10.3-2
Safety relief	Active	MS-RV-2A	10.3-2
	Active	MS-RV-3A	10.3-2
	Active	MS-RV-2D	10.3-2
	Active	MS-RV-2C	10.3-2
	Active	MS-RV-1B	10.3-2
	Active	MS-RV-2B	10.3-2
	Active	MS-RV-3C	10.3-2
	Active	MS-RV-3B (B22F013A-H)	10.3-2
	Active	MS-RV-1A	10.3-2
	Active	MS-RV-1D	10.3-2
	Active	MS-RV-1C	10.3-2
	Active	MS-RV-4C	10.3-2
	Active	MS-RV-5C (B22F013J-N)	10.3-2
	Active	MS-RV-4D (B22F013P)	10.3-2
	Active	MS-RV-4B	10.3-2
	Active	MS-RV-4A	10.3-2

Table 5.4-1

Reactor Coolant Pressure Boundary Pump and Valve Description[a] (Continued)

Location	Active/Inactive	Valve	Reference Figure
		(B22F013R-S)	
	Active	MS-RV-5B	10.3-2
	Active	MS-RV-3D	10.3-2
		(B22F013U-V)	
Reactor water cleanup system	Inactive	RWCU-V-103 (G33F103)	5.4-22
Line suction	Active	RWCU-V-1 (G33F001)	5.4-22
	Active	RWCU-V-4 (G33F004)	5.4-22
	Inactive	RWCU-V-100 (G33F100)	5.4-22
	Inactive	RWCU-V-101	5.4-22
	Inactive	(G33F101)	
	Inactive	RWCU-V-102	5.4-22
		(G33F102)	
		RWCU-V-106	5.4-22
		(G33F106)	
Line discharge	Active	RWCU-V-40 (G33F040)	5.4-22
Drain to condenser	Active	MS-V-16	10.3-2
	Active	(B22F016)	
		MS-V-19	10.3-2
		(B22F019)	
MSIV	Active	MS-V-22A	10.3-2
	Active	MS-V-22B	10.3-2
	Active	MS-V-22C	10.3-2
	Active	MS-V-22D	10.3-2
	Active	(B22F022)	
	Active	MS-V-28A	10.3-2
	Active	MS-V-28B	10.3-2
	Active	MS-V-28C	10.3-2
		MS-V-28D	10.3-2
		(B22F028)	

Table 5.4-1

Reactor Coolant Pressure Boundary Pump and Valve Description[a] (Continued)

Location	Active/Inactive	Valve	Reference Figure
Drain to condenser (Recirculation)	Active	MS-V-67A	10.3-2
	Active	MS-V-67B	10.3-2
	Active	MS-V-67C	10.3-2
	Active	MS-V-67D (B22F067)	10.3-2
Recirculation pump suction	Inactive	RRC-V-23A	5.4-7
	Inactive	RRC-V-23B (B35F023)	5.4-7
Flow control (pump discharge)	Inactive[b]	RRC-V-60A	5.4-7
	Inactive[b]	RRC-V-60B (B35F060)	5.4-7
	Inactive	RRC-V-67A	5.4-7
	Inactive	RRC-V-67B (B35F067)	5.4-7
RCIC vessel head in	Active	RCIC-V-13 (E51F013)	5.4-11
	Active	RCIC-V-65 (E51F065)	5.4-11
	Active	RCIC-V-66 (E51F066)	5.4-11
HPCS in	Active	HPCS-V-4 (E22F005)	6.3-4
	Active	HPCS-V-5 (E22F004)	6.3-4
	Inactive	HPCS-V-38 (E22F038)	6.3-4
LPCS in	Active	LPCS-V-5 (E21F005)	6.3-4
	Active	LPCS-V-6 (E21F006)	6.3-4
	Inactive	LPCS-V-51 (E21F051)	6.3-4
Standby liquid control in	Active	SLC-V-4A	9.3-14
	Active	SLC-V-4B	9.3-14
	Active	SLC-V-6	9.3-14
	Active	SLC-V-7	9.3-14
	Inactive	SLC-V-8	9.3-14
Pump description			
Recirculation pump	Inactive	RRC-P-1A	5.4-7
	Inactive	RRC-P-1B (B35C001)	5.4-7

Table 5.4-1

Reactor Coolant Pressure Boundary Pump and Valve Description[a] (Continued)

[a] In addition to the process valves listed herein, there are instrument test conditions, drain valves, and sampling valves less than 1 in. nominal size within the RCPB. See associated system flow diagram figures.

[b] Mechanically blocked in the full open position.

NOTE:

Active components are those whose operability is relied on to perform a safety function during the transients or accidents.

Inactive components are those whose operability (e.g., valve opening or closure, pump operation or trip) is not relied on to perform the system's safety function during the transients or accidents.

Table 5.4-2

Reactor Recirculation System Design Characteristics

Description	
External loops	2
Pump sizes (nominal O.D.)	
Pump suction, in.	24
Pump discharge, in.	24
Discharge manifold, in.	16
Recirculation inlet lines, in.	12
Design pressure (psig)/design temperature (°F)	
Suction piping and valve up to and including pump suction nozzle	1250/575
Pump, discharge valves, and piping between	1650/575
Piping after discharge blocking valve up to vessel	1550/575
Vessel bottom drain	1275/575
Operation at pump related conditions	
Recirculation pump	
Flow, gpm	47,200
Flow, lb/hr	17.85×10^6
Total developed head, ft	$\boxed{805}$
Suction pressure (static), psia	1025
Required NPSH, ft	115
Water temperature (maximum), °F	533
Pump brake hp (minimum)	8340
Flow velocity at pump suction (approximate), ft/sec	41.5
Pump motor	
Voltage rating	6600
Speed, rpm	1780
Motor rating, hp	8900
Phase	3
Frequency	60
Motor rotor inertia (lb-ft^2)	21,500 (RRC-M-P/1B) 20,600 (RRC-M-P/1A)
Jet pumps	
Number	20
Total jet pump flow, lb/hr	108.5×10^6
Total I.D., in.	6.4

Table 5.4-2

Reactor Recirculation System Design Characteristics (Continued)

Description	
Diffuser I.D., in.	19.0
Nozzle I.D. (five each), in.	1.3
Diffuser exit velocity, ft/sec	16.2
Jet pump head, ft	88.19
Flow control valve[a]	
Type	Ball
Material	Austenitic stainless steel
Valve wide open C_V (minimum), gpm/psi	7000
Valve size diameter, in.	24
Recirculation block valve	
Type	Gate valve
Actuator	Motor
Material	Austenitic stainless steel
Valve size diameter, in.	24
Recirculation pump flow measurement	
Type	Elbow taps
Rated flow (gpm)	47,200
Flow element location	Pump suction line
Range	20-115% rated pump flow
Accuracy (% rated pressure drop)	± 9%
Repeatability (% rated pressure drop)	± 4%

[a] Mechanically blocked in the full open position.

Table 5.4-3

Operating Experience of Ingersoll-Rand Emergency Core Cooling Systems Pumps[a,b]

Plant	*Pump*	*Time (hr)*
Hatch 2	*RHR 2A*	*864*
	2B	*1112*
	2C	*629*
	2D	*569*
	LPCS 2A	*13.5*
	2B	*11.8*
Chinshan 1	*RHR*	*100*
	Core spray	*30*
Chinshan 2	*RHR*	*75*
	Core spray	*20*

[a] *The italicized information is historical and was provided to support the application for an operating license.*

[b]*No problems have been reported on these pumps. Pump design principles applied by Ingersoll-Rand to these units are not unique. Assurance of a predictable functional reliability is also provided by a history of design, production, and application of pumps for similar pumping requirements in other nuclear and nonnuclear applications.*

Table 5.4-4

Operating Experience of Similar Ingersoll-Rand Pumps for BWR Projects Under Review[a,b]

Year	*Size Range (gpm)*	*Number of Pumps*
1963	*< 4000*	*12*
1964	*< 3000*	*24*
1965	*< 5000*	*32*
1966	*< 4500*	*39*
1967	*< 5000*	*39*
	8000	*3*
1968	*< 6500*	*25*
	9000	*6*
	11000	*9*
1969	*< 6500*	*39*
	8000-9000	*9*
1970	*< 6500*	*33*
	8000	*14*
	12,000	*6*
1971	*< 6500*	*53*
	9000	*3*
	10,000-12,000	*12*
1972	*< 6500*	*44*
	8000	*18*
	10,000-12,000	*18*
1973	*< 6500*	*41*
	8000	*8*
	10,000-13,800	*20*
1974	*< 6500*	*32*
	8000	*2*
	10,000-13,800	*30*
1975	*< 7500*	*76*
	8500	*18*
	10,000-13,800	*50*
1976	*8500*	*9*

[a] *The italicized information is historical and was provided to support the application for an operating license.*

[b] *The vertical pumps used for ECCS functions at CGS are sized at 1200 to 8100 gpm. They are multistaged axial pumps. Included here is a partial list of the application history for similar pumps made by the same vendor.*

Although the operating experience in nuclear applications is just beginning, the postoperating experience in nonnuclear applications with these vertical pumps is very extensive. It indicates that the CGS ECCS pumps can be expected to operate as required. In reviewing this table, the generic pump design should be recalled because larger capacity pumps are configured from stages that comprise the smaller capacity pumps. Design refinements are evident in the capacity growth of these stages, whether in single, double, or multiple axial stackups.

Table 5.4-5

Reactor Water Cleanup System

Equipment	Design Data	
Main Cleanup Recirculation Pumps		
Number		2
Capacity (each)		100% (@90 bhp)
Design temperature, °F		575
Design pressure, psig		1420
Discharge head at shutoff, ft		575
Minimum available NPSH, ft		16
Heat Exchangers		
	Regenerative	Nonregenerative
Number	1 (3 shells)	1 (2 shells)
Shell design pressure, psig	1420	150
Shell design temperature, °F	575	370
Tube design pressure, psig	1420	1420
Tube design temperature, °F	575	575
Filter-Demineralizers		
Type		Pressure precoat
Number		2
Design temperature, °F		150
Design pressure, psig		1450

Table 5.4-6

Safety and Relief Valves for Piping Systems
Connected to the Reactor Coolant Pressure Boundary

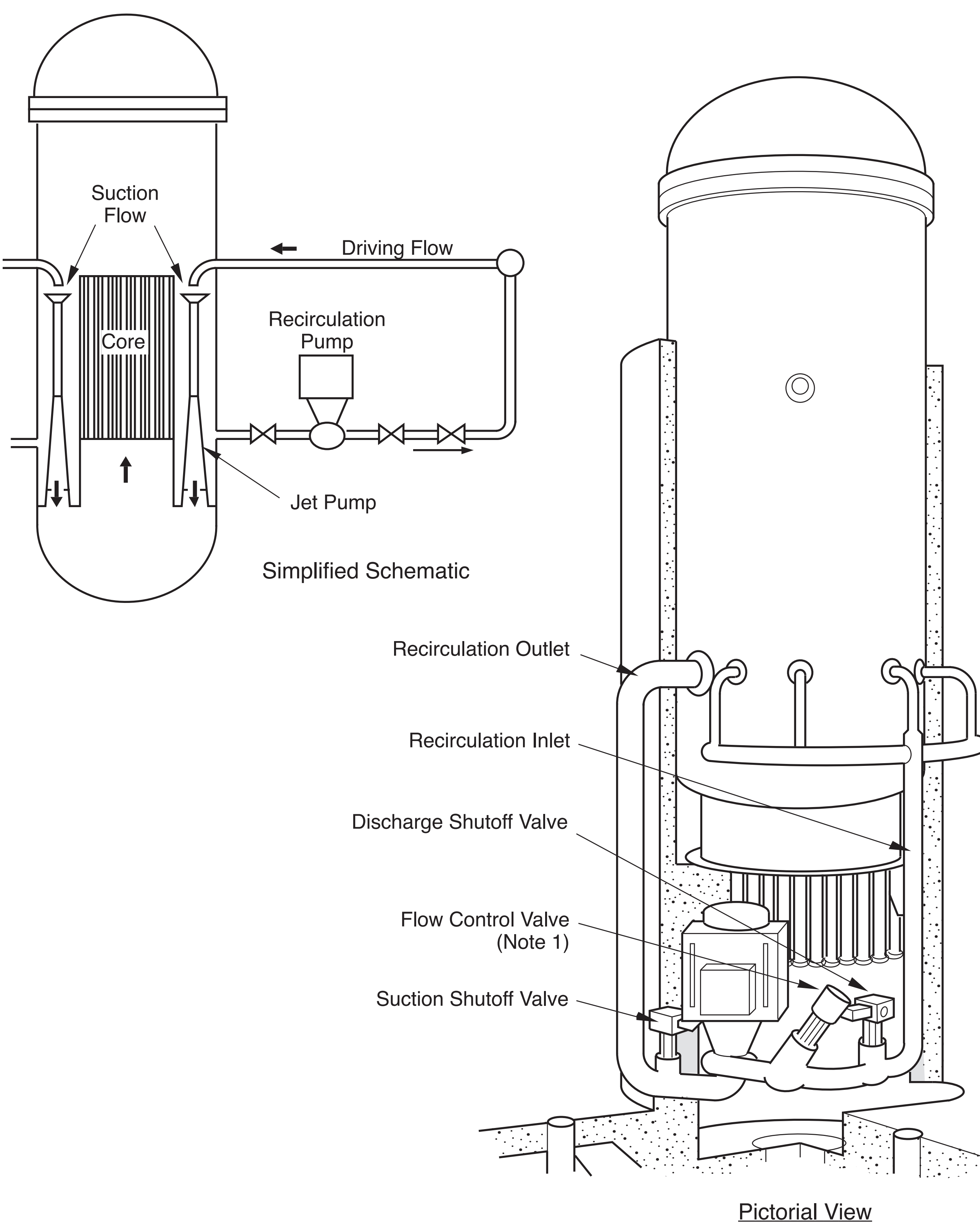
Main steam line safety/relief valves	MS-RV-1A (B22F013J-N) MS-RV-1B (B22F013A-H) MS-RV-1C (B22F013J-N) MS-RV-1D (B22F013J-N) MS-RV-2A (B22F013A-H) MS-RV-2B (B22F013A-H) MS-RV-2C (B22F013A-H) MS-RV-2D (B22F013A-H) MS-RV-3A (B22F013A-H) MS-RV-3B (B22F013A-H) MS-RV-3C (B22F013A-H) MS-RV-3D (B22F013U-V) MS-RV-4A (B22F013R-S) MS-RV-4B (B22F013R-S) MS-RV-4C (B22F013J-N) MS-RV-4D (B22F013P) MS-RV-5B (B22F013U-V) MS-RV-5C (B22F013J-N)
RCIC system discharge line	RCIC-RV-3
RCIC system suction line	RCIC-RV-17 (E51F017)
RCIC lube oil cooler supply line	RCIC-RV-19T
RCIC vacuum tank	RCIC-RV-33 (E51F033)[a]
Shutdown cooling supply line	RHR-RV-5 (E12F005)
Shutdown cooling return line	RHR-RV-25A RHR-RV-25B (F12F025A, B)
Suppression pool supply for RHR	RHR-RV-88A RHR-RV-88B RHR-RV-88C (E12F088A, B, C)
RHR flush line	RHR-RV-30 (E12F030)
RHR heat exchanger (shell side)	RHR-RV-1A RHR-RV-1B
RWCU regenerative heat exchanger (shell side)	RWCU-RV-1[a]

Table 5.4-6

Safety and Relief Valves for Piping Systems
Connected to the Reactor Coolant Pressure Boundary (Continued)

RWCU regenerative heat exchanger (tube side)	RWCU-RV-3[a]
RWCU blowdown to radwaste system or condenser	RWCU-RV-36 (G33F036)[a]
HPCS suction line	HPCS-RV-14 (E22F014)
HPCS discharge line	HPCS-RV-35 (E22F035)
LPCS discharge line	LPCS-RV-18 (E21F018)
LPCS suction line	LPCS-RV-31 (E21F031)
SLC pump discharge line	SLC-RV-29A SLC-RV-29B (C41F029A, B)

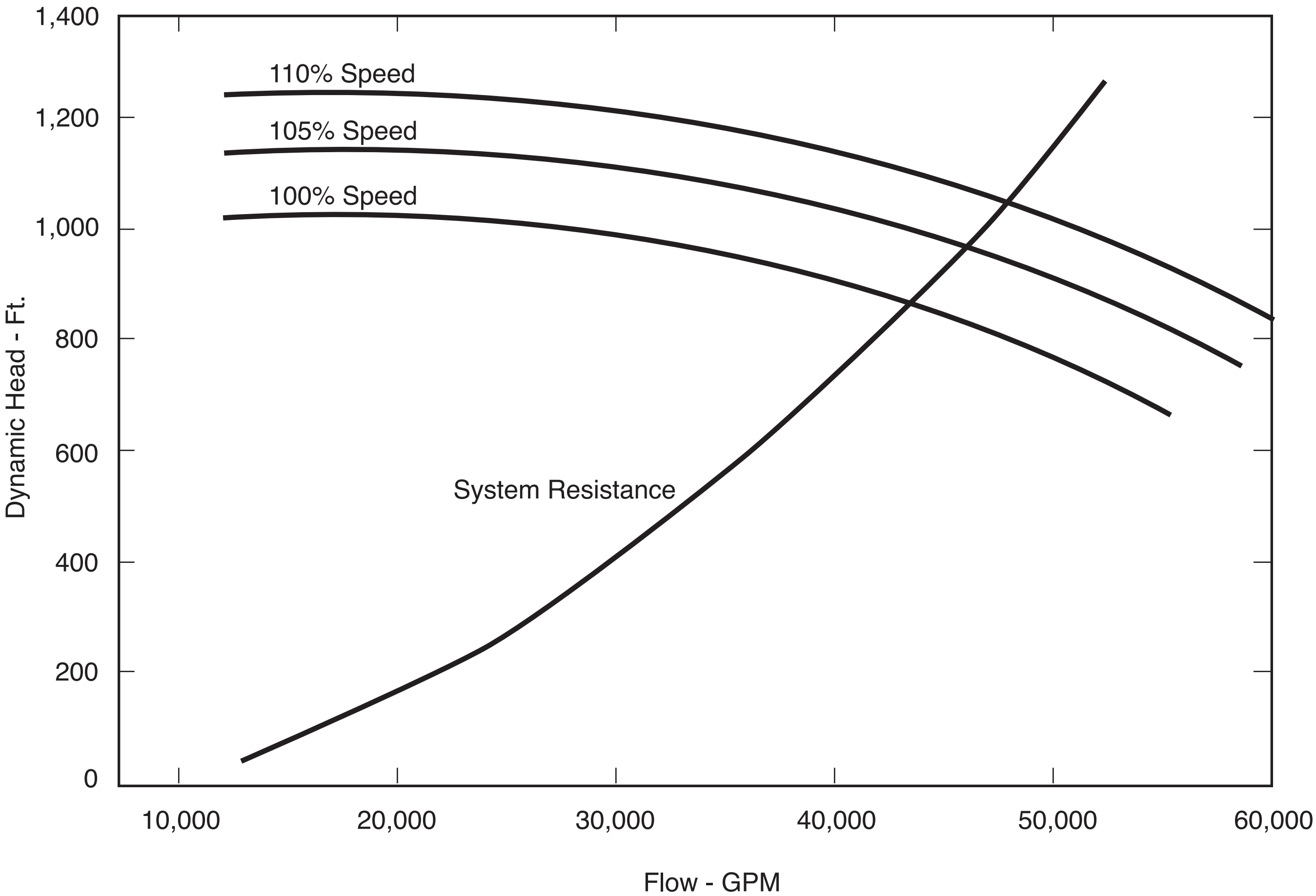
[a] These relief valves are installed in a B31.1 system; not subject to Section XI testing and inspection.



Note 1: FCVs Are Mechanically Blocked Full Open.

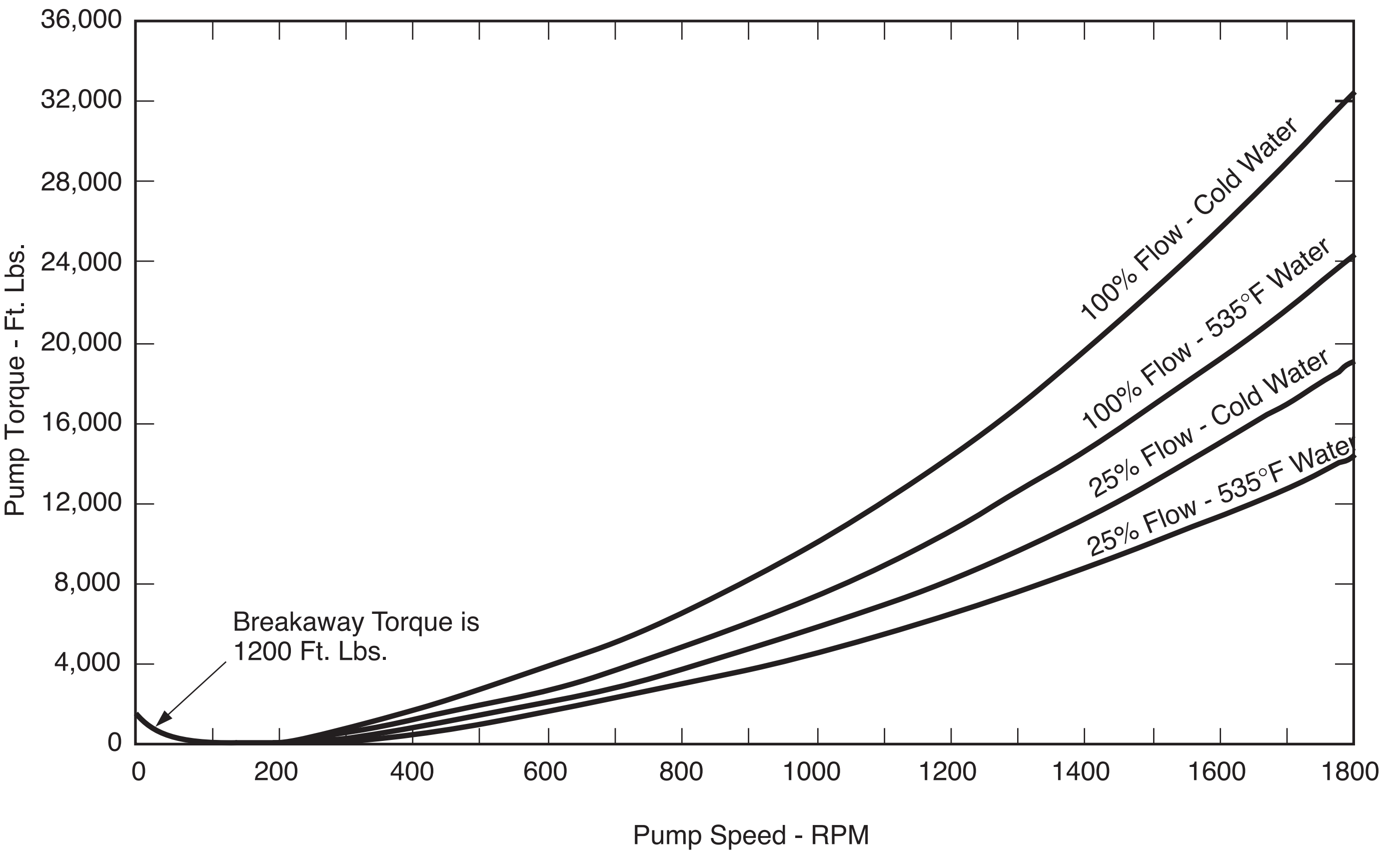
Recirculation System Evaluation and Isometric

Draw. No. 960690.07 | Rev. | Figure 5.4-1

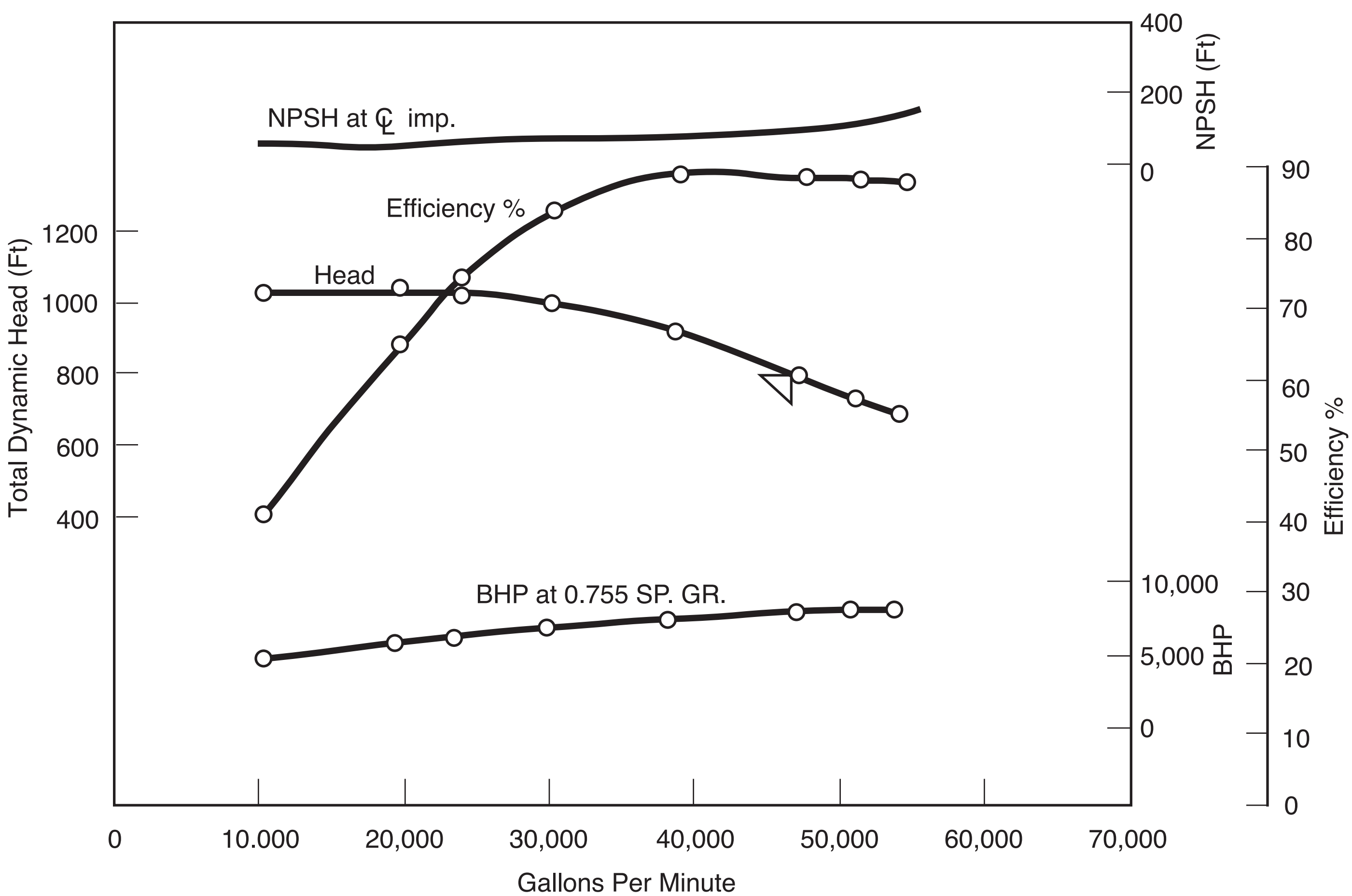


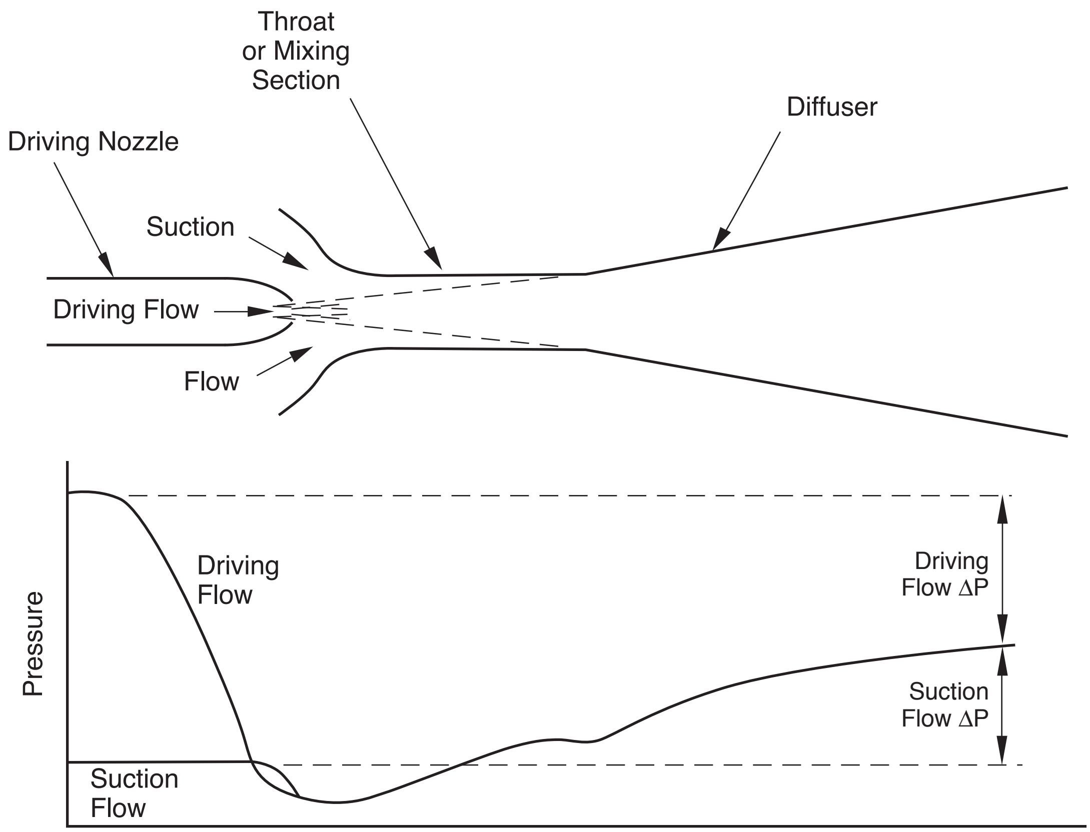
RRC Pump Dynamic Head-Flow Curve

Draw. No. 960690.05 | Rev. | Figure 5.4-2

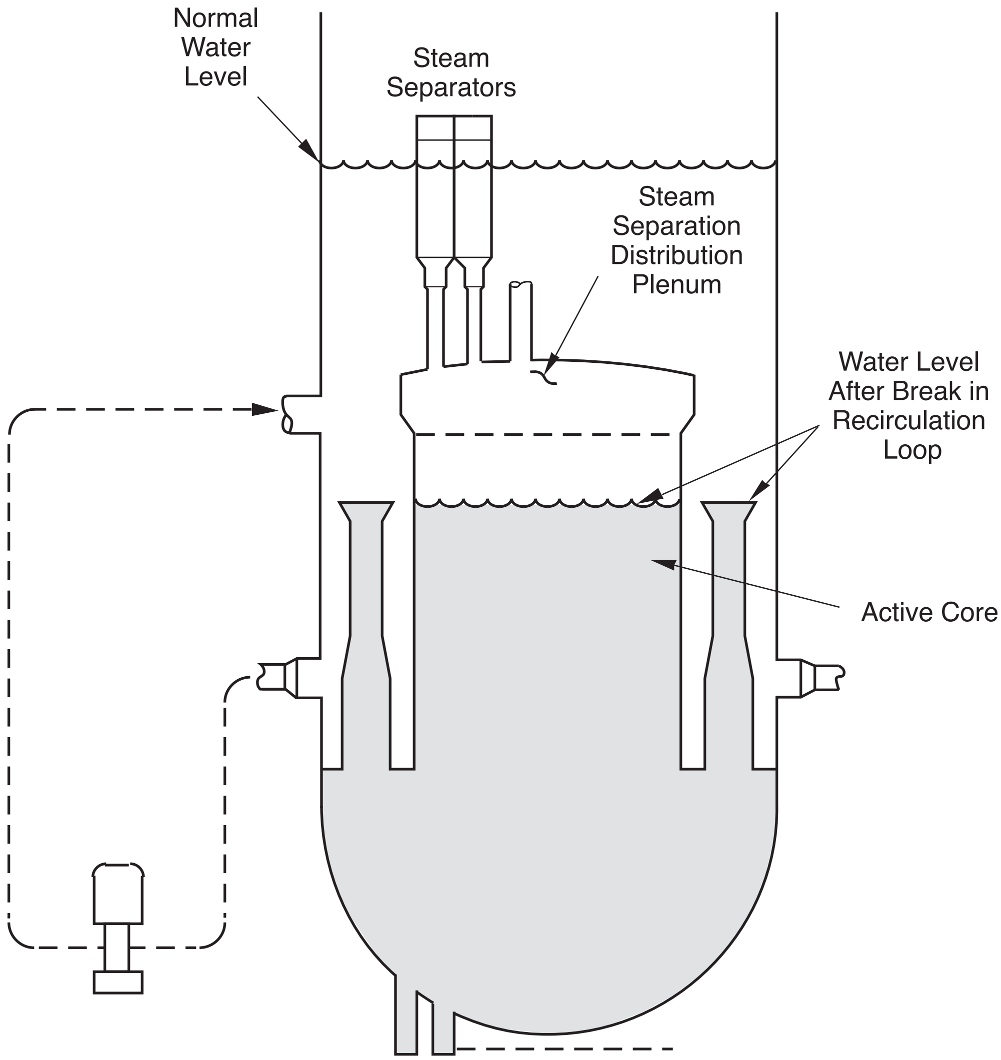


RRC Pump Speed - Torque Curve


NPSH at ℄ imp.
Efficiency %
Head
BHP at 0.755 SP. GR.
Total Dynamic Head (Ft)
1200
1000
800
600
400
400
200
0
NPSH (Ft)
10,000
5,000
0
BHP
90
80
70
60
50
40
30
20
10
0
Efficiency %
0
10.000
20,000
30,000
40,000
50,000
60,000
70,000
Gallons Per Minute



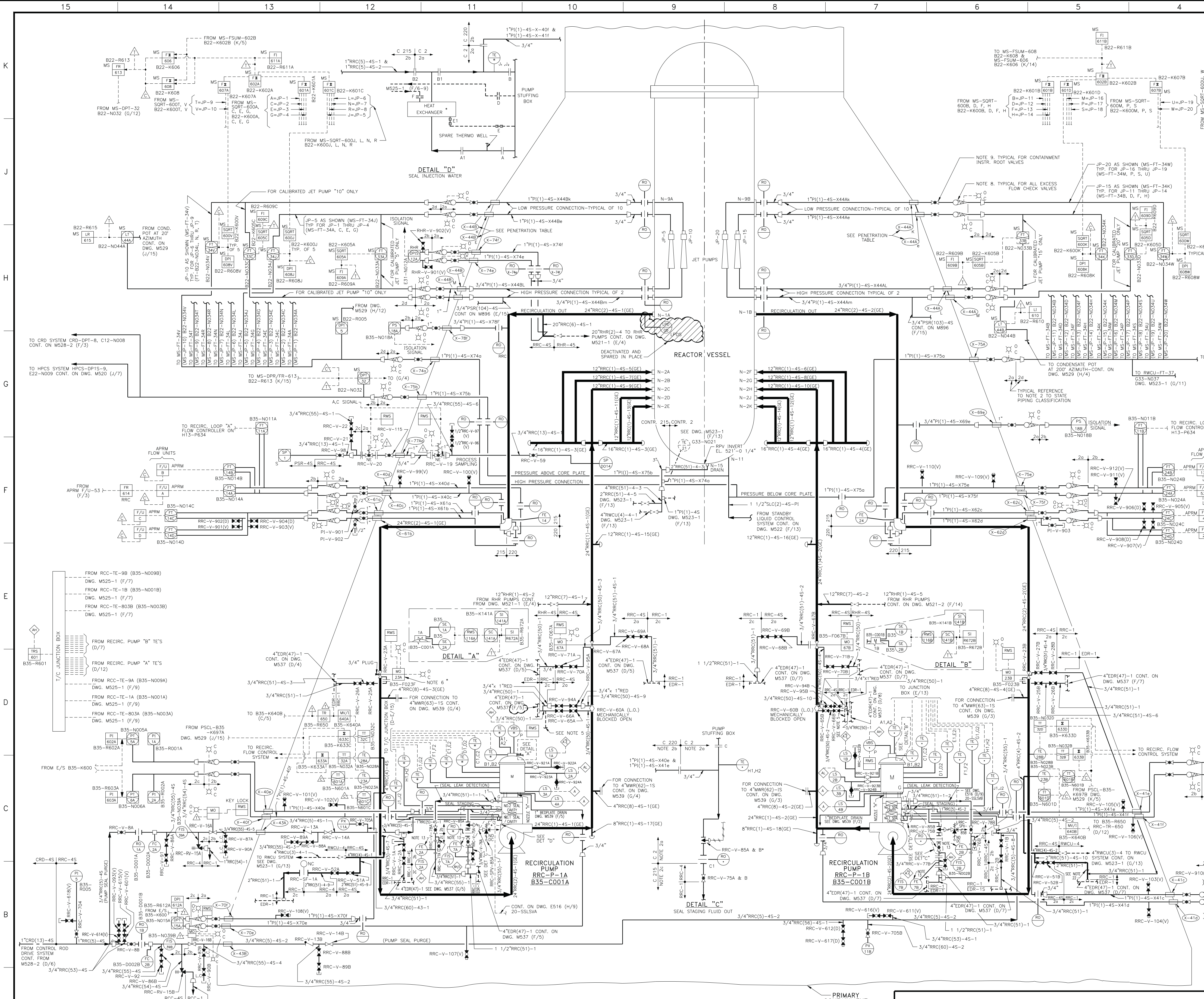
Operating Principle of Jet Pump



Core Flooding Capability of Recirculation System

Draw. No. 960690.60 | Rev. | Figure 5.4-6

Amendment 62
December 2013



NOTES: CONT. FROM (B/2)

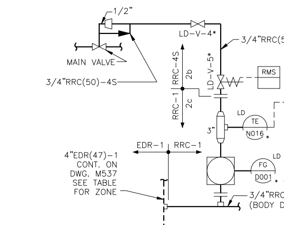
12. CODE BREAK DEFINITIONS FOR THERMOWELLS ARE SHOWN ON M610, "INSTALLATION OF THERMOWELLS AND SAMPLE PROBES".
13. INSTRUMENTATION SO NOTED SHALL BE CODE GROUP E.
14. DELETED.
15. SEE VENDOR DWG 81999-F-001 (CVI 02B35-15,5) FOR INTERNALS INFORMATION & FLOW PATHS ON HY-HP-3A & 3B
16. DELETED.
17. FOR SAMPLE FUNCTION & INSTRUMENTATION ASSOCIATED WITH THE SAMPLE POINTS SEE DWG. SERIES M607 STEAM & LIQUID SAMPLING FLOW DIAGRAMS AND DWG. SERIES 60A-00 SAMPLE RACKS DETAILS & FLOW DIAGRAMS.
(NOTES CONTINUED ON DWG. M530-2)

LEGEND

1. ALL VALVES (EXCEPT THOSE ON INSTRUMENT LINES) SUFFIXED WITH A (V) DENOTE A 3/4" VENT VALVE (EXCEPT AS NOTED).
2. ALL VALVES (EXCEPT THOSE ON INSTRUMENT LINES) SUFFIXED WITH A (D) DENOTE A 3/4" DRAIN VALVE.
3. ALL VALVES SUFFIXED WITH A (TH) DENOTE A THROTTLED VALVE.

NOTES:

1. ALL ITEMS MARKED * ARE FURNISHED WITH ASSOCIATED EQUIPMENT.
2. PIPING, VALVES, AND ASSOCIATED COMPONENTS ON THIS DWG. SHALL BE CLASSIFIED AS FOLLOWS (BREAK POINTS ARE INDICATED ON FLOW DIAGRAM):
   a. PIPING AND VALVES OF SUBSYSTEM RRC (1), (2) AND (7) AND INSTRUMENT AND PROCESS PIPING AND VALVES AS NOTED:
      SEISMIC CATEGORY I
      QUALITY CLASS I
      CODE GROUP A
   b. INSTRUMENT AND PROCESS PIPING AND VALVES AS NOTED.
      SEISMIC CATEGORY I
      QUALITY CLASS I
      CODE GROUP B
   c. INSTRUMENT AND PROCESS PIPING AND VALVES AS NOTED
      SEISMIC CATEGORY II ⊕
      QUALITY CLASS II
      CODE GROUP D } Ø
   d. PUMP SEAL DRAIN LINES AND INSTRUMENT PIPING AS NOTED
      SEISMIC CATEGORY I
      QUALITY CLASS I
      CODE GROUP C
   ⊕ HANGERS TO BE DESIGNED TO SEISMIC CATEGORY I LOADS.
3. ALL INSTRUMENT ROOT VALVES NOT LABELED WILL BE 3/4" GLOBE VALVES UNLESS SPECIFICALLY NOTED OTHERWISE.
4. ALL PIPING SYSTEMS IDENTIFIED BY THE PREFIX "PI" & "PSR" SHALL BE SUPPLIED AND INSTALLED BY CONTRACT #220.
5. DETAIL SHOWS VALVE STEM LEAK-OFF CONNECTION WITH LEAK DETECTION.



THE FOLLOWING VALVES ARE PROVIDED WITH THIS ARRANGEMENT.

MAIN VALVE	*TE AND LEAK-OFF VALVE SUFFIX	**FG SUFFIX	ZONE	UNIQUE LINE No.
RRC-V-60A	G	B-3	(D/10)	3/4"RRC(50)-4S-5
				3/4"RRC(50)-4S-7
RRC-V-60B	H	B-4	(D/8)	3/4"RRC(50)-4S-6
				3/4"RRC(50)-4S-8

JET PUMP TRANSMITTER L.P. SENSING LINE PENETRATION TABLE

UNIQUE LINE No.	PUMP No.	INSTR. PENETR. No.	FT-No.	FT-No.
PI(1)-4S-X44Ba	MS-JP-1	X-44Ba	MS-FT-34A	–
PI(1)-4S-X44Bb	MS-JP-2	X-44Bb	MS-FT-34C	–
PI(1)-4S-X44Bc	MS-JP-3	X-44Bc	MS-FT-34E	–
PI(1)-4S-X44Bd	MS-JP-4	X-44Bd	MS-FT-34G	–
SEE (J/11)	MS-JP-5	X-44Be	MS-FT-34J	MS-FT-33A
PI(1)-4S-X44Bf	MS-JP-6	X-44Bf	MS-FT-34L	–
PI(1)-4S-X44Bg	MS-JP-7	X-44Bg	MS-FT-34N	–
PI(1)-4S-X44Bh	MS-JP-8	X-44Bh	MS-FT-34R	–
PI(1)-4S-X44Bj	MS-JP-9	X-44Bj	MS-FT-34T	–
SEE (J/11)	MS-JP-10	X-44Bk	MS-FT-34V	MS-FT-33C
PI(1)-4S-X44Aa	MS-JP-11	X-44Aa	MS-FT-34B	–
PI(1)-4S-X44Ab	MS-JP-12	X-44Ab	MS-FT-34D	–
PI(1)-4S-X44Ac	MS-JP-13	X-44Ac	MS-FT-34F	–
PI(1)-4S-X44Ad	MS-JP-14	X-44Ad	MS-FT-34H	–
SEE (J/7)	MS-JP-15	X-44Ae	MS-FT-34K	MS-FT-33B
PI(1)-4S-X44Af	MS-JP-16	X-44Af	MS-FT-34M	–
PI(1)-4S-X44Ag	MS-JP-17	X-44Ag	MS-FT-34P	–
PI(1)-4S-X44Ah	MS-JP-18	X-44Ah	MS-FT-34S	–
PI(1)-4S-X44Aj	MS-JP-19	X-44Aj	MS-FT-34U	–
SEE (J/7)	MS-JP-20	X-44Ak	MS-FT-34W	MS-FT-33D

6. THIS EQUIPMENT IS NORMALLY CONTROLLED FROM THE MAIN CONTROL ROOM. IF THE MAIN CONTROL ROOM MUST BE VACATED, THIS CONTROL POINT MAY BE ISOLATED AND CONTROL TRANSFERRED TO THE REMOTE SHUT-DOWN PANEL C61-P001.
7. SEE DWG. M530-2 FOR RRC MISCELLANEOUS SYSTEMS.
8. EXCESS FLOW CHECK VALVES SHALL BE TAGGED AS FOLLOWS:
   PI-EFC-X(PENETRATION NO.)
9. CONTAINMENT INSTRUMENTATION ROOT VALVE SHALL BE TAGGED AS FOLLOWS:
   PI-V-X(PENETRATION NO.)
   ØEXCEPT AS NOTED ON M619, INSTRUMENT CONNECTION DIAGRAMS.
10. ALL INSTRUMENTATION ON THIS DRAWING TO BE IDENTIFIED BY PREFIX "RRC" UNLESS SPECIFICALLY NOTED OTHERWISE.
11. ALL INSTR. LINES ARE DESIGNATED BY CODE & LINE #
   A. EXAMPLE: LINES ORIGINATING INSIDE CONTAINMENT
      $\underbrace{\text{1/2" PI(1)-4S}}_{\text{CODE DESIGNATION}}$ – $\underbrace{\text{X106}}_{\text{LINE NUMBER}}$
   B. EXAMPLE: CONTINUATION OF THE ABOVE LINE OUTSIDE CONTAINMENT IS DESIGNATED AS FOLLOWS, WITH LINE NUMBERS NOT SHOWN.
      $\underbrace{\text{1/2"PI(1)-ST-X106}}_{\text{CODE DESIGNATION}}$ – (H22-P021)- LINE NUMBER

NOTES CONT. AT ZONE (K/4)

REFERENCES TO CONTRACT NUMBERS, SUCH AS 215, 206, 220, FOUND IN THE NOTES ON THIS DRAWING REFER TO ORIGINAL CONSTRUCTION DESIGN REQUIREMENTS.

CURRENT DESIGN REQUIREMENTS ARE DOCUMENTED IN THE STATION DESIGN SPECIFICATIONS AND ACTIVE CONTRACT SPECIFICATIONS WHICH STILL APPLY.

Columbia Generating Station
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Reactor Recirculation System - P&ID

Draw. No. M530-1 | Rev. 89 | Figure 5.4-7.1

NOTES: (CONT. FROM M530-1)

18. ALL SYSTEM COMPONENTS, INCLUDING SUPPORTS, PIPING, AND EQUIPMENT REFERRING TO THIS NOTE SHALL BE CLASSIFIED AS FOLLOWS:

    SEISMIC CATEGORY II
    QUALITY CLASS 2

19. PIPING BETWEEN DRIVE COOLING SKID AND HEAT EXCHANGER IS:

    SEISMIC CATEGORY II
    QUALITY CLASS 2
    CODE GROUP D

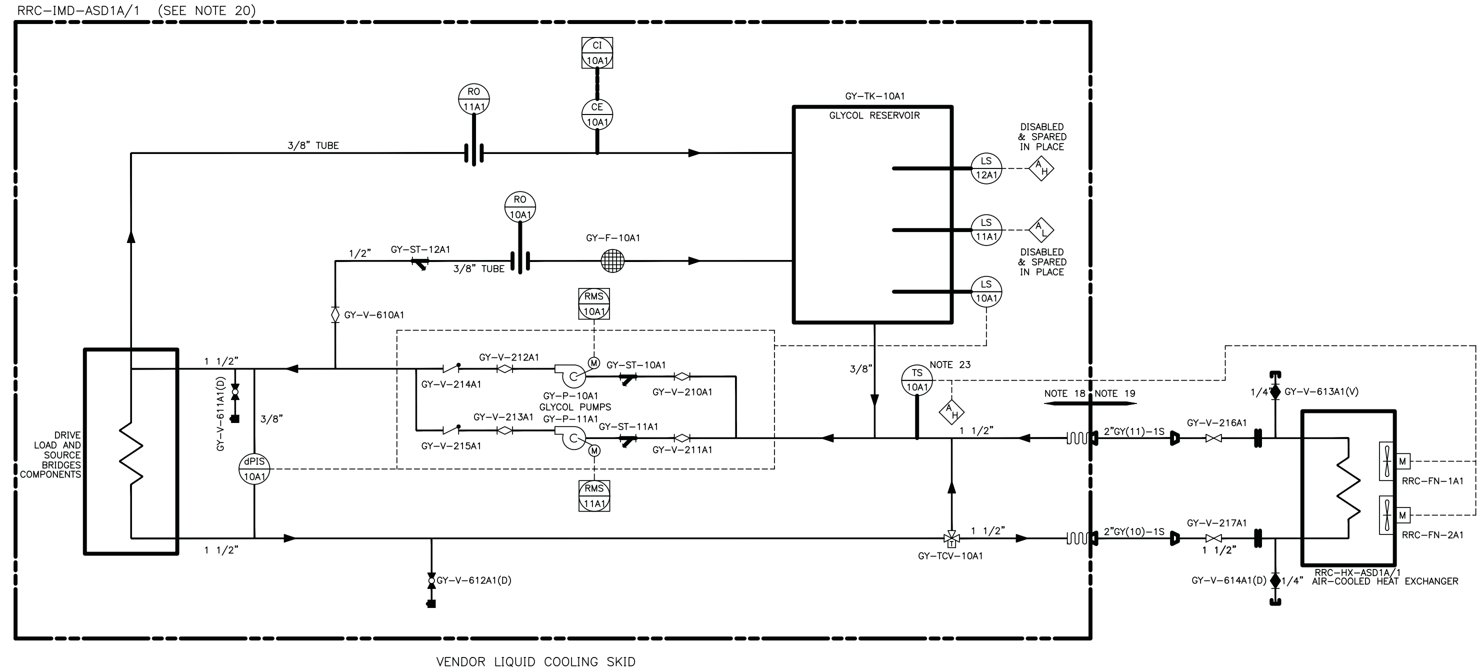
20. EPN'S SHOWN ARE FOR ONE OF FOUR DRIVE LINEUPS. THE APPLICABLE SUFFIX FOR ALL EPN'S IS AS FOLLOWS:

	DRIVE	COMPONENT EPN
SHOWN –	RRC-IMD-ASD1A/1	GY-X-XXA1
	RRC-IMD-ASD1A/2	GY-X-XXA2
	RRC-IMD-ASD1B/1	GY-X-XXB1
	RRC-IMD-ASD1B/2	GY-X-XXB2

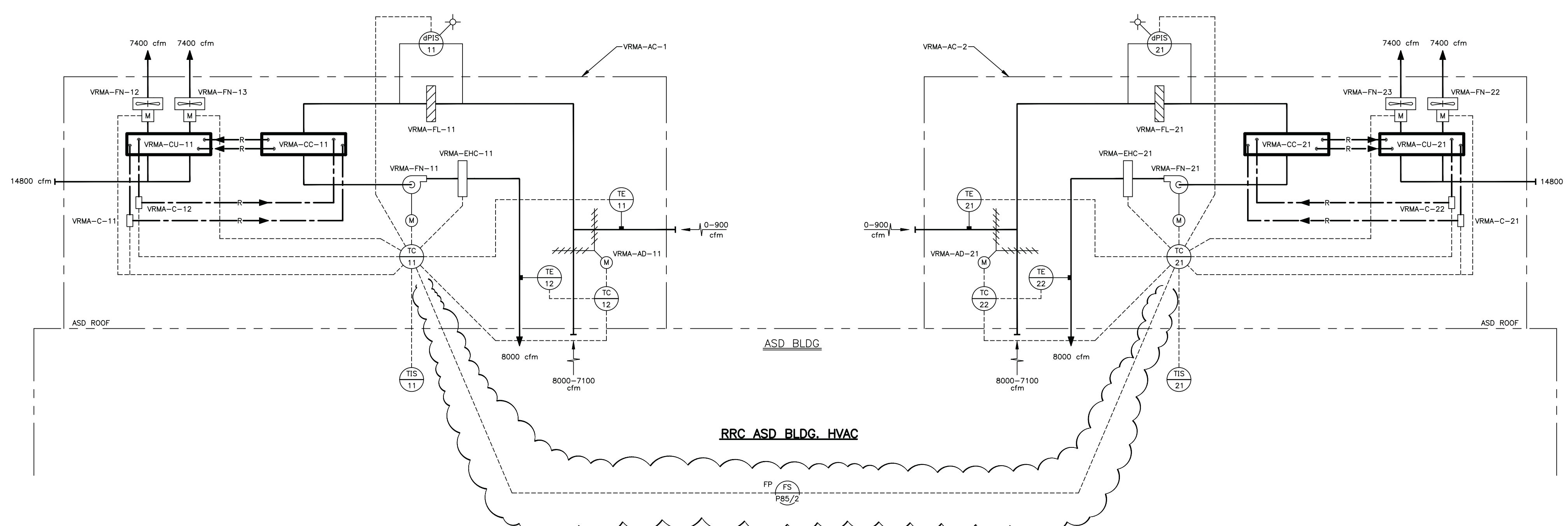
21. LIQUID COOLANT IS DISTILLED OR DE-IONIZED WATER AND PURE PROPYLENE GLYCOL. DO NOT USE RUST-INHIBITING GLYCOL SOLUTIONS.

22. PIPING IS BLANKED OFF DOWNSTREAM OR PRC-V-52B AND THAXTON EXPANDABLE PLUG IS INSTALLED BETWEEN VALVE AND PLATE.

23. THERE IS AN INDUCTIVE MOTOR DRIVE (IMD) TRIP FUNCTION ASSOCIATED WITH TEMPERATURE SWITCHES GY-TS-10A1, -10A2, -10B1, AND -10B2.



INDUCTION MOTOR DRIVES LIQUID COOLING SYSTEM – RRC PUMP ADJUSTABLE SPEED DRIVES
(SEE NOTES 20 & 21)

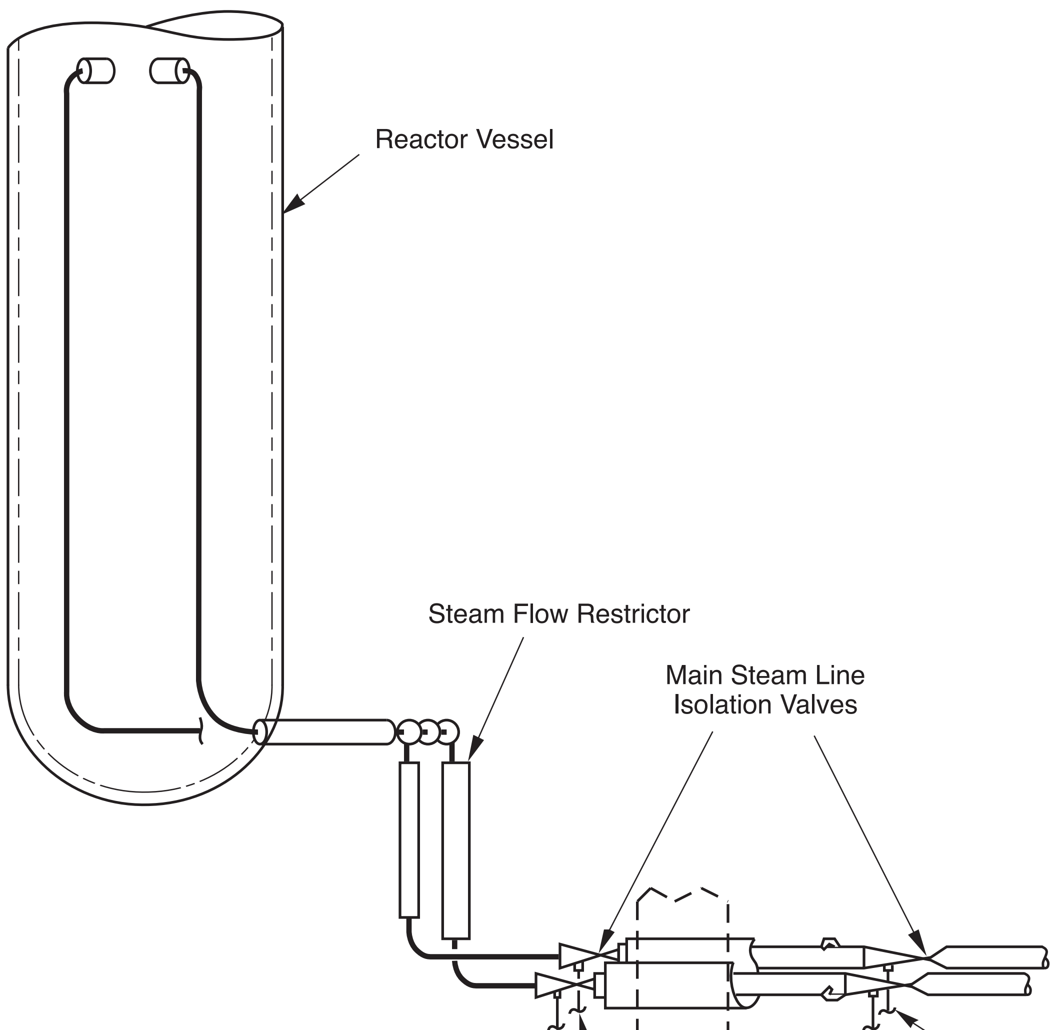


RRC ASD BLDG. HVAC

Columbia Generating Station
Final Safety Analysis Report

Reactor Recirculation System - P&ID

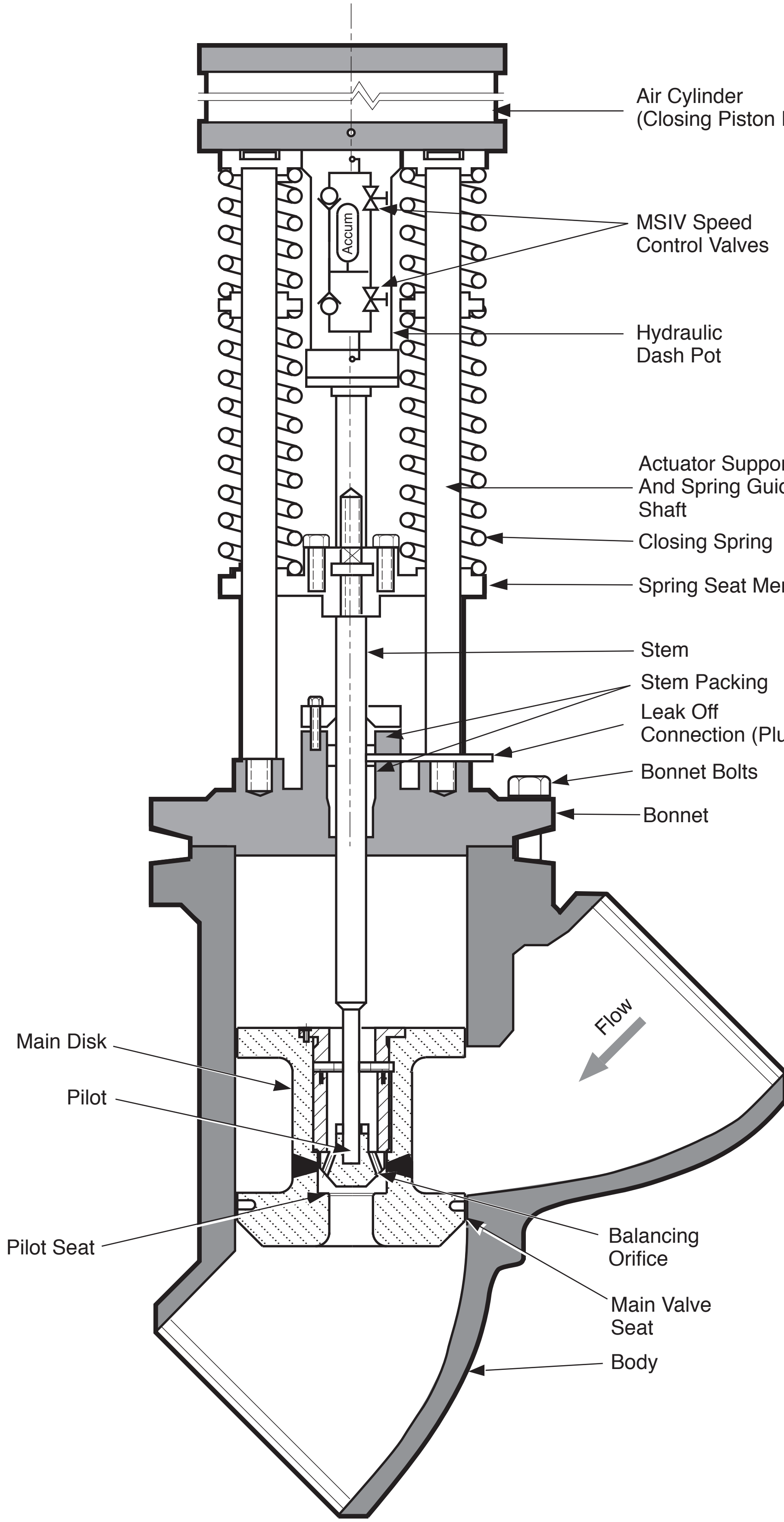
Draw. No. M530-2 | Rev. 7 | Figure 5.4-7.2

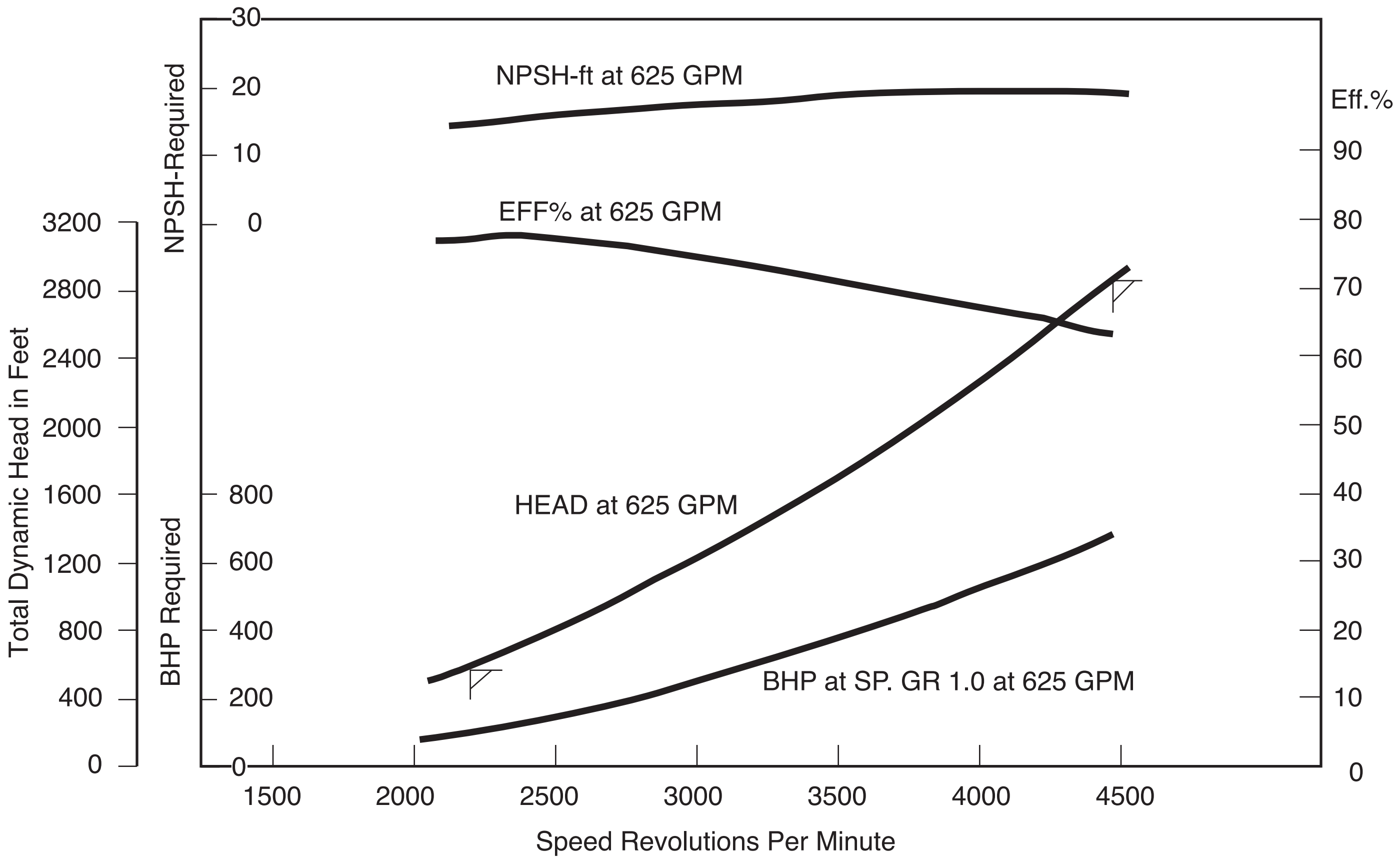


Columbia Generating Station
Final Safety Analysis Report

Main Steam Line Flow Restrictor Location

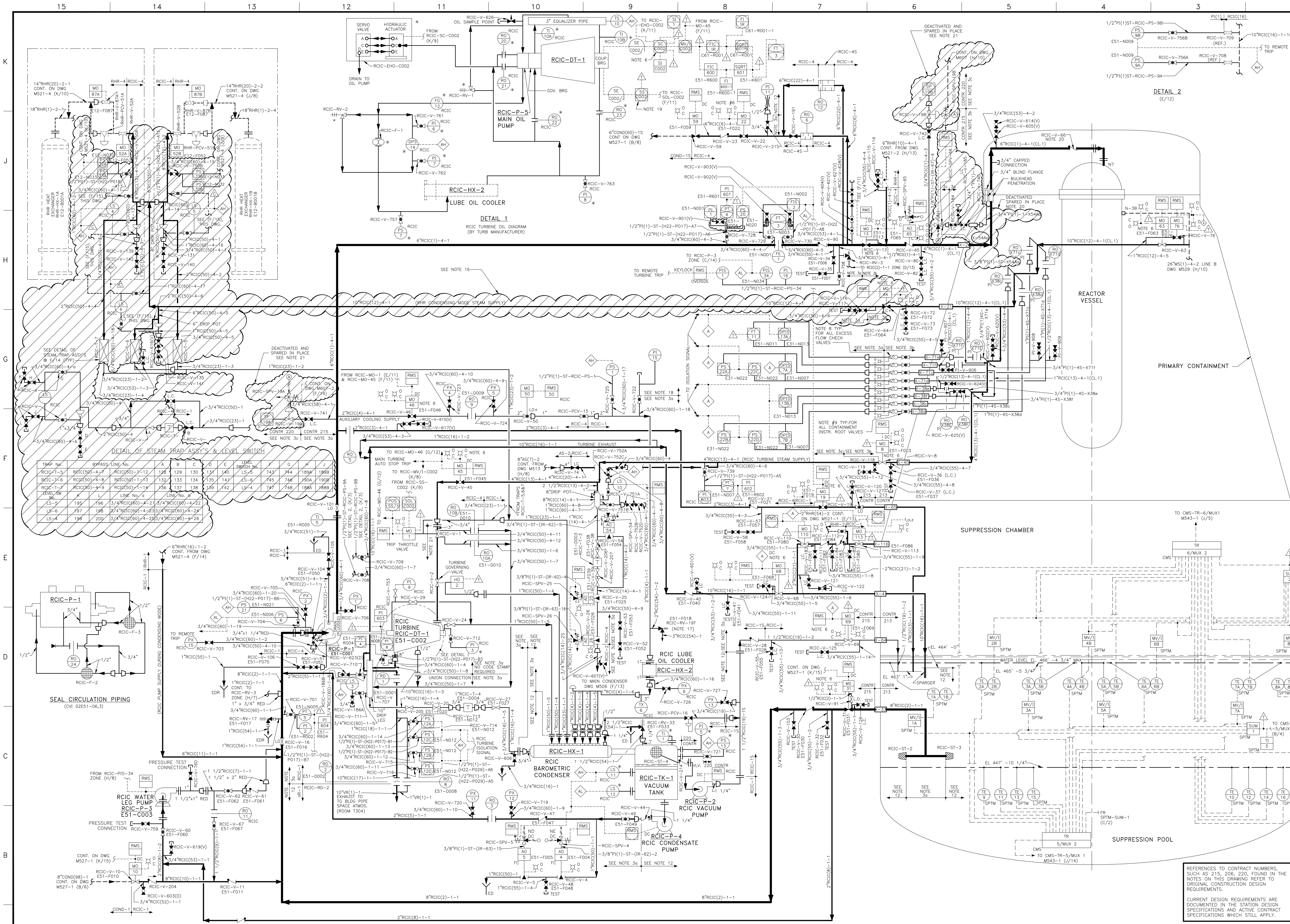
Draw. No. 960690.61	Rev.	Figure 5.4-8


Air Cylinder
(Closing Piston Inside)
Accum
MSIV Speed
Control Valves
Hydraulic
Dash Pot
Actuator Support
And Spring Guide
Shaft
Closing Spring
Spring Seat Member
Stem
Stem Packing
Leak Off
Connection (Plugged)
Bonnet Bolts
Bonnet
Flow
Main Disk
Pilot
Pilot Seat
Balancing
Orifice
Main Valve
Seat
Body



Witness Test Performance
Bingham-Willamette Co.
Portland, Oregon

Columbia Generating Station Final Safety Analysis Report	RCIC Pump Performance Curve (Constant Flow)		
	Draw. No. 960690.55	Rev.	Figure 5.4-10



NOTES:

1. ALL INSTRUMENT ROOT VALVES NOT LABELED WILL BE 3/4" GLOBE VALVES UNLESS SPECIFICALLY NOTED OTHERWISE.
2. ALL ITEMS MARKED * ARE FURNISHED WITH ASSOCIATED EQUIPMENT.
3. PIPING, VALVES, AND ASSOCIATED COMPONENTS ON THIS DRAWING SHALL BE CLASSIFIED AS FOLLOWS EXCEPT AS STATED IN NOTES 12 AND 13: (BREAK POINTS ARE INDICATED ON THIS DIAGRAM.)

   a. ALL PIPING AND VALVES, EXCEPT AS NOTED

      SEISMIC CATEGORY I
      QUALITY CLASS I
      CODE GROUP B*

   *FOR PORTIONS OF SYSTEM NOT REQUIRING CODE STAMP, SYSTEM SHALL BE DESIGNED, FABRICATED, INSTALLED, AND INSPECTED PER CODE GROUP B REQUIREMENTS. TESTING SHALL BE PER CODE GROUP D REQUIREMENTS.

   b. PORTIONS OF SUBSYSTEM RCIC (1), (12) AND (13) AND PI (1) AS NOTED

      SEISMIC CATEGORY I
      QUALITY CLASS I
      CODE GROUP A

   c. INSTRUMENTATION AIR LINES, AND LEAK-OFF PIPING OUTSIDE CONTAINMENT ISOLATION VALVES:

      SEISMIC CATEGORY II⊕
      QUALITY CLASS II
      CODE GROUP D
      } Ø ⊕ HANGERS TO BE DESIGNED TO SEISMIC CATEGORY I LOADS.

   d. INSTRUMENTATION AIR LINES AND LEAK-OFF PIPING INSIDE CONTAINMENT ISOLATION VALVES:

      SEISMIC CATEGORY I
      QUALITY CLASS I
      CODE GROUP B

   Ø EXCEPT AS NOTED ON M619, INSTRUMENT CONNECTION DIAGRAMS

4. (DELETED)
5. ALL PIPING SYSTEMS IDENTIFIED BY THE PREFIX "PI" SHALL BE SUPPLIED AND INSTALLED BY CONTRACT #220.
6. THIS EQUIPMENT IS NORMALLY CONTROLLED FROM THE MAIN CONTROL ROOM. IF THE MAIN CONTROL ROOM MUST BE VACATED, THIS CONTROL POINT MAY BE ISOLATED AND CONTROL TRANSFERRED TO THE REMOTE SHUTDOWN PANEL C61-P001.
7. (DELETED)
8. EXCESS FLOW CHECK VALVES SHALL BE TAGGED AS FOLLOWS: PI-EFC-X (PENETRATION NO.)
9. CONTAINMENT INSTRUMENTATION ROOT VALVES SHALL BE TAGGED AS FOLLOWS: PI-V-X (PENETRATION NO.) EXCEPT WHERE NOTED ON FACE OF DRAWING.
10. (DELETED)
11. EQUIPMENT AND INSTRUMENTS ARE PREFIXED WITH RCIC UNLESS OTHERWISE NOTED.
12. PIPING, VALVES, AND ASSOCIATED COMPONENTS ON THIS DRAWING SHALL BE CLASSIFIED AS FOLLOWS: (BREAK POINTS ARE INDICATED ON THIS DIAGRAM)

    SEISMIC CATEGORY I
    QUALITY CLASS I
    CODE GROUP D*

    *STRAINERS SHALL HAVE ALL FABRICATION AND MATERIALS TO APPLICABLE REQUIREMENTS OF ASME III-2 EXCEPT CODE STAMP IS NOT REQUIRED AND PERFORATED PLATE MAY BE ASTM MATERIAL AND COMPLIANCE WITH NA-3700 IS NOT MANDATORY. FURNISHED AND INSTALLED BY CONTRACT 213A.

13. ALL PIPING DOWN STREAM OF THE LAST ISOLATION VALVE AND OPEN TO THE ATMOSPHERE WITH THE SUBSYSTEM DESIGNATION "SYSTEM" (50) THRU (59) SHALL BE CLASSIFIED AS CODE GROUP D WITH QUALITY CLASS AND SEISMIC CATEGORY PROVIDED BY THE APPLICABLE NOTES ON THIS DRAWING.
14. ALL INSTRUMENT LINES ARE DESIGNATED BY CODE AND LINE NUMBER.

    a. EXAMPLE: LINES ORIGINATING OUTSIDE OF CONTAINMENT

       $\underbrace{\text{1/2"PI(1)-ST}}_{\text{CODE DESIGNATION}} - \underbrace{\text{(H22-P021)-A6}}_{\text{LINE NUMBER}}$

    b. EXAMPLE: LINES ORIGINATING INSIDE OF CONTAINMENT

       $\underbrace{\text{1/2"PI(1)-4S}}_{\text{CODE DESIGNATION}} - \underbrace{\text{X-106}}_{\text{LINE NUMBER}}$

    c. EXAMPLE: CONTINUATION OF THE ABOVE LINE OUTSIDE CONTAINMENT IS DESIGNATED AS FOLLOWS, WITH LINE NUMBERS NOT SHOWN.

       $\underbrace{\text{1/2"PI(1)-ST - X106}}_{\text{CODE DESIGNATION}} - \underbrace{\text{(H22-P021)-}}_{\text{LINE NUMBER}}$

15. CODE BREAK DEFINITIONS FOR THERMOWELLS ARE SHOWN ON M610, "INSTALLATION OF THERMOWELLS AND SAMPLE PROBES".
16. ALL ITEMS IN CROSSHATCHED AREA ARE PART OF THE RHR STEAM CONDENSING MODE WHICH HAS BEEN DEACTIVATED. THE PIPING, VALVES, AND INSTRUMENTATION SHOWN ON THIS DIAGRAM ARE INSTALLED. MOTOR OPERATED VALVE RCIC-V-64 HAS BEEN DE-ENERGIZED AND LOCKED CLOSED. (REF. PED. 215-M-X467). BDC 86-0501-0A REMOVED ALL CONTROL ROOM PANEL WIRING, REMOVED AND SPARED PANEL INSTRUMENTATION, LIFTED AND SPARED ALL FIELD CABLES, AND SPARED IN PLACED ALL PROCESS INSTRUMENTATION.
17. RCIC-RV-19T IS A PERMANENT VALVE. IT WILL RETAIN THE "T " DESIGNATION TO ALLOW DOCUMENT TRACEABILITY.
18. PIPING, VALVES, AND ASSOCIATED COMPONENTS ON THIS DRAWING SHALL BE CLASSIFIED AS FOLLOWS: (BREAK POINTS ARE INDICATED ON THIS DIAGRAM)

    SEISMIC CATEGORY II
    QUALITY CLASS II⊕
    CODE GROUP D +
    HANGERS TO BE ⊕ DESIGNED TO SEISMIC CATEGORY I LOADS

19. ELECTRONIC OVERSPEED TRIP HAS BEEN DE-ACTIVATED PER MDC 95-0268-0A.
20. THE AIR OPERATOR (RCIC-A0-66) FOR RCIC-V-66 HAS BEEN REMOVED AND THE AIR SUPPLY LINE REMOVED BACK TO THE BULKHEAD AND A BLIND FLANGE INSTALLED. RCIC-SPV-66 (LOCATED IN IR-71) IS DEACTIVATED AND SPARED IN PLACE.
21. SAMPLE POINT IS NO LONGER FUNCTIONAL. ALTERNATE METHOD OF SAMPLING IS DONE PER THE APPLICABLE PPM.

LEGEND:

1. ALL VALVES (EXCEPT THOSE ON INSTRUMENT LINES) SUFFIXED WITH A (V) DENOTE A 3/4" VENT VALVE EXCEPT WHERE NOTED.
2. ALL VALVES (EXCEPT THOSE ON INSTRUMENT LINES) SUFFIXED WITH A (D) DENOTE A 3/4" DRAIN VALVE.
3. ALL VALVES SUFFIXED WITH A (TH) DENOTE A THROTTLED VALVE.

REFERENCES TO CONTRACT NUMBERS, SUCH AS 215, 206, 220, FOUND IN THE NOTES ON THIS DRAWING REFER TO ORIGINAL CONSTRUCTION DESIGN REQUIREMENTS.

CURRENT DESIGN REQUIREMENTS ARE DOCUMENTED IN THE STATION DESIGN SPECIFICATIONS AND ACTIVE CONTRACT SPECIFICATIONS WHICH STILL APPLY.

Columbia Generating Station
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RCIC System - P&ID

Draw. No. M519	Rev. 95	Figure 5.4-11

(SEE NOTE 12 & 13)

LOCATION ⬡	1-2	3-3A	3A-4	7-2	8-8A	8A-9	3-12	12-13	13-14	15	17	18-2	19-19A	19A-6	20-20A	20A-1	10-11
PEAK PRESSURE PSIG	N/A	1481	1481	N/A	1337	1337	1481	N/A	N/A	N/A	N/A	N/A	1481	N/A	1481	N/A	160
RECOMMENDED DESIGN PRESSURE	100	1500	1500	100	1250	1250	1500	100	100	75	75	100	1500	100	1500	100	150
DESIGN TEMPERATURE °F	170	170	575	170	575	575	170	170	170	170	250	190	190	170	170	170	267
ESTIMATED LINE SIZE, IN.	6	6	6	6	10	4	2	2	2	1	1 1/4	2	2	2	4	4	10

15. MAXIMUM LINEAR LENGTH BETWEEN POSITIONS (5) AND (22) EQUALS 60 FEET. THE INTERNAL CROSS-SECTIONAL AREA OF THE VACUUM BREAKER PIPING THAT IS SHARED BY THE RCIC AND RHR SHALL BE EQUAL TO OR GREATER THAN THE COMBINED INTERNAL CROSS-SECTIONAL AREAS OF THE INDIVIDUAL RCIC AND RHR VACUUM BREAKER LINES.

16. STEAM FLOWS FOR TEST MODE F AT POSITION 9, 10 AND 11 ARE BASED UPON A PUMP TDH OF 2490 FEET.

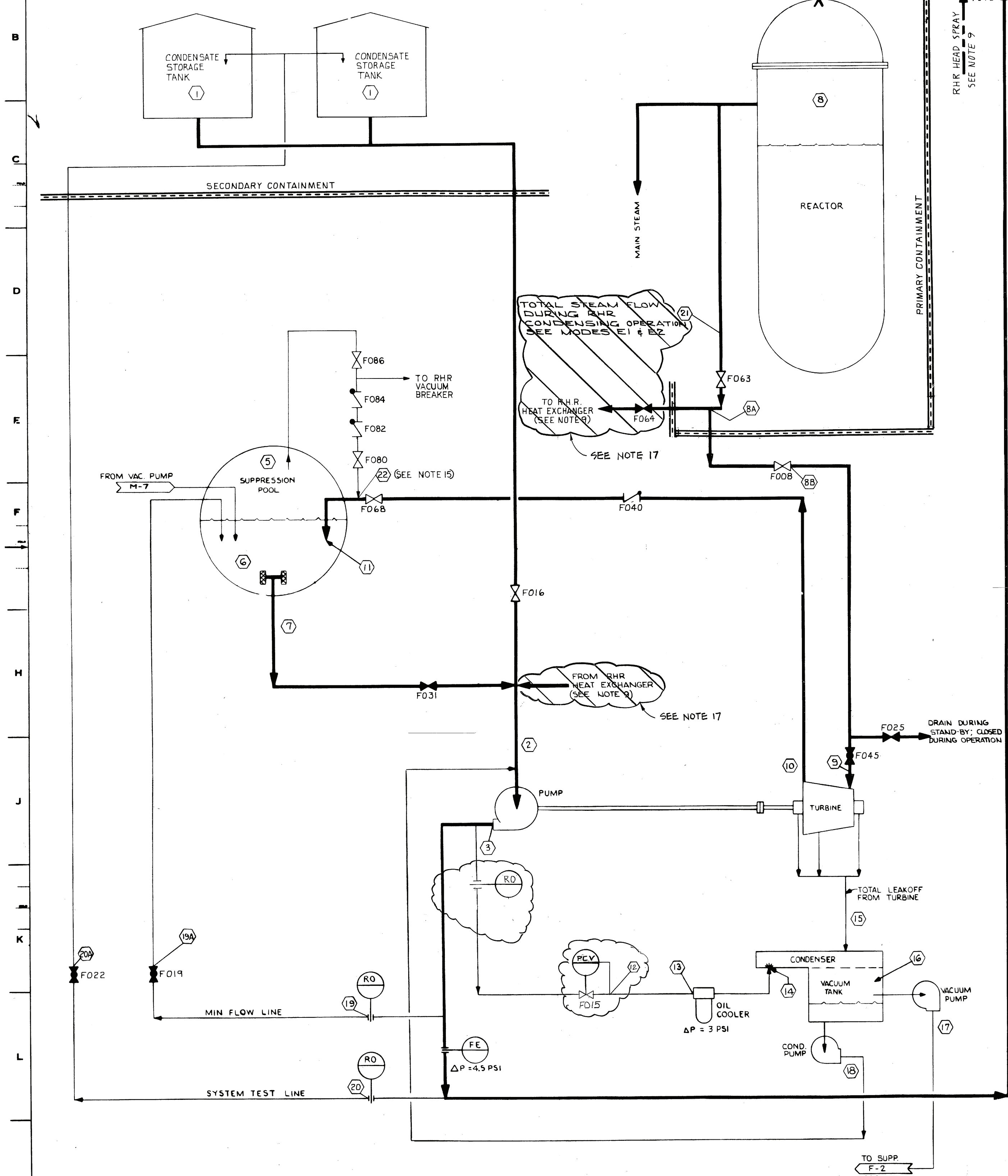
17. RHR STEAM CONDENSING MODE HAS BEEN DEACTIVATED.

~~18. ONLY HIGH PRESSURE MODES WERE REVISED FOR RPU. MODES B&D WOULD NOT BE SIGNIFICANTLY AFFECTED.~~

12. DESIGN PRESSURES AND TEMPERATURES GIVEN ARE THE BASIS FOR DESIGN OF GE SUPPLIED EQUIPMENT. ESTIMATED LINE SIZES ARE FOR INFORMATION ONLY. ACTUAL LINE SIZES AS DETERMINED BY PIPING DESIGNER, SHALL MEET THE PROCESS DATA HYDRAULIC REQUIREMENTS.

13. "PEAK PRESSURE" IS THE MAXIMUM PRESSURE ANTICIPATED DURING A TRANSIENT PERIOD WITH ALL OF THE CONTRIBUTING ELEMENTS AT A MAXIMUM. IT WOULD BE EXPECTED TO OCCUR LESS THAN 1% OF SYSTEM OPERATING TIME.

14. FLOW VALUES SHOWN IN MODES C & D ARE BASED UPON SUCTION PIPING DESIGN PERMITTING THE MINIMUM REQUIRED NPSH TO CONTINUE TO BE PROVIDED TO THE RCIC PUMP WHEN THE SUPPRESSION POOL SUCTION STRAINER IS 50% PLUGGED.



* THE PRESSURE AT THIS LOCATION DEPENDS ON PIPING ARRANGEMENT, AND MAY BE VARIED WITHIN THE FOLLOWING LIMITS.

LOCATION

- (2) MINIMUM NPSH AT PUMP SUCTION = 21 FEET
- (3) MAXIMUM PUMP TOTAL DYNAMIC HEAD 3016 FEET FOR MODES A & C; 610 FEET FOR MODES B & D
- (9) MAXIMUM PRESSURE DROP BETWEEN LOCATION (8) AND (9) = 15 PSI (SEE NOTE 6)
- (10) MAXIMUM PRESSURE ALLOWED = 25 PSIA
- (13) MAXIMUM PRESSURE ALLOWED = 75 PSIA
- (15) SUFFICIENT VACUUM TO PREVENT TURBINE SHAFT-OUT-LEAKAGE.
- (17) MAXIMUM PRESSURE AVAILABLE = 25 PSIA
- (18) MAXIMUM PRESSURE AVAILABLE = 65 PSIA
- (19) SUFFICIENT PRESSURE TO RETURN TO SUPPRESSION POOL
- (20) SUFFICIENT PRESSURE TO RETURN TO COND. STORAGE

NOTES:

MPL ITEM Nº : E51-1020

1. ATMOSPHERIC PRESSURE OF 14.7 PSIA WAS USED IN CALCULATIONS.
2. WATER FLOWS ARE SHOWN IN GPM, STEAM FLOWS IN 1000 LB/HR.
3. THE MAXIMUM POOL WATER TEMPERATURE FOR CONTINUOUS SYSTEM OPERATION WILL NOT EXCEED 140°F. HOWEVER, DUE TO POTENTIAL SHORT TERM OPERATION AT HIGHER TEMPERATURES, PIPING EXPANSION SHALL BE BASED ON 170°F.
4. THE UNRECOVERED FLOW NOZZLE PRESSURE DROP OF 4.5 PSI IS A FIXED LOSS BETWEEN LOCATIONS (3) AND (4)
5. THE LUBE OIL COOLER PRESSURE DROP OF 3.0 PSI IS A FIXED LOSS BETWEEN LOCATIONS (13) AND (14)
6. THE CONTROLLING MODES FOR LINE SIZING AND ARRANGEMENT ARE:
   - SUCTION FROM COND STORAGE - - - - - MODE A & B
   - SUCTION FROM SUPPRESSION POOL - - - - - MODE C & D
   - PUMP DISCHARGE - - - - - MODE C & D
   - STEAM SUPPLY - - - - - MODE A & B
   - TURBINE EXHAUST - - - - - MODE A & C
   - TEST LINE - - - - - MODE F
   - COOLING SYSTEM - - - - - MODE A
7. SYSTEM OPERATION IS POSSIBLE WITH INTERMEDIATE PRESSURES IN THE REACTOR VESSEL AND THE SUPPRESSION POOL. HOWEVER, THESE CONDITIONS DO NOT CONTROL PIPE OR VALVE SIZING OR SPECIFICATION, AND NO DATA IS SHOWN.
8. PUMP MINIMUM FLOW REQUIREMENT MAY OCCUR DURING ANY OPERATING MODE. FLOW REQUIREMENT IS 75 GPM MINIMUM.
9. FOR VALVE AND FLOW INFORMATION, REFER TO THE RHR SYSTEM P&ID MPL E12-1010 AND PROCESS DIAGRAM MPL E12-1020.
10. DURING SYSTEM STANDBY, EQUIPMENT IS NOT OPERATING. INTERMITTENT FLOW OCCURS THROUGH THE STEAM SUPPLY LINE DRAIN TRAP SYSTEM AT 1000 PSIA AND 560°F.
11. HEAD SPRAY NOZZLE PRESSURE DROP IS 10 PSI.

MODE A — SUCTION FROM CONDENSATE STORAGE, REACTOR AT HIGH PRESSURE, SUPPRESSION POOL AT HIGH PRESS.

LOCATION ⬡	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	20
FLOW-SEE NOTE 2	—	600	625	600	—	—	0	—	30.73	30.57	30.57	25	25	25	0.15	—	0.01	25+	—	0
OPERATING PRESSURE-PSIA	14.7	*	*	1235	18	—	—	1225	*	*	19.9	75	*	45	*	9.8	*	*	*	—
EXPECTED TEMPERATURE °F	100	100	100	100	110	—	—	565	SAT	SAT	228	100	100	100	230	120	120	120	100	—
MAX / MIN TEMPERATURE °F	100/40	100/40	100/40	100/40	170/40	—	—	570/70	570/70	250/70	250/70	100/40	100/40	100/40	250/40	120/40	120/40	120/40	100/40	—

MODE B — SUCTION FROM CONDENSATE STORAGE, REACTOR AT LOW PRESSURE, SUPPRESSION POOL AT HIGH PRESS.

LOCATION ⬡	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	20
FLOW-SEE NOTE 2	—	600	616	600	—	—	0	—	9.30	9.15	9.15	16	16	16	0.15	—	0.01	16+	—	0
OPERATING PRESSURE-PSIA	14.7	*	*	178	18	—	—	165	*	*	19.8	75	*	45	*	9.8	*	*	*	—
EXPECTED TEMPERATURE °F	100	100	100	100	140	—	—	366	SAT.	SAT.	227	100	100	100	230	120	120	120	100	—
MAX / MIN TEMPERATURE °F	100/40	100/40	100/40	100/40	170/40	—	—	366/70	366/70	250/70	250/70	100/40	100/40	100/40	250/40	120/40	120/40	120/40	100/40	—

MODE C — SUCTION FROM SUPPRESSION POOL, REACTOR AT HIGH PRESSURE, SUPPRESSION POOL AT LOW PRESS.

LOCATION ⬡	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	20
FLOW-SEE NOTE 2	—	600	625	600	—	—	600	—	29.82	29.66	29.66	25	25	25	0.15	—	0.01	25+	—	0
OPERATING PRESSURE-PSIA	—	*	*	1235	14.7	19.2	18.8	1225	*	*	16.6	75	*	45	*	9.8	*	*	*	—
EXPECTED TEMPERATURE °F	—	140	140	140	140	140	140	565	SAT.	SAT.	218	140	140	140	230	160	160	160	140	—
MAX/MIN TEMPERATURE °F	—	170/40	170/40	170/40	170/40	170/40	170/40	570/70	570/70	250/70	250/70	170/40	170/40	170/40	250/40	190/40	190/40	190/40	170/40	—

MODE D — SUCTION FROM SUPPRESSION POOL, REACTOR AT LOW PRESSURE, SUPPRESSION POOL AT LOW PRESS.

LOCATION ⬡	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	20
FLOW-SEE NOTE 2	—	600	616	600	—	—	600	—	8.40	8.25	8.25	16	16	16	0.15	—	0.01	16+	—	0
OPERATING PRESSURE-PSIA	—	*	*	178	14.7	19.2	18.8	165	*	*	16.5	75	*	45	*	9.8	*	*	*	—
EXPECTED TEMPERATURE °F	—	140	140	140	140	140	140	366	SAT.	SAT.	218	140	140	140	230	160	160	160	140	—
MAX/MIN TEMPERATURE °F	—	170/40	170/40	170/40	170/40	170/40	170/40	366/70	366/70	250/70	250/70	170/40	170/40	170/40	250/40	190/40	190/40	190/40	170/40	—

SEE NOTE 17
MODE. E-1 — SUCTION FROM RHR HEAT EXCHANGERS, REACTOR AT HIGH PRESS., SUPPRESS. POOL AT LOW PRESS.

LOCATION ⬡	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	20	21
FLOW-SEE NOTE 2	—	380	396	380	—	—	0	—	20.15	20.00	20.00	16	16	16	0.15	—	0.01	16+	—	0	207.3
OPERATING PRESSURE-PSIA	—	*	*	1025	14.7	—	—	1015	*	*	16.6	75	*	45	*	9.8	*	*	*	—	1015
EXPECTED TEMPERATURE °F	—	140	140	140	140	—	—	560	SAT.	SAT.	218	140	140	140	230	160	160	160	140	—	560
MAX/MIN TEMPERATURE °F	100/40	170/40	170/40	170/40	170/40	—	—	560/70	560/70	250/70	250/70	170/40	170/40	170/40	250/40	190/40	190/40	190/40	170/40	—	560/70

SEE NOTE 17
MODE E-2 — SUCTION FROM RHR HEAT EXCHANGERS, REACTOR AT LOW PRESS, SUPPRESS. POOL AT LOW PRESS.

LOCATION ⬡	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	20	21
FLOW-SEE NOTE 2	—	215	231	215	—	—	0	—	5.15	5.00	5.00	16	16	16	0.15	—	0.01	16+	—	0	119.6
OPERATING PRESSURE-PSIA	—	*	*	163	14.7	—	—	150	*	*	16.6	75	*	45	*	9.8	*	*	*	—	150
EXPECTED TEMPERATURE °F	—	140	140	140	140	—	—	358	SAT.	SAT.	218	140	140	140	230	160	160	160	140	—	358
MAX/MIN TEMPERATURE °F	100/40	170/40	170/40	170/40	170/40	—	—	358/70	560/70	250/70	250/70	170/40	170/40	170/40	250/40	190/40	190/40	190/40	170/40	—	358/70

F

MODE F — TEST MODE: SUCTION FROM CONDENSATE STORAGE, REACTOR AT HIGH PRESS., SUPPRESS POOL AT LOW PRESS.

LOCATION ⬡	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	20
FLOW-SEE NOTE 2	—	600	616	—	—	—	0	—	23.50	23.35	23.35	16	16	16	0.15	—	0.01	16+	—	600
OPERATING PRESSURE-PSIA	14.7	*	*	—	14.7	—	—	1000	*	*	16.6	75	*	45	*	9.8	*	*	*	*
EXPECTED TEMPERATURE °F	100	100	100	—	100	—	—	545	SAT.	SAT.	218	100	100	100	230	120	120	120	100	100
MAX/MIN TEMPERATURE °F	100/40	100/40	100/40	—	170/40	—	—	545/70	545/70	250/70	250/70	100/40	100/40	100/40	250/40	120/40	120/40	120/40	100/40	100/40

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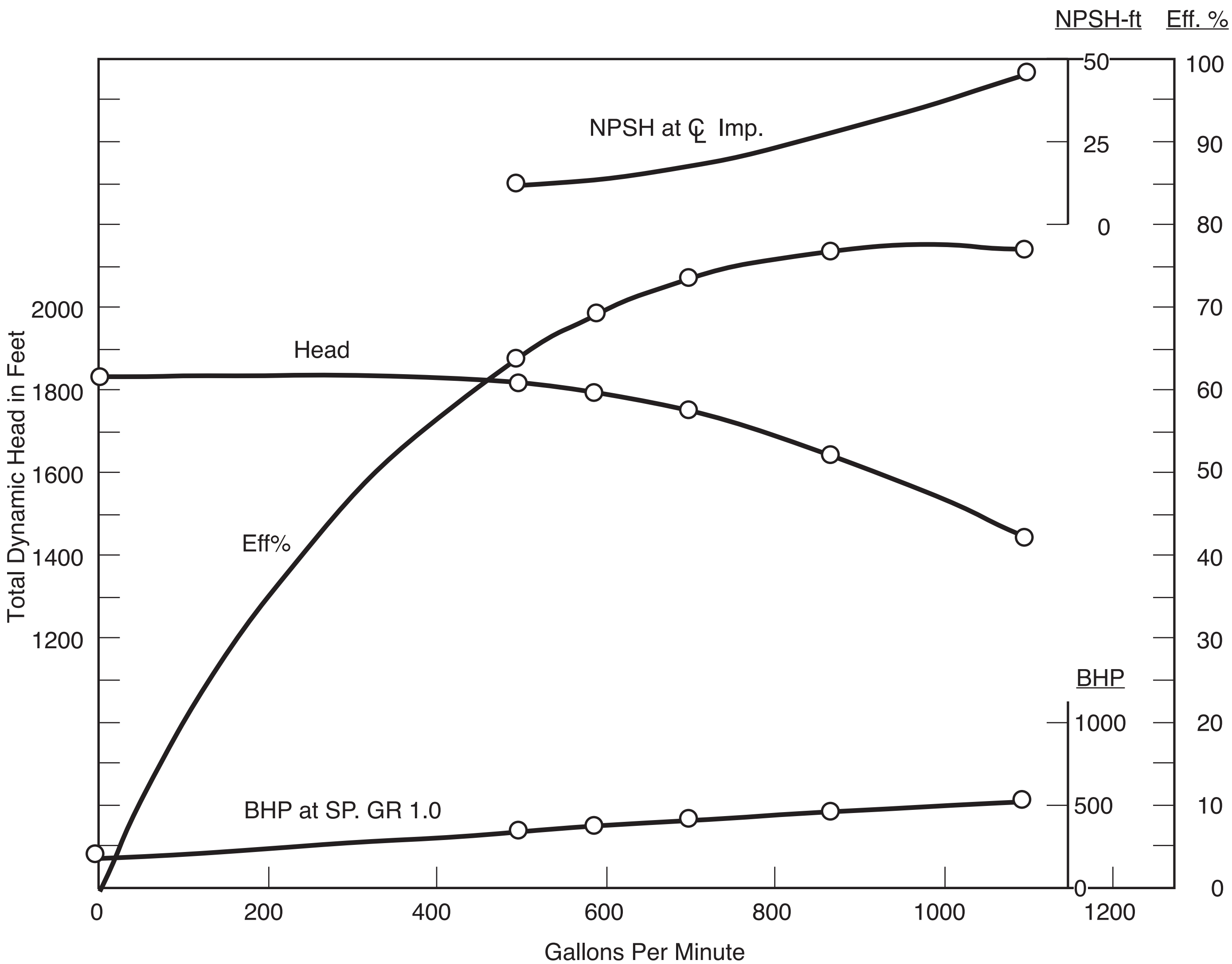
RCIC System Process Diagram

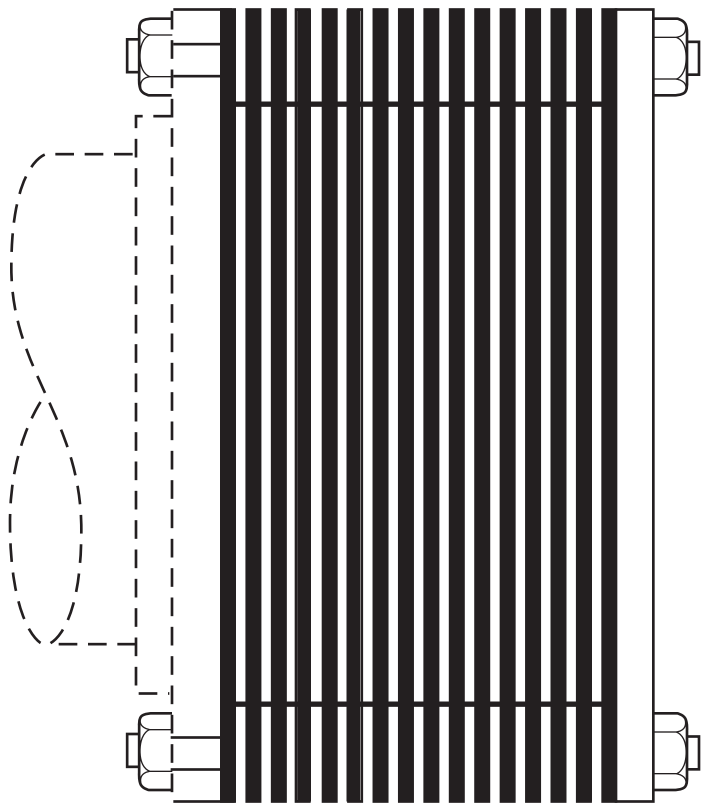
Draw. No. 02E51-04,4,1	Rev. 10	Figure 5.4-12



Test Speed RG. 3591-3585 RPM

Whitness Test Performance
Bingham-Willamette Co.
Portland, Oregon.

Measurements for Strainers at
Penetration X-33
Rated Flow: 600 gal/min



Notes:

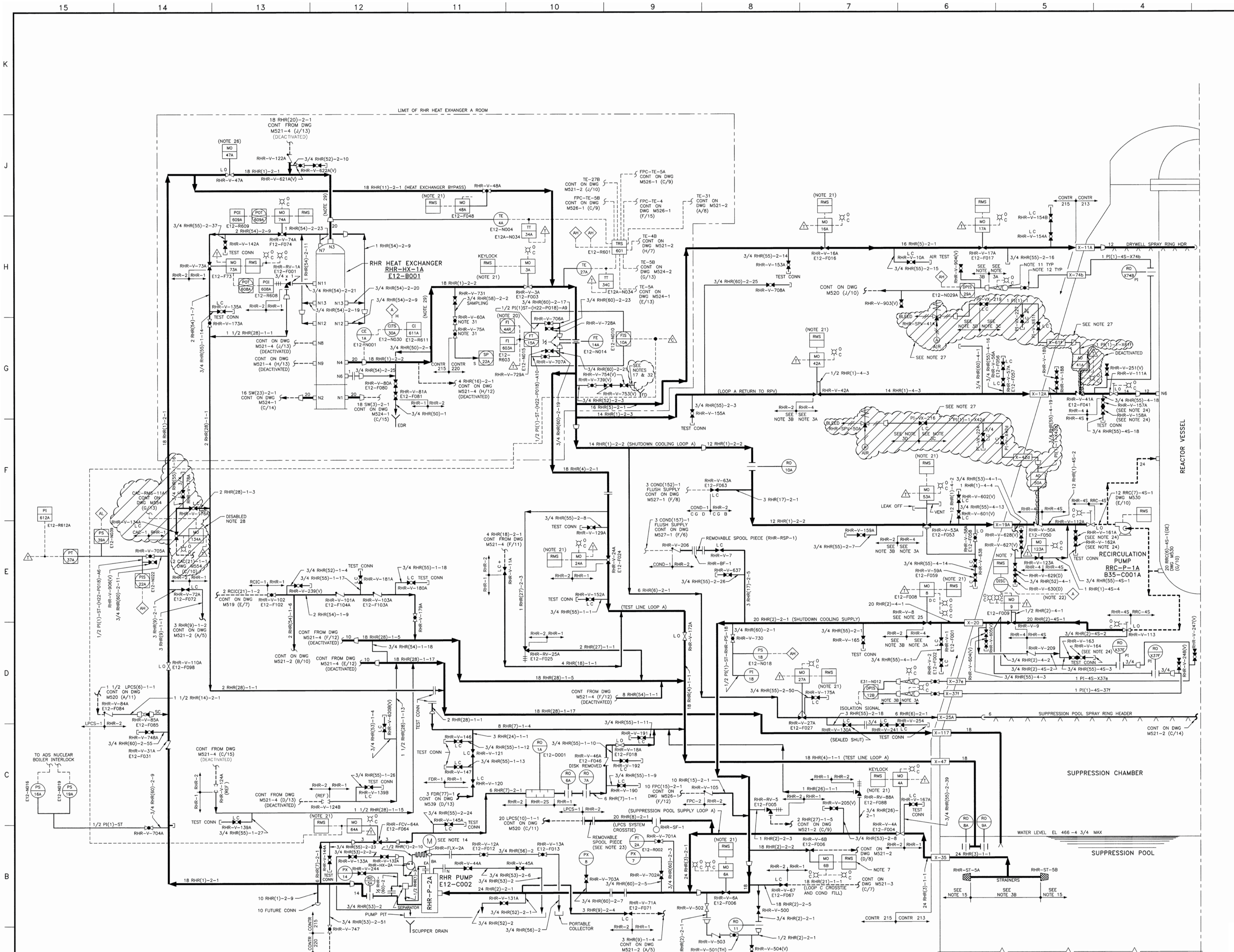
1. Flow stated above is per penetration with two (2) units described above required per penetration.

2. Units are designed, manufactured and inspected in accordance with ASME Section III, Class 2 (not stamped) 1974 Ed. with Addenda thru Winter 1976.

3. Design temp: 220°F

Typical Strainer

Draw. No. 960690.87 | Rev. | Figure 5.4-14



NOTES

1 ALL PRESSURE & FLOW INSTRUMENT ROOT VALVES NOT LABELLED WILL BE 3/4 GLOBE VALVES (UNLESS SPECIFICALLY NOTED OTHERWISE)

2 ALL ITEMS MARKED * ARE FURNISHED WITH ASSOCIATED EQUIPMENT

3 PIPING VALVES AND ASSOCIATED COMPONENTS ON THIS DWG SHALL BE CLASSIFIED AS FOLLOWS EXCEPT AS STATED IN NOTES 15 & 17 (BREAK POINTS ARE INDICATED ON FLOW DIAGRAM)

A PIPING AND VALVES OUT THROUGH OUTERMOST REAC ISOL VALVES
SEISMIC CATEGORY I
QUALITY CLASS I
CODE GROUP A } Ø

B PIPING AND VALVES BEYOND OUTERMOST REAC ISOL VALVES EXCEPT THOSE LINES DESIG AS 3/4 RHR(56)-2
SEISMIC CATEGORY I
QUALITY CLASS I
CODE GROUP B } Ø

C INSTM AIR LINES & LEAK-OFF PIPING INSIDE CONT ISOL VALVES
SEISMIC CATEGORY I
QUALITY CLASS I
CODE GROUP B

D INSTR AIR LINES OUTSIDE CONTAINMENT ISOL VALVES
SEISMIC CATEGORY I
QUALITY CLASS II
CODE GROUP D } Ø

E LEAK-OFF PIPING BEYOND LEAK-OFF ISOL VALVES & THE LINES DESIGNATED AS 3/4 RHR(56)-2
SEISMIC CATEGORY II
QUALITY CLASS II ⊕
CODE GROUP D

F PIPING SUBSYSTEMS EDR(20)-1 & FDR(43)
SEISMIC CATEGORY II
QUALITY CLASS II ⊕
CODE GROUP D+

⊕ HANGERS TO BE DESIGNED TO SEISMIC CATEGORY I LOADS
Ø EXCEPT AS NOTED ON M619 INSTR CONN DIAGRAMS
+ SEE NOTE 12 WNP-2 SPEC SECT 15B 1 TABLE 2 NOTES

4 CODE BREAK DEFINITIONS FOR THERMOWELLS ARE SHOWN ON M610 INSTALLATION OF THERMOWELLS AND SAMPLE PROBES

5 DELETED

6 ALL PIPING SYSTEMS IDENTIFIED BY THE PREFIX PI SHALL BE SUPPLIED & INSTALLED BY CONTRACT #220

7 THIS EQUIPMENT IS NORMALLY CONTROLLED FROM THE MAIN CONTROL ROOM IF THE MAIN CONTROL ROOM MUST BE VACATED THIS CONTROL POINT MAY BE ISOLATED AND CONTROL TRANSFERRED TO THE REMOTE SHUTDOWN PANEL C61-P001

8 DELETED

9 ALLOW ADEQUATE PIPING SURFACE AREA FOR COOLING OF PUMP RHR-P-3

10 ALL TEST CONNECTION PIPING WILL ASSUME RATING OF SOURCE PIPE AND BE IDENTIFIED AS 3/4 RHR(55)- UNLESS OTHERWISE NOTED BY SPECIFIC LINE NUMBER

11 EXCESS FLOW CHECK VALVES SHALL BE TAGGED AS FOLLOWS PI-EFC-X* (*=PENETRATION NUMBER)

12 CONTAINMENT INSTRUMENTATION ROOT VALVES SHALL BE TAGGED AS FOLLOWS PI-VX-* (*=PENETRATION) EXCEPT WHERE NOTED ON FACE OF DRAWING

13 EQUIPMENT AND INSTRUMENTS ARE PREFIXED WITH RHR UNLESS OTHERWISE NOTED

14 SEE COMPUTER I/O LIST FOR MOTOR WINDING AND BEARING TEMPERATURE ELEMENTS AND POINT NUMBERS

15 PIPING VALVES AND ASSOCIATED COMPONENTS ON THIS DWG SHALL BE CLASSIFIED AS FOLLOWS (BREAK POINTS ARE INDICATED ON THIS DIAGRAM)
SEISMIC CATEGORY I
QUALITY CLASS I
CODE GROUP B

(ASME III/2 WITH EXCEPTIONS SEE PROCUREMENT SPECIFICATION 12023)

16 THESE INSTRUMENTS ARE LOCATED ON COLD SHUTDOWN PANEL

17 ALL PIPING DOWNSTREAM OF THE LAST ISOLATION VALVE AND OPEN TO THE ATMOSPHERE WITH THE SUBSYSTEM DESIGNATION SYSTEM (50) THRU (59) SHALL BE CLASSIFIED AS CODE GROUP D WITH QUALITY CLASS AND SEISMIC CATEGORY PROVIDED BY THE APPLICABLE NOTES ON THIS DRAWING

18 ALL INSTRUMENT LINES ARE DESIGNATED BY CODE AND LINE NUMBER

A EXAMPLE LINES ORIGINATING OUTSIDE CONTAINMENT
$\underbrace{\text{1/2 PI(1)-ST}}_{\text{CODE DESIGNATION}} - \underbrace{\text{(H22-P021)-A6}}_{\text{LINE NUMBER}}$

B EXAMPLE LINES ORIGINATING INSIDE CONTAINMENT
$\underbrace{\text{1/2 PI(1)-4S - X106}}_{\text{LINE NUMBER}}$
$\underbrace{\text{1/2 PI(1)-4S}}_{\text{CODE DESIGNATION}}$

C EXAMPLE CONTINUATION OF THE ABOVE LINE OUTSIDE CONTAINMENT IS DESIGNATED AS FOLLOWS (WITH LINE NUMBERS NOT SHOWN)
$\underbrace{\text{1/2 PI(1)-ST-X106}}_{\text{CODE DESIGNATION}} - \underbrace{\text{(H22-P021)-}}_{\text{LINE NUMBER}}$

19 ALL ITEMS IN CROSSHATCHED AREA ARE PART OF THE RHR STEAM CONDENSING MODE WHICH HAS BEEN DEACTIVATED THE PIPING VALVES AND INSTRUMENTATION AS SHOWN ON THE DIAGRAM ARE INSTALLED MOTOR OPERATED VALVES THAT HAVE BEEN DE-ENERGIZED AND LOCKED CLOSED ARE RHR-V-52A & B RHR-V-11A & B RHR-V-26A & B RHR-V-87A & B RHR-V-124A & B RHR-V-125A & B (REF PED 215-M-X467) EXCEPTION CONTROL AND INDICATION FOR RHR-V-26B HAS BEEN REMOVED REMOVED ALL CONTROL ROOM PANEL WIRING REMOVED AND SPARED PANEL INSTRUMENTATION LIFTED AND SPARED ALL FIELD CABLES AND SPARED IN PLACE ALL PROCESS INSTRUMENTATION

20 THESE INSTRUMENTS ARE LOCATED ON THE ALTERNATE REMOTE SHUTDOWN PANEL E-CP-ARS

21 THIS EQUIPMENT IS NORMALLY CONTROLLED FROM THE MAIN CONTROL ROOM IF THE MAIN CONTROL ROOM MUST BE VACATED THIS CONTROL POINT MAY BE ISOLATED AND CONTROL TRANSFERRED TO THE ALTERNATE REMOTE SHUTDOWN PANEL E-CP-ARS

22 POWER HAS BEEN REMOVED FROM THIS MOV BY A MAINTAINED OPEN DISCONNECT SWITCH REMOTE FROM THE MAIN CONTROL ROOM WITH THE VALVE MAINTAINED CLOSED ALL AUTOMATIC INTERLOCKS STILL FUNCTION WHEN THE MOTOR FEEDER POWER SWITCH IS CLOSED CONTROL OF THIS VALVE IS RE-ESTABLISHED AND A HI/LOW RPV PRESSURE INTERFACE ALARM FOR MAIN CONTROL ROOM OPERATION DURING A DESIGN BASIS FIRE IS PROVIDED FOR OPERATOR INFORMATION

23 RHR-LPCS CROSSTIE SPOOL PIECE IS NOT INSTALLED

24 THESE VALVES ARE 1/2 VALVES BUTT WELDED TO SPECIALLY PREPARED 3/4 NIPPLES

25 1/4 HOLE DRILLED ON REACTOR SIDE OF RHR-V-8 WEDGE

26 VALVES RHR-V-47A & B HAVE BEEN DEACTIVATED THE PIPING AND VALVES AS SHOWN ON THIS DIAGRAM ARE INSTALLED THE ELECTRICAL POWER CONTROL ROOM AND REMOTE SHUTDOWN CONTROL SWITCHES PANEL WIRING AND FIELD CABLES (EXCEPT AT THE VALVES DUE TO RADIOLOGICAL CONCERNS) HAVE BEEN REMOVED

27 AIR OPERATOR AND AIR SUPPLY FOR TESTABLE CHECK VALVES RHR-V-41A 41B 41C 50A 50B & 89 ARE DEACTIVATED AND SPARED IN PLACE THESE VALVES ARE TESTED PER THE IST PROGRAM PENETRATION AND ISOLATION/TEST VALVES ARE STILL ACTIVE

28 ALL ITEMS IN THE CROSSHATCHED AREA ARE PART OF THE HYDROGEN RECOMBINER SYSTEM (CONTAINMENT ATMOSPHERE CONTROL- CAC) WHICH HAS BEEN DISABLED PRIMARY CONTAINMENT VALVES RHR-V-134A AND RHR-V-134B HAVE BEEN DE-ENERGIZED AND LOCKED CLOSED VALVES RHR-V-176A AND RHR-V-176B HAVE BEEN LOCKED CLOSED (REF PDC 3539)

29 FOR PERIODIC THERMAL PERFORMANCE TESTING OF RHR-HX-1A FOUR THERMISTORS ARE INSTALLED ON THE RHR WATER INLET SIDE OF RHR-HX-1A EIGHT THERMISTORS ARE INSTALLED ON THE RHR WATER OUTLET SIDE OF RHR-HX-1A AND AN ULTRASONIC FLOW ELEMENT IS ON THE RHR WATER INLET PIPE OF RHR-HX-1A THESE ARE MOUNTED QCII+ SC IM

30 FOR PERIODIC THERMAL PERFORMANCE TESTING OF RHR-HX-1B FOUR THERMISTORS ARE INSTALLED ON THE RHR WATER INLET SIDE OF RHR-HX-1B EIGHT THERMISTORS ARE INSTALLED ON THE RHR WATER OUTLET SIDE OF RHR-HX-1B AND AN ULTRASONIC FLOW ELEMENT IS ON THE RHR WATER INLET PIPE OF RHR-HX-1B THESE ARE MOUNTED QCII+ SC IM

31 RHR-V-60A AND RHR-V-75A HAVE BEEN DEACTIVATED-DISABLED VALVE INTERNALS HAVE BEEN REMOVED TO ALLOW CONTINUOUS FLOW PER EC 11109

32 TYGON HOSE (REF DWG D-VENTDRAINHOSE-319) WITH QUICK DISCONNECT IS PROVIDED AS A MEANS TO VISIBLY VERIFY THAT ANY AIR HAS BEEN PROPERLY VENTED FROM THE SYSTEM AND MAY BE LEFT IN PLACE

LEGEND

1 ALL VALVES (EXCEPT THOSE ON INSTRUMENT LINES AND AS NOTED ON DIAGRAM) SUFFIXED WITH A (V) DENOTE A 3/4 VENT VALVE

2 ALL VALVES (EXCEPT THOSE ON INSTRUMENT LINES) SUFFIXED WITH A (D) DENOTE A 3/4 DRAIN VALVE

3 ALL VALVES SUFFIXED WITH A (TH) DENOTE A THROTTLED VALVE

REFERENCES TO CONTRACT NUMBERS SUCH AS 215 206 220 FOUND IN THE NOTES ON THIS DRAWING REFER TO ORIGINAL CONSTRUCTION DESIGN REQUIREMENTS

CURRENT DESIGN REQUIREMENTS ARE DOCUMENTED IN THE STATION DESIGN SPECIFICATIONS AND ACTIVE CONTRACT SPECIFICATIONS WHICH STILL APPLY

Columbia Generating Station
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Residual Heat Removal System – P&ID

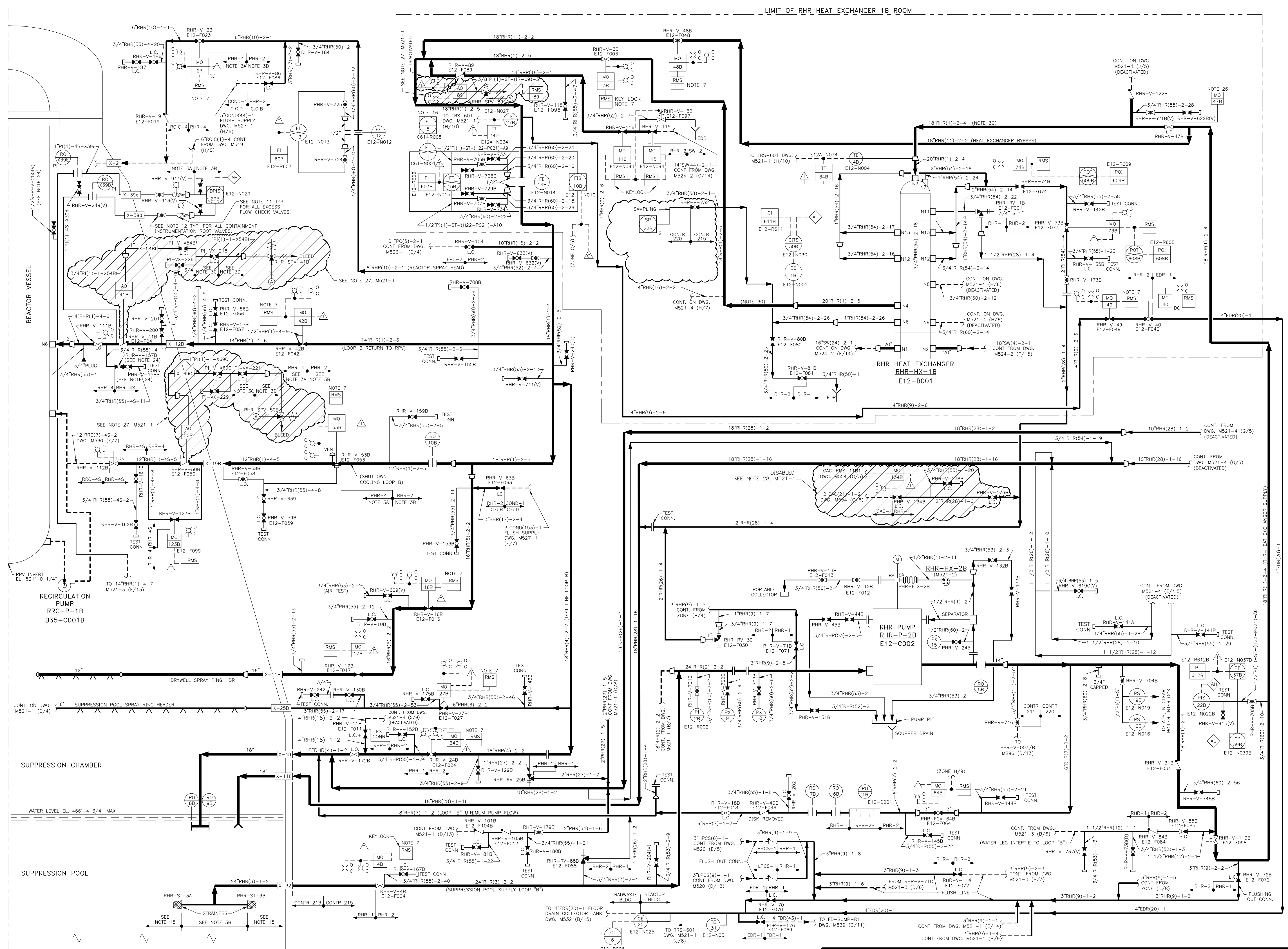
Draw. No. M521-1 | Rev. 112 | Figure 5.4-15.1

NOTES:

FOR NOTES SEE M521-1.

LEGEND:

1. ALL VALVES (EXCEPT THOSE ON INSTRUMENT LINES) SUFFIXED WITH A (V) DENOTE A 3/4" VENT VALVE EXCEPT AS NOTED.
2. ALL VALVES (EXCEPT THOSE ON INSTRUMENT LINES) SUFFIXED WITH A (D) DENOTE A 3/4" DRAIN VALVE.
3. ALL VALVES SUFFIXED WITH A (TH) DENOTE A THROTTLED VALVE.



REFERENCES TO CONTRACT NUMBERS, SUCH AS 215, 206, 220, FOUND IN THE NOTES ON THIS DRAWING REFER TO ORIGINAL CONSTRUCTION DESIGN REQUIREMENTS.

CURRENT DESIGN REQUIREMENTS ARE DOCUMENTED IN THE STATION DESIGN SPECIFICATIONS AND ACTIVE CONTRACT SPECIFICATIONS WHICH STILL APPLY.

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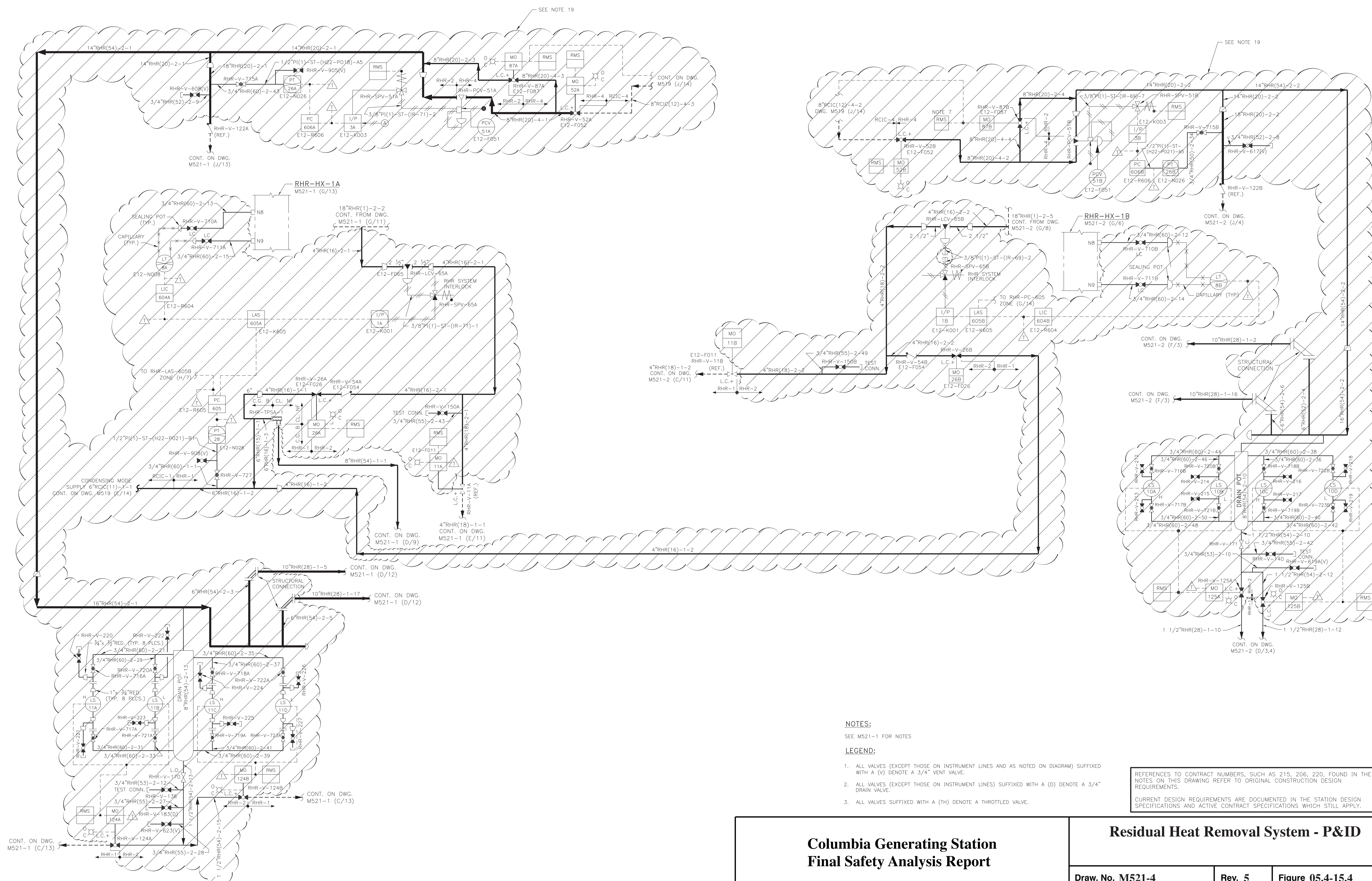
Residual Heat Removal System – P&ID

Draw. No. M521-2	Rev. 112	Figure 5.4-15.2

Figure Not Available For Public Viewing



NOTES:

SEE M521-1 FOR NOTES

LEGEND:

1. ALL VALVES (EXCEPT THOSE ON INSTRUMENT LINES AND AS NOTED ON DIAGRAM) SUFFIXED WITH A (V) DENOTE A 3/4" VENT VALVE.
2. ALL VALVES (EXCEPT THOSE ON INSTRUMENT LINES) SUFFIXED WITH A (D) DENOTE A 3/4" DRAIN VALVE.
3. ALL VALVES SUFFIXED WITH A (TH) DENOTE A THROTTLED VALVE.

REFERENCES TO CONTRACT NUMBERS, SUCH AS 215, 206, 220, FOUND IN THE NOTES ON THIS DRAWING REFER TO ORIGINAL CONSTRUCTION DESIGN REQUIREMENTS.

CURRENT DESIGN REQUIREMENTS ARE DOCUMENTED IN THE STATION DESIGN SPECIFICATIONS AND ACTIVE CONTRACT SPECIFICATIONS WHICH STILL APPLY.

Columbia Generating Station
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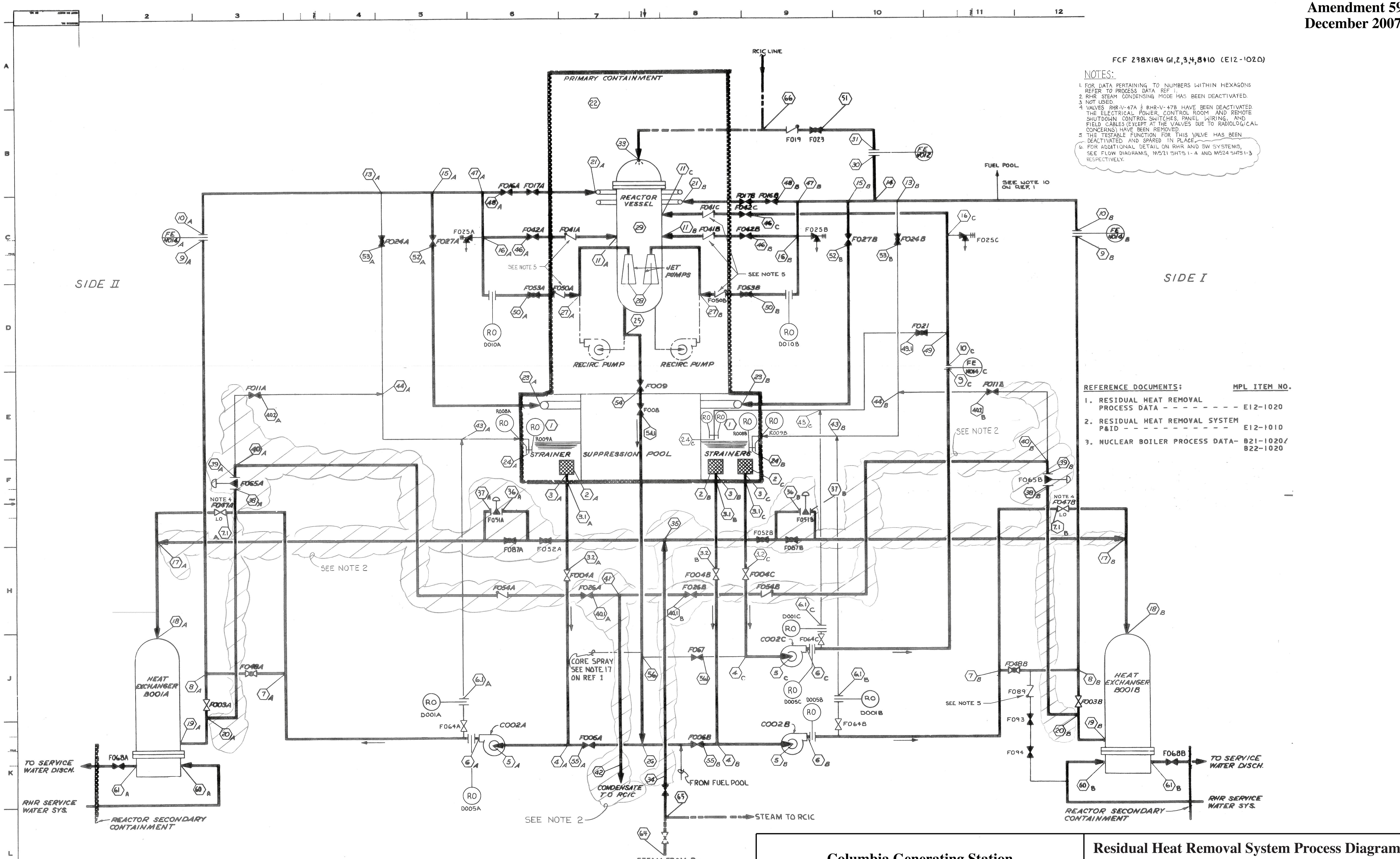
Residual Heat Removal System - P&ID

Draw. No. M521-4	Rev. 5	Figure 05.4-15.4



Columbia Generating Station
Final Safety Analysis Report

Residual Heat Removal System Process Diagram

Draw. No. 02E12-04,1,1	Rev. 9	Figure 5.4-16

MODE F (SEE NOTE 3)

POSITION ○	1	2	3	4	5	6	49	7	8	9	10	13	23/24	1
FLOW-GPM	—	7450	←									→	7450	—
PRESS-PSIA	14.7		X	X	X	X	X	X	X	X	X			14.7
TEMP °F	—	120/40	←									→	120/40	—
MAX. PRESS. DROP-FEET	←				TDH=267								→	

MODE G (Rx PRESS. 135 PSIG)

POSITION ○	29	25	26	4	5	6	43	24	1		1	2 BC	3 BC	4 BC	5 BC	6 BC	6.2	43 B	24 B	1
FLOW-GPM	—	550	←				→	550	—		—	550	←					→	550	—
PRESS-PSIA	150	X	X	X				X	14.7		14.7	X	X	X	X				X	14.7
TEMP °F	—	358	←				→	358	—		—	125/40	←					→	125/40	—
MAX. PRESS. DROP-FEET	←			TDH=650 SEE NOTE 22				→			←				TDH=650 SEE NOTE 22				→	

MODE S

POSITION ○	1	2 ABC	3 ABC	4 ABC	5 ABC	6 ABC	18 AB	19 AB	20 AB	9 ABC	10 ABC	46 ABC	11 ABC	48 AB	21 AB	50 AB	27 AB	51	33	52 AB	53 AB
FLOW-GPM	N/A	←																		→	N/A
PRESS-PSIA	14.7	X	X	X	X	X	X	X	X	X	X	X	1015	X	155	X	1015	X	1015	X	X
TEMP °F	90/40	←										90/40	547	90/40	135	90/40	547	90/40	547	90/40	90/40
MAX. PRESS DROP-FEET																					

MODE S (CONT'D)

POSITION ○	54	56	55 A,B		34	35	36 A,B	37 A,B	17 A,B		20 A,B	38 A,B	39 A,B	41	42
FLOW-GPM	N/A	N/A	N/A		N/A	←			N/A		N/A	←			N/A
PRESS-PSIA	X	X	X		X	X	X	X	X		X	X	X	X	X
TEMP °F	AMB	AMB	AMB		AMB	AMB	AMB	90/40	90/40		90/40	90/40	AMB	AMB	AMB
MAX PRESS DROP-FEET															

MODE C-3 (SEE NOTE 24) 4 HRS 135PSIG Rx PRESS

POSITION ○	29	64	65	34	35	36 A,B	37 A,B	17 A,B	18 A,B	19 A,B	20 A,B	38 A,B	39 A,B	40 A,B	41	42				60 A,B	61 A,B		
FLOW (SEE NOTE 15)	—	119.6	114.4	←	114.4	57.2	←	→	57.2	116	←		→	116	232	232				7400	7400		
PRESS-PSIA	150							X	X	>60	X	X	X	X	X	59.7/14.7							
TEMP °F	—	323	←						→	323	140	←				→ 140				95	111.8		
MAX PRESS DROP-PSI	←						ΔP=90.3/135.3									→							

HEAT REMOVAL PER STEAM CONDENSING HX 62.15 X 10^6 BTU/HR (2 HX STEAM CONDENSING)

MODE C-4 (SEE NOTE 24) 8 HRS 135PSIG Rx PRESS

POSITION ○	29	64	65	34	35	36 A	37 A	17 A	18 A	19 A	20 A	38 A	39 A	40 A	41	42				60 A	61 A		
FLOW (SEE NOTE 15)		93.7	93.5	←				→	93.5	190	←				→	190				7400	7400		
PRESS PSIA	150	X	X	X	X				>60	X					X	59.7/14.7				X	X		
TEMP °F	—	323	←					→	323	140	←				→	140				95	122.5		
MAX PRESS DROP-PSI	←						ΔP=90.3/135.3									→							

HEAT REMOVAL PER STEAM CONDENSING HX 101.6 X 10^6 BTU/HR (1 HX STEAM CONDENSING)

MODE A-1 (SEE NOTE 3 & 14)

POSITION ○	1	2	3	4	5	6	7	8	9	10	16	11	29
FLOW-GPM	—	7450	←								→	7450	—
PRESS-PSIA	59.7/14.7	X	X	X	X								85.7/40.7
TEMP °F	—	170/40	←								→	170/40	—
MAX. PRESS. DROP-FEET	←				TDH=267				ΔH=17			ΔH=40	→

MODE A-2 (SEE NOTE 14)

SEE NOTE 27

POSITION ○	1	2	3	4	5	6	7	8	9	10	16	11	29
FLOW-GPM	—	8000	←								→	8000	—
PRESS-PSIA	14.7	X	X	X	X	X	X	X	X	X	X	X	14.7
TEMP °F	—	180/90	←								→	180/90	—
MAX. PRESS DROP-FEET	←				TDH=235								→

MODE B

POSITION ○	1	2 A,B	3 A,B	4 A,B	5 A,B	6 A,B	7 A,B	17 A,B	18 A,B	19 A,B	20 A,B	8 A,B	9 A,B	10 A,B	13 A,B	14	15 A,B	21 A,B	22	23 A,B	1		60 A,B	61 A,B
FLOW-GPM	—	7900	←													→	7900	7450	—	450	—		7400	7400
PRESS-PSIA	59.7/14.7					X	X	X	X	X	X	X	X	X	X	X			59.7/14.7		59.7/14.7		X	X
TEMP °F	—	212	←					→	212	179.3	179.3	179.3	179.3	179.3	179.3	179.3	179.3	179.3	—	179.3	—		95	128.8
MAX. PRESS. DROP-FEET	←				TDH=235														→		→			

HEAT REMOVAL PER HX 121.7X10^6 BTU/HR (1 HX OPERATING) SEE NOTE 16

INDICATES FLOW PASSING THRU ABOVE POSITIONS (TYP)

SEE NOTE 7 (TYP)

MODE C-1 (SEE NOTE 24) Rx PRESS 1000 PSIG

POSITION ○	29	34	35	36 A,B	37 A,B	17 A,B	18 A,B	19 A,B	20 A,B	38 A,B	39 A,B	40 A,B	41	42	40.2 A,B	24 A,B	1		60 A,B	61 A,B
FLOW (SEE NOTE 15)	—	187.1	187.1	93.5	←	→	93.5	190	←		→	190	380	380	190	190	—		7400	7400
PRESS-PSIA	1015	X	X	X	214.7	X	X	X	X	X	X	X		59.7			14.7		X	X
TEMP °F	—	546	546	546	387.7	387.7	387.7	140	←						→	140	—		95	122.4
MAX. PRESS. DROP-PSI	←	ΔP=800		→	←					ΔP=200 / ΔP=155					→		→			

HEAT REMOVAL PER HX 101.4 X10^6 BTU/HR (2 HX STEAM CONDENSING)

MODE C-2 (SEE NOTE 24) Rx PRESS 1000 PSIG

POSITION ○	29	34	35	36 A	37 A	17 A	18 A	19 A	20 A	38 A	39 A	40 A	41	42		60 A	61 A
FLOW (SEE NOTE 15)	—	136.2	←			→	136.2	277	←				→	277		7400	7400
PRESS-PSIA	1015	X	X	X	214.7	X	X	X					X	59.7		X	X
TEMP °F	—	546	546	546	387.7	387.7	387.7	140	←				→	140		95	134.8
MAX PRESS DROP-PSI	←		ΔP=800		→ ←					ΔP=155				→			

DUTY PER STEAM CONDENSING HX 147.5 x10^6 BTU/HR (1 HX STEAM CONDENSING)

MODE C-2 (CONT'D) (SEE NOTE 24) (SEE NOTE 14)

POSITION ○	1	2 A,B	3 A,B	4 A,B	5 A,B	6 A,B	7 A,B	17 A,B	18 A,B	19 A,B	20 A,B	9 A,B	10 A,B	13 A,B	43 A,B	24 A,B	1		60 A,B	61 A,B
FLOW/GPM	—	7450	←												→	7450	—		7400	7400
PRESS.-PSIA	14.7	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	14.7		X	X
TEMP °F	—	120/70	←					→	120/70	103.1	103.1	103.1	103.1	103.1	103.1	103.1	—		95	103.2
MAX. PRESS. DROP-FEET	←				TDH=267												→			

HEAT REMOVAL PER POOL COOLING HX 26.0 X 10^6 BTU/HR (1 HX OPERATING)

MODE D (SEE NOTE 26) RX PRESS 48 PSIG

POSITION ○	29	25	26	4 A,B	5 A,B	6 A,B	18 A,B	19 A,B	8 A,B	9 A,B	10 A,B	14	27 A,B	28	30	31	51	33	C9				60 A,B	61 A,B
FLOW/GPM	—	7450	7450	7450	7450	7450	7450	7450	7450	←	→	7450	7450/6580	7450/6580	870/0	870/0	870/0	870/0	—				7400	7400
PRESS-PSIA	62.7			X	X	X	X	X	X	X	X	X	X	X	62.7				110				X	X
TEMP °F	—	295	←		→	295	295	295	295	295	295	295	295	295	295	295	295	295	—				85	125
MAX PRESS DROP-FEET	←				TDH=267										ΔH=	17		ΔH=	→					

HEAT REMOVAL PER HX 148 X10^6 BTU/HR (1 HX OPERATING) SEE NOTE 21

SEE RCIC (REF 3)

MODE E RX PRESS 0 PSIG

POSITION ○	29	25	26	55 A,B	4 A,B	5 A,B	6 A,B	18 A,B	19 A,B	9 A,B	10 A,B	11 A,B	16 A,B	27 A,B	28 A,B	29		60 A,B	61 A,B
FLOW-GPM	—	14,900/7450	14,900	7450	←									→	7450	—		7400	7400
PRESS-PSIA	14.7	X	X	X	X	X	X	X	X	X	X	X				14.7		X	X
TEMP °F	—	125/AMB	←				→	125/AMB	113.8	113.8	113.8	113.8	113.8	113.8	113.8	—		85	96.3
MAX PRESS DROP-FEET	←					TDH=267		ΔH=23.3							ΔH=18	→		ΔH=23.1	

LOOPS A & B ONLY

DUTY PER HX 41.6X10^6 BTU/HR (2 HX'S OPERATING)

MODES:

A-1 ACCIDENT WITH RECIRCULATION LINE BREAK IN EITHER SIDE AND THREE PUMP OPERATION. SEE NOTE 25

A-2 ACCIDENT WITH RECIRCULATION LINE BREAK IN EITHER SIDE AND THREE PUMP OPERATION. SEE NOTE 25 (10 MIN)

B POST ACCIDENT CONTAINMENT SPRAY WITH HEAT REJECTION WITH ONE PUMP OPERATION. SEE NOTE 25 (PEAK SUPPRESSION POOL TEMPERATURE)

C-1 REACTOR ISOLATION - 2HX STEAM CONDENSING (1/2 HR) SEE NOTE 24

C-2 REACTOR ISOLATION - 1HX STEAM CONDENSING AND 1HX POOL COOLING (1 1/2 HR) SEE NOTE 24

C-3 REACTOR ISOLATION - 2HX STEAM CONDENSING (4 HR) SEE NOTE 24

C-4 REACTOR ISOLATION - 1HX STEAM CONDENSING (8 HR) SEE NOTE 24

D NORMAL SHUTDOWN AFTER BLOWDOWN TO MAIN CONDENSER

E CONTINUATION OF NORMAL SHUTDOWN FROM PLANT MODE "D" (AT 0 PSIG) AND FUNCTIONAL PUMP TEST AFTER SHUTDOWN.

F RHR SYSTEM TEST DURING PLANT OPERATION.

G MINIMUM FLOW BYPASS MODE.

S SYSTEM ON STANDBY DUTY

LEGEND

Rx PRESS----REACTOR VESSEL PRESSURE
TDH---------TOTAL DYNAMIC HEAD
SOH---------SHUTOFF HEAD
ΔH----------HEAD LOSS
ΔP----------PRESSURE LOSS

REFERENCE DOCUMENTS:

		MPL ITEM NO.
1.	RESIDUAL HEAT REMOVAL SYSTEM P&ID	E12-1010
2.	RESIDUAL HEAT REMOVAL SYSTEM DESIGN SPEC	E12-4010
3.	RCIC PROCESS DIAGRAM	E51-1020
4.	RECIRC SYSTEM PROCESS FLOW DIAGRAM	B35-1040
5.	LOW PRESSURE CORE SPRAY PFD	E21-1020

NOTES

1. PROCESS DIAGRAM 731E966 SHALL BE USED WITH AND FORM A PART OF THIS PROCESS DATA. IF THERE ARE ANY CONFLICTS BETWEEN THE PROCESS DIAGRAM AND THIS PROCESS DATA, THE PROCESS DATA SHALL GOVERN.
2. PIPING BETWEEN POINTS WITH EMPTY DATA BLANKS (SEE ALSO TABLE 3) SHALL BE SIZED BY OTHERS BASED ON SPECIFIED OPERATING CONDITIONS. EMPTY DATA BLANKS CAN BE FILLED IN BASED ON ACTUAL ARRANGEMENT BY OTHERS. X - INDICATES THE DATA IS NOT SIGNIFICANT.
3. SHOWN AS TYPICAL FOR ONE LOOP. IF LOOPS ON SIDE I AND SIDE II ARE NOT SYMMETRICALLY ARRANGED, VALUES FOR BOTH SIDES SHALL BE SUBMITTED.
4. ΔP VALUES FOR EQUIPEMENT WITHIN GE-BWRSD SCOPE ARE AS NOTED.
5. ELEVATIONS ARE NOT INCLUDED IN ΔP, VALUES GIVEN, ELEVATIONS SHALL BE INCLUDED WHEN DETERMINING FINAL VALUES FOR THE EMPTY DATA BLANKS.
6. X/Y - INDICATES MAXIMUM (X) AND MINIMUM (Y) VALUES FOR THE MODE SPECIFIED.
7. DASHED LINES INDICATE FLOW DOES NOT PASS THRU THESE POINTS.
8. THE NPSH AVAILABLE IN MODES B & D AT THE CENTERLINE OF THE SUCTION NOZZLE ARE DEFINED IN CALCULATION 5.17.19
9. SERVICE WATER CROSSTIE, SHALL BE SIZED TO FLOW 300 GPM.
10. THIS PORTION OF PIPING TO BE SIZED BASED ON FLOW SHOWN ON THE FUEL POOL PROCESS DIAGRAM.
11. TABLE 1 INDICATES TYPICAL VALVE POSITIONS DURING VARIOUS MODES OPERATED.
12.
13.
14. TYPICAL VALUES FOR MAX. AND MIN. SUPPRESSION POOL TEMP SHOWN. FINAL TEMPERATURES DEPEND ON INITIAL POOL WATER TEMPERATURE & POOL WATER VOLUME.
15. WATER FLOWS ARE IN GPM, STEAM FLOWS ARE IN 1000 LBS/HR.
16. Hx DUTY BASED UPON 7450 GPM SHELL SIDE FLOW.
17. FOR LINE SIZING INFORMATION SEE REF. 5.
18. THE WEIGHT OF WATER IN THE SHUTDOWN COOLING SUBSYSTEM PIPING, INCLUDING THE HEAT EXCHANGERS AND PUMPS SHALL NOT EXCEED 225,000LBS AT 70°F TO PREVENT DILUTION OF STANDBY LIQUID CONTROL NEUTRON ABSORBER BELOW MINIMUM REQUIREMENTS.
19.
20. PIPING SYSTEM DESIGN PRESSURE AND TEMPERATURE AND THE ESTIMATED LINE SIZES ARE FOR INFORMATION ONLY. ACTUAL DESIGN PRESSURE & TEMPERATURE AND LINE SIZES AS DETERMINED BY OTHERS SHALL MEET THE PROCESS DATA HYDRAULIC REQUIREMENTS.
21. HEAT EXCHANGER CAPABILITY SHOWN AT SERVICE WATER OUTLET TEMP. OF 125°F.
22. PUMP SHUTOFF HEAD 800 FT. MAX.
23. THE HX STEAM INLET PRESSURE SHALL BE GREATER THAN 60 PSIA TO MINIMIZE THE POSSIBILITY OF FLOW INDUCED VIBRATION.
24. THE STEAM CONDENSING MODE HAS BEEN DEACTIVATED.
25. STRAINER PLUGGING CRITERIA ARE DEPICTED IN CALCULATIONS 5.17.19 AND ME-02-97-03
26. DURING REFUELING OUTAGES, THE HEAD SPRAY LINE (NODES 30, 31 & 51) MAY ALSO BE USED TO PROVIDE 1000 GPM FLOW TO THE REACTOR CAVITY FOR DECAY HEAT REMOVAL.
27. MAX FLOW (RUNOUT) FOR RHR-P-2B IS 8100 GPM.

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Draw. No. 02E12-04,22,1 | Rev. 8 | Figure 5.4-17.1

Form No. 960690ai

SEE NOTE 20

POSITION ○	3.1	3.2	4	5	6	7	C	9	10	47	16	46	11	7	7.1	18	19	20	8	3.2C	4C	5C	6	9	10	46	11	23	54.1	26	4AA	5AA	16	50	27
DESIGN PRESS. IN PSIG	125	220		RHR PUMP	500							1250		500		RHR HX	500		220			RHR PUMP	500			1250		SEE REF 4 PSI 1-2	220				500	SEE REF 4 PSI 1-16	
DESIGN TEMP IN °F	212	358			480						480	575		358			358		358				480			575			358				480		
ESTIMATED LINE SIZE	24"				18"						12"			18"			18"		24"				18"			12"		20"		18"	24"		12"		
	LPCI LINE (VIA RHR Hx BYPASS)													Hx LINE						LPCI LINE (DIRECT TO REACTOR)								SHUTDOWN SUCTION					SHUTDOWN RETURN LINE		

POSITION ○	34	35	36	37	17	18	15	20	38	39	40	40.1	41	42	14	30	51	51	66	40	40.2	44	47	48	21	15	52	23	13	53	44	45	24
DESIGN PRESS. IN PSIG	1250				500		500				500		125		500			SEE REF. 3		500	125		500			500			500	125			
DESIGN TEMP IN °F	575				480		480	358					212		358					358	500		358		212	358	212		358	300			
ESTIMATED LINE SIZE	10"		8		18		18"	4"							6"					4"			16"			4"			18"				
	STEAM TO HX						CONDENSATE FROM HX								HEAD SPRAY LINE					CONDENSATE TO POOL			DRYWELL SPRAY			SUPPRESSION POOL SPRAY			SYSTEM TEST LINES (LOOPS A & B)				

POSITION ○	56	56.1	4	49	49.1	24	6	6.1		45	6	6.1	45
DESIGN PRESS. IN PSIG	220	125	500		125		500	125			500	125	
DESIGN TEMP IN °F	358	212	358				358	300			358	212	
ESTIMATED LINE SIZE	18"		18"				4"		4"		4"		
	SYSTEM TEST LINES (LOOP C)						MIN FLOW BYPASS LINES (LOOP C)				MIN FLOW BYPASS (LOOP...)		

TABLE I - VALVE POSITION CHART

SEE NOTE 11

LOOPS A & B / LOOP C																																	STRAINER AB	STRAINER C
MODE A-1	O	O						O		O	O																						1-50% P	
MODE A-2	O	O						O		O	O																						1-50% P	
MODE B		O					O		O							O	O	O													O		[illegible]	---
MODE C-1				O	T																			T	O	O	O				O		—	—
MODE C-1				O	T																			T		O	O				O		—	—
MODE C-2				O	T																			T		O	O				O		—	—
C-2 CONT'D		O					O		O										T-O												O		CLEAR	—
MODE C-3				O	O	O																		T		O	O				O		—	—
MODE C-4				O	O	O																		T		O	O				O		—	—
MODE D			O				O	O-T	T				O	O						T		O							O	O	O			
MODE E			O				O		T				O	O														O	O	O	O			
MODE F	O	O					O	O	O									T-C	O				T-C										CLEAR	CLEAR
MODE G	O	O	O																									O	O	O		O	CLEAR	CLEAR
MODE S	O	O					O	O	O																								CLEAR	CLEAR

O - VALVE OPEN
T - VALVE THROTTLED
P - STRAINER PLUGGED
T-C - VALVE THROTTLED OR CLOSED
O-T - VALVE OPEN OR THROTTLED
BLANK SPACE INDICATES VALVE IS CLOSED

TABLE 3 LIMITING LINE LOSSES

MODE A-1	6-11 PUMP DISCHARGE LINE TO RPV FLOODING PENETRATION
MODE A-2	NONE
MODE B	1-2-5 (SUCTION LINE SUPPRESSION POOL TO PUMP), 47-21 & 15-23 (CONTAINMENT SPRAY LINES PUMP DISCHARGE TO TO SPRAY HEADERS)
MODE C-1	40-44 (Hx CONDENSATE TO SUPPRESSION POOL), 41-42 (Hx (CONDENSATE LINE TO RCIC LINE)
MODE C-2	20-40-41 (Hx CONDENSATE TO RCIC LINE POINT 41)
C-2 CONT'D	NONE
MODE C-3	29-34-35 (STEAM LINE FROM RCIC LINE TO Hx HEADER)
MODE C-4	35-17 (STEAM LINE HEADER TO Hx)
MODE D	14-51-53-29 (VESSEL HEADER SPRAY LINE) 29-25-5 (SHUTDOWN SUCTION LINE RPV TO PUMP) 56-4 (SHUTDOWN SUCTION LINE TEST BRANCH TO PUMP)
MODE E	16-27 (SHUTDOWN DISCHARGE LINE LPCI BRANCH TO RECIRC LOOP)
MODE F	13-13-24-11 49-45 (TEST LINE TO SUPPRESSION POOL)
MODE G	6-43 (PUMP MINIMUM BYPASS LINE)

RHR SYSTEM RELIEF VALVE SET PRESSURES

RELIEF VALVE EPN	GE MPL NUMBER	LINE DESIGN PRESSURE	RELIEF VALVE SET PRESSURE	POSITION DESIGNATION
RHR-RV-25A	E12-F025A	500 PSIG	487 PSIG	16A
RHR-RV-25B	E12-F025B	500 PSIG	488 PSIG	16B
RHR-RV-25C	E12-F025C	500 PSIG	493 PSIG	16C
RHR-RV-30	E12-F030	125 PSIG	103 PSIG	—
RHR-RV-88A	E12-F088A	220 PSIG	205 PSIG	4
RHR-RV-88B	E12-F088B	220 PSIG	209 PSIG	4
RHR-RV-88C	E12-F088C	220 PSIG	110 PSIG	4

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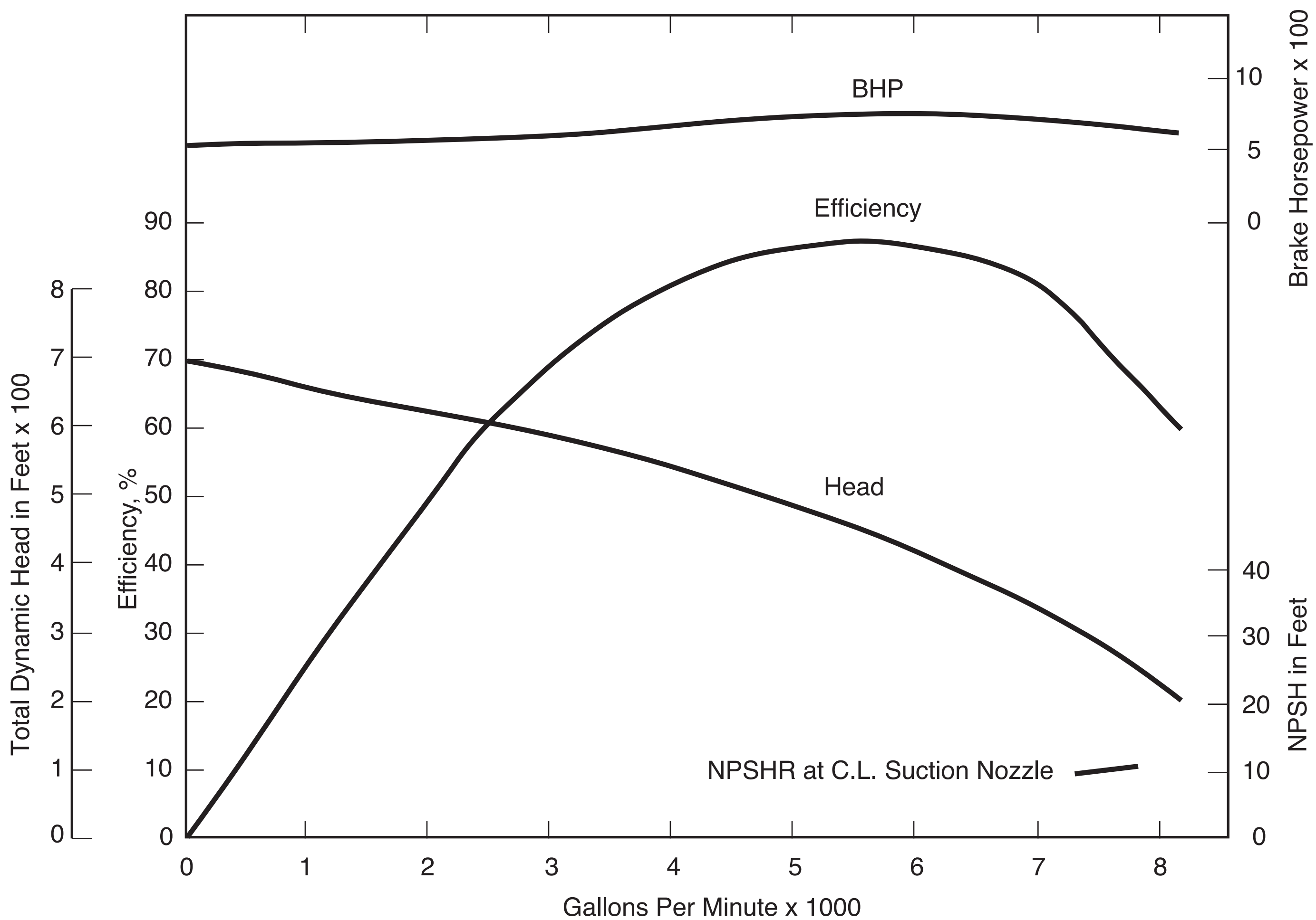
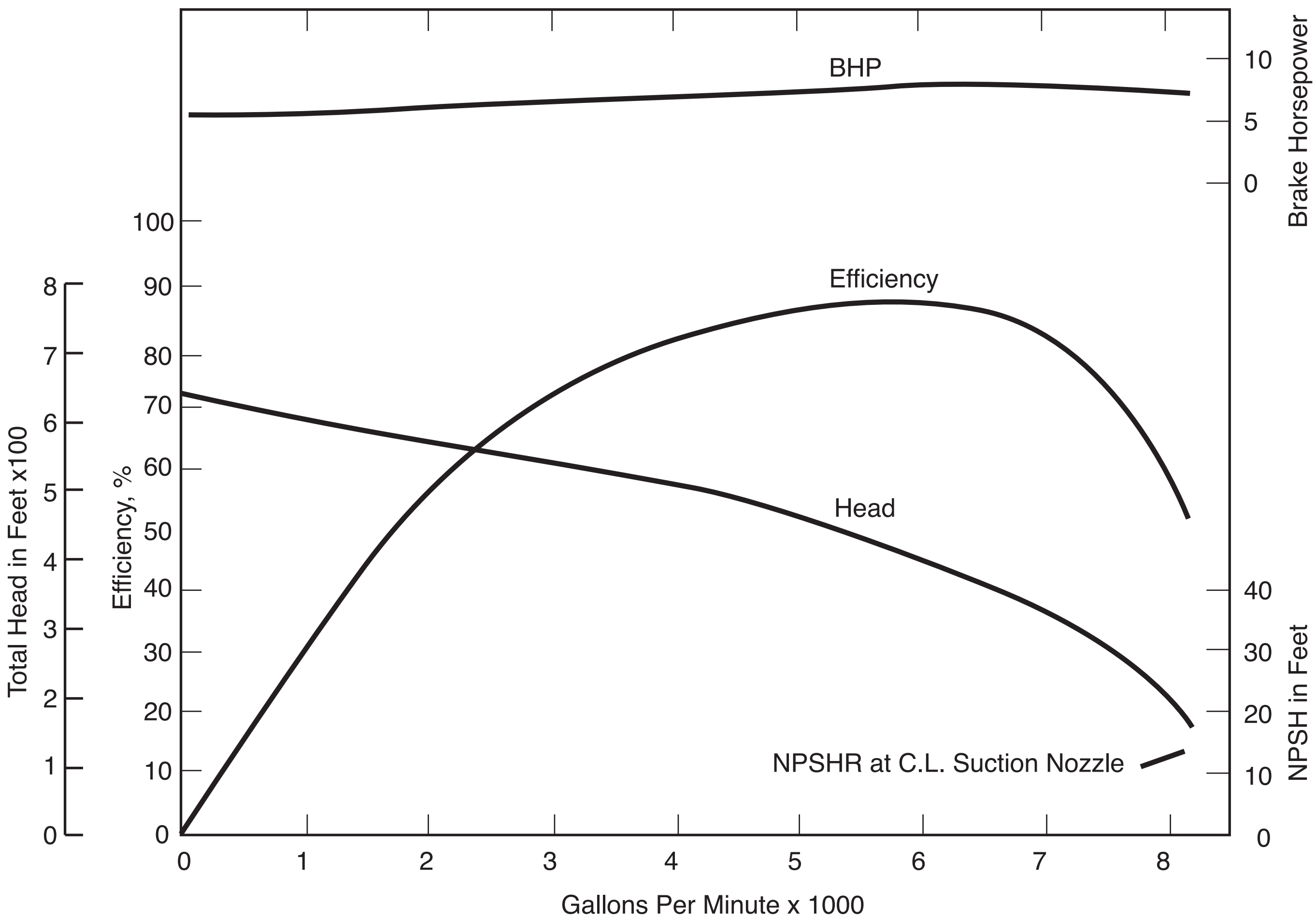
Draw. No. 02E12-04,22,2 | Rev. 6 | Figure 5.4-17.2

Form No. 960690

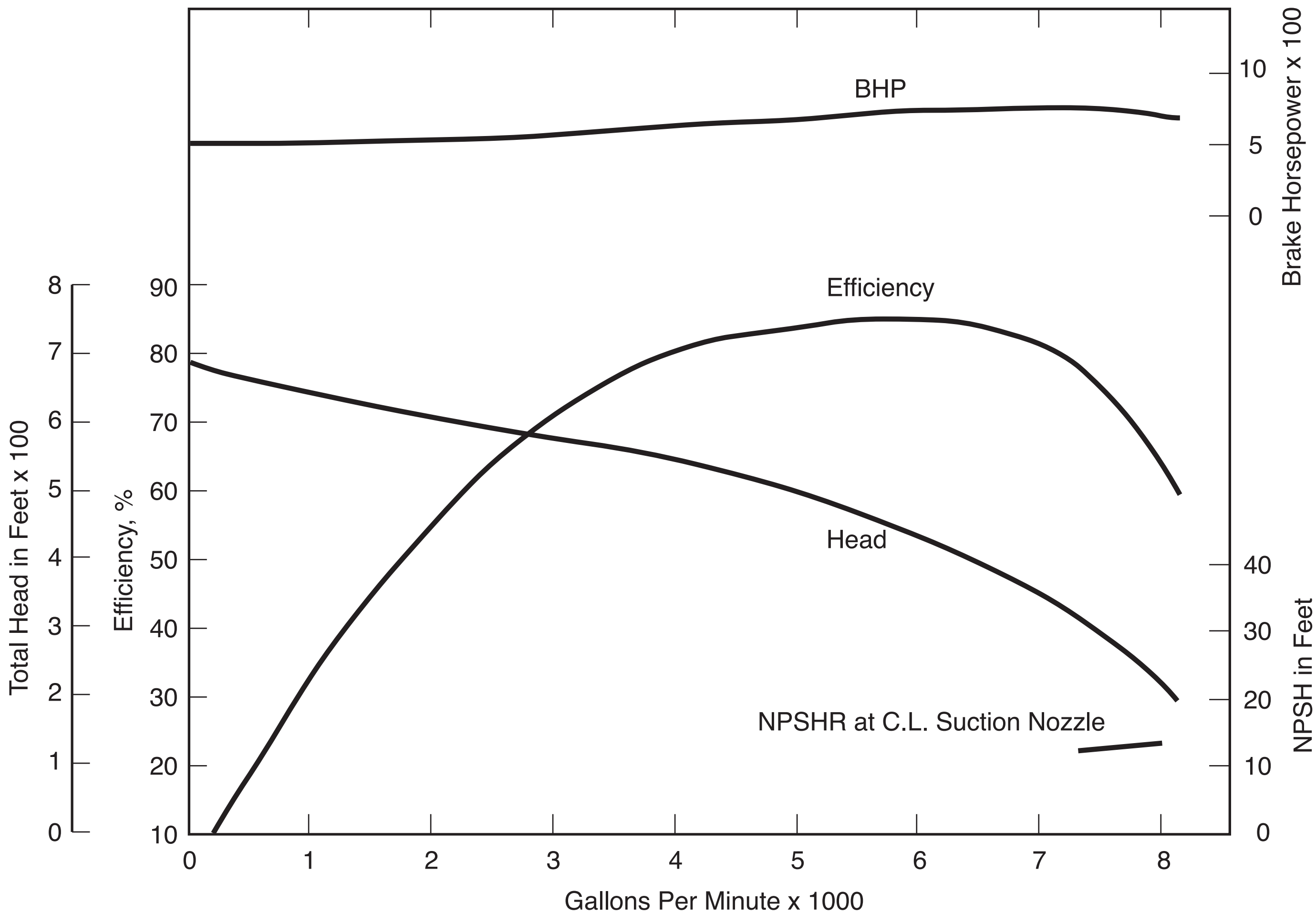


**RHR (LPCI) Pump Characteristics
(S/N 0473113) P-2A**

Draw. No. 960690.89 | Rev. | Figure 5.4-18



RHR (LPCI) Pump Characteristics (S/N 0801MP004399-1) P-2B



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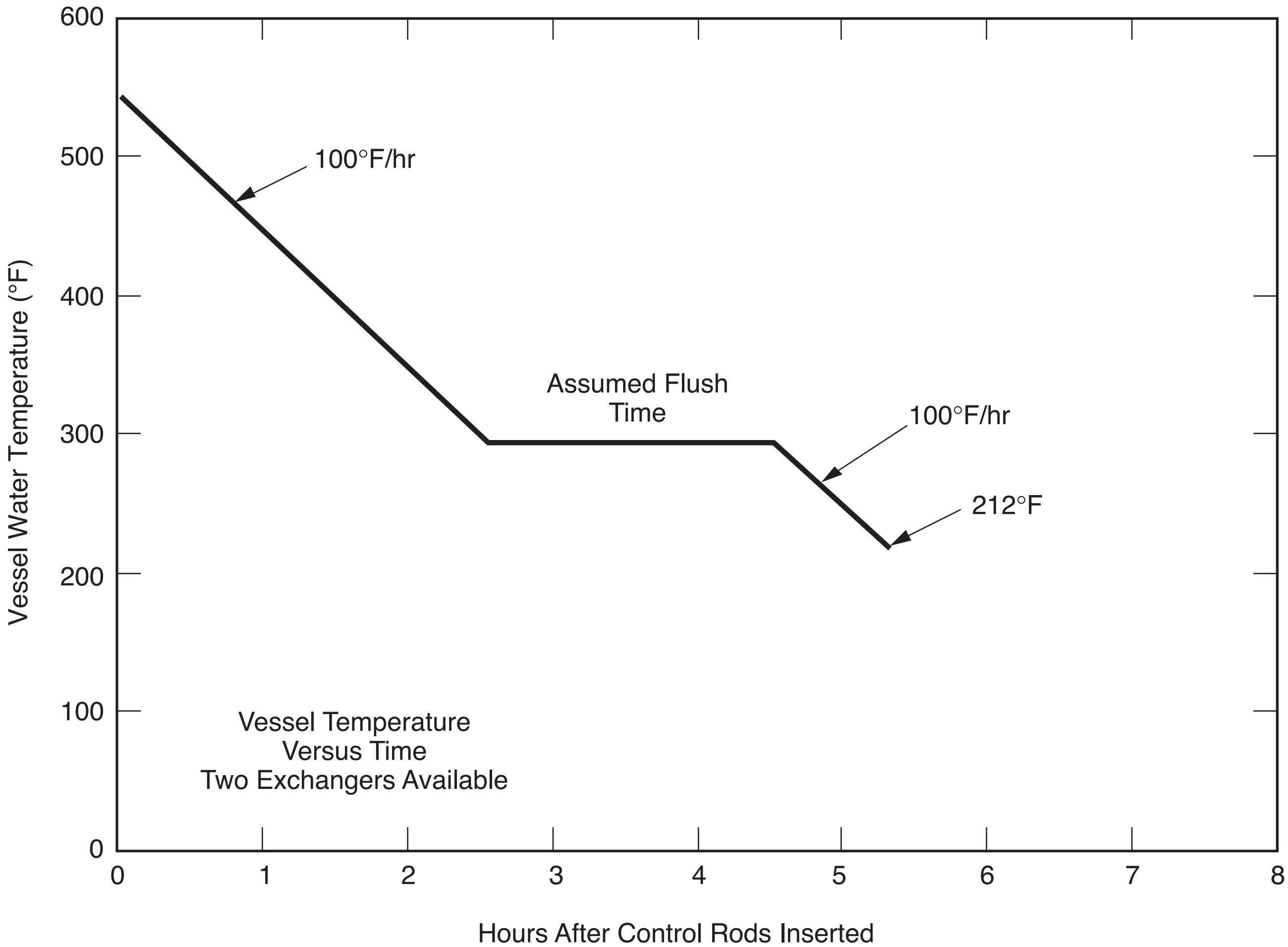
RHR (LPCI) Pump Characteristics
(S/N 0473112) P-2C

Draw. No. 960690.91 | Rev. | Figure 5.4-20



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Vessel Coolant Temperature Versus Time
(Two Heat Exchangers Available)

Draw. No. 960690.88 | Rev. | Figure 5.4-21

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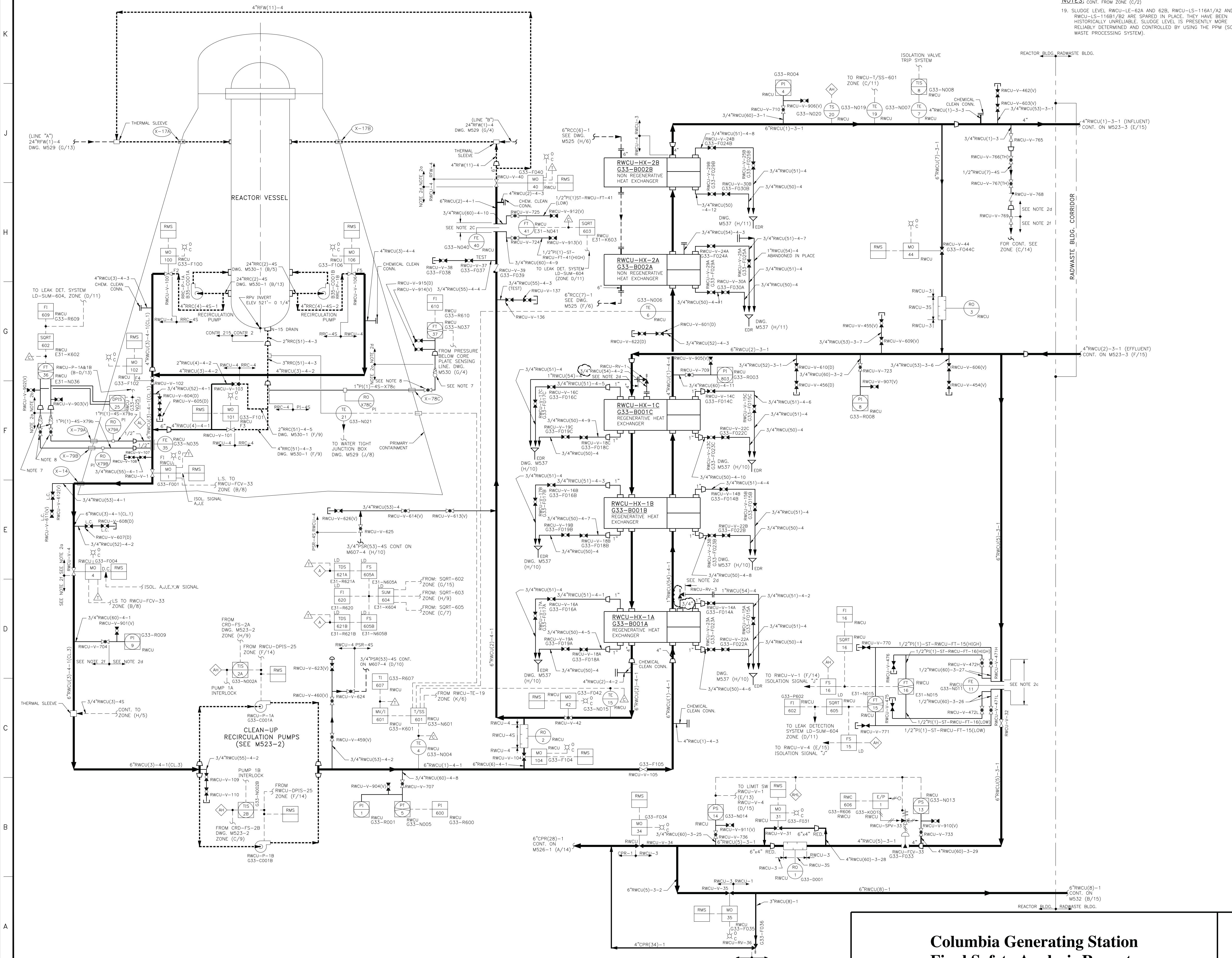

NOTES:

1. ALL ITEMS MARKED * ARE FURNISHED WITH ASSOCIATED EQUIPMENT.

2. PIPING, VALVES, AND ASSOCIATED COMPONENTS ON THIS DRAWING SHALL BE CLASSIFIED AS FOLLOWS, EXCEPT AS STATED IN NOTE 10: (BREAK POINTS INDICATED ON FLOW DIAGRAM).

   a. PIPING AND VALVES OUT THROUGH THE OUTERMOST ISOLATION VALVES, EXCEPT INSTRUMENT PIPING AS NOTED:
      SEISMIC CATEGORY: I
      QUALITY CLASS: I
      CODE GROUP: A

   b. INSTRUMENT PIPING, AND VALVES CONNECTED TO RCPB, ISOLATED BY A FLOW LIMITING VALVE, AND PROVIDING A SIGNAL TO THE LEAK DETECTION SYSTEM:
      SEISMIC CATEGORY: I
      QUALITY CLASS: I
      CODE GROUP: B

   c. INSTRUMENT PIPING, VALVES, AND FLOW ORIFICE CONNECTED TO THE RWCU SYSTEM THAT PROVIDE A SIGNAL TO THE LEAK DETECTION SYSTEM:
      SEISMIC CATEGORY: I
      QUALITY CLASS: I
      CODE GROUP: C

   d. PIPING, VALVES, AND ASSOCIATED COMPONENTS BEYOND OUTERMOST ISOLATION VALVES (INCLUDING IN THE CF, COND, CPR, MWR, AND SA SYSTEMS), EXCEPT AS STATED IN NOTE 2e & 2f.
      SEISMIC CATEGORY: II ⅄
      QUALITY CLASS: II
      CODE GROUP: D *

   e. PIPING, VALVES, AND ASSOCIATED COMPONENTS IN RADWASTE BUILDING: (AS INDICATED BY BREAKPOINTS)
      SEISMIC CATEGORY: II
      QUALITY CLASS: II +
      CODE GROUP: D +

   f. PIPING AND ASSOCIATED COMPONENTS BEYOND OUTERMOST ISOLATION VALVE (RWCU-V-4) TO, AND INCLUDING VALVES RWCU-V-5A & 5B: (AS INDICATED BY BREAKPOINTS)
      SEISMIC CATEGORY: II ⅄
      QUALITY CLASS: II
      CODE GROUP: C

   ⅄ HANGERS TO BE DESIGNED TO SEISMIC CATEGORY I IN SEISMIC AREA, SUCH AS THE REACTOR BUILDING AND THE RADWASTE BUILDING CORRIDOR.

   * PRESSURE VESSELS AND ASSOCIATED COMPONENTS WERE INSTALLED AS ASME III/CLASS 3. IF REPLACED, ASME SECTION VIII, DIVISION 1 NOW APPLIES.

   + SEE NOTE 12, WNP-2 SPECIFICATION, SECTION 15B.1, TABLE 2 NOTES.

   g. PIPING, VALVES, AND ASSOC. COMPONENTS REFERRING TO THIS NOTE:
      SEISMIC CATEGORY: I ø
      QUALITY CLASS: I ø
      CODE GROUP: C ø

   ⅄ HANGERS TO BE DESIGNED TO SEISMIC CATEGORY I LOADS.

   ø EXCEPT AS NOTED ON M619 INSTRUMENT CONNECTION DIAGRAMS. (+SEE NOTE 12, CONTRACT 215, SECTION 15B, TABLE 2 NOTES.)

3. ALL PIPING SYSTEMS INDENTIFIED BY THIS PREFIX "PI" SHALL BE SUPPLIED AND INSTALLED BY CONTRACT 220.

4. (DELETED)

5. ALL INSTRUMENT ROOT VALVES NOT LABELED WILL BE 3/4" GLOBE VALVES UNLESS SPECIFICALLY NOTED OTHERWISE.

6. (DELETED)

7. EXCESS FLOW CHECK VALVES SHALL BE TAGGED AS FOLLOWS: PI-EFC-X (PENETRATION NO.)

8. CONTAINMENT INSTRUMENTATION ROOT VALVES SHALL BE TAGGED AS FOLLOWS: PI-V-X (PENETRATION).

9. (DELETED)

10. ALL PIPING DOWN STREAM OF THE LAST ISOLATION VALVE, AND OPEN TO THE ATMOSPHERE WITH THE SUBSYSTEM DESIGNATION "SYSTEM" (50) THRU (59), SHALL BE CLASSIFIED AS CODE GROUP "D", WITH QUALITY CLASS AND SEISMIC CATEGORY PROVIDED BY THE APPLICABLE NOTES IN THIS DRAWING. EXCEPT WHERE THE APPLICABLE NOTES CALL FOR QUALITY CLASS II+, PIPING SHALL BE QUALITY CLASS II.

11. VALVES RWCU-V-210A&B ARE TO BE INSTALLED WITH THEIR FLOW ARROWS POINTED IN THE DIRECTION OF VALVES RWCU-V-234A&B.

12. LINE NUMBERS IN PARENTHESES DENOTE RWCU-DM-1A SYSTEM INDENTIFICATION.

13. NO UNIQUE LINE NUMBERS GIVEN FOR RWCU(ALL)-1 PIPING AS THESE ARE RADWASTE LINES.

14. ALL INSTRUMENT LINES ARE DESIGNATED BY CODE AND LINE NUMBER.

   a. EXAMPLE: LINES ORIGINATING OUTSIDE CONTAINMENT.
      $\underbrace{\text{1/2"PI(1)-ST}}_{\text{CODE DESIGNATION}} - \underbrace{\text{(H22-P021)-A6}}_{\text{LINE NUMBER}}$

   b. EXAMPLE: LINES ORIGINATING INSIDE CONTAINMENT.
      $\underbrace{\underbrace{\text{1/2"PI(1)-4S}}_{\text{CODE DESIGNATION}} \quad - \quad \text{X106}}_{\text{LINE NUMBER}}$

   c. EXAMPLE: CONTINUATION OF THE ABOVE LINE OUTSIDE CONTAINMENT IS DESIGNATED AS FOLLOWS, WITH LINE NUMBERS NOT SHOWN.
      $\underbrace{\text{1/2"PI(1)-ST-X106}}_{\text{CODE DESIGNATION}} - \underbrace{\text{(H22-P021)}}_{\text{LINE NUMBER}}$

15. CODE BREAK DEFINITIONS FOR THERMOWELLS ARE SHOWN ON M610, "INSTALLATION OF THERMOWELLS AND SAMPLE PROBES."

16. RWCU-PS-114 CONTACTS JUMPERED CLOSED TO INDICATE NORMAL PUMP SUCTION PRESSURE.

17. VALVES RWCU-V-443 AND RWCU-SPV-443 HAVE BEEN DEACTIVATED TO PREVENT INADVERTENT RESIN SPILLS.

18. FOR SAMPLE FUNCTION & INSTRUMENTATION ASSOCIATED WITH THE SAMPLE POINTS SEE DWG. SERIES M607 STEAM & LIQUID SAMPLING FLOW DIAGRAMS AND DWG. SERIES 60A-00 SAMPLE RACKS DETAILS & FLOW DIAGRAMS.

NOTES CONT. (K/4)

LEGEND:

1. ALL VALVES (EXCEPT THOSE ON INSTRUMENT LINES) SUFFIXED WITH A (V) DENOTE A 3/4" VENT VALVE EXCEPT AS NOTED.

2. ALL VALVES (EXCEPT THOSE ON INSTRUMENT LINES) SUFFIXED WITH A (D) DENOTE A 3/4" DRAIN VALVE.

3. ALL VALVES SUFFIXED WITH A (TH) DENOTE A THROTTLED VALVE.

REFERENCES TO CONTRACT NUMBERS, SUCH AS 215, 206, 220, FOUND IN THE NOTES ON THIS DRAWING REFER TO ORIGINAL CONSTRUCTION DESIGN REQUIREMENTS.

CURRENT DESIGN REQUIREMENTS ARE DOCUMENTED IN THE STATION DESIGN SPECIFICATIONS AND ACTIVE CONTRACT SPECIFICATIONS WHICH STILL APPLY.

NOTES: CONT. FROM ZONE (C/2)

19. SLUDGE LEVEL RWCU-LE-62A AND 62B, RWCU-LS-116A1/A2 AND RWCU-LS-116B1/B2 ARE SPARED IN PLACE. THEY HAVE BEEN HISTORICALLY UNRELIABLE. SLUDGE LEVEL IS PRESENTLY MORE RELIABLY DETERMINED AND CONTROLLED BY USING THE PPM (SOLID WASTE PROCESSING SYSTEM).

Columbia Generating Station
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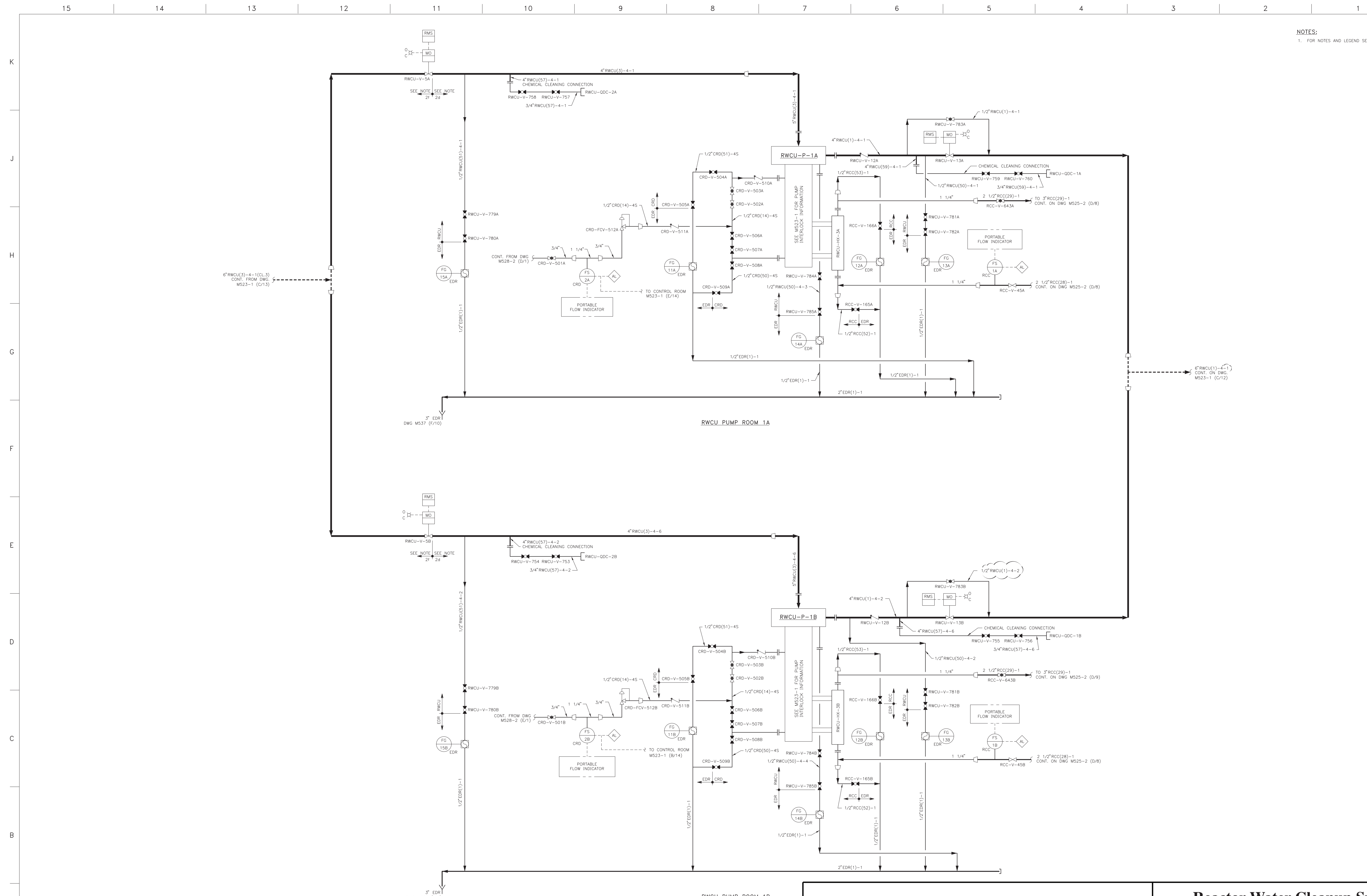
Reactor Water Cleanup System – P&ID

Draw. No. M523-1	Rev. 109	Figure 5.4-22.1



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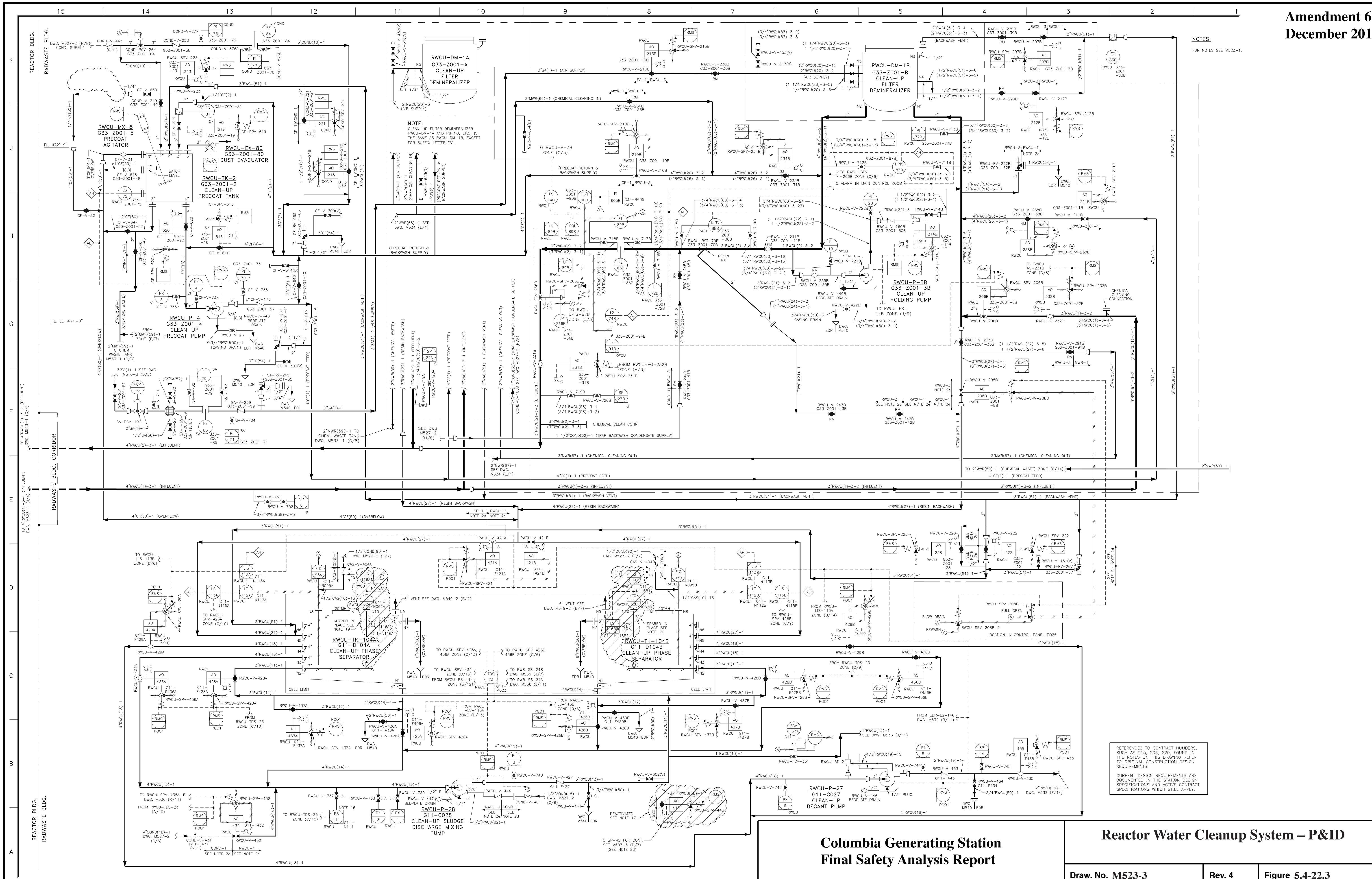
Reactor Water Cleanup System – P&ID

Draw. No. M523-2 | Rev. 7 | Figure 5.4-22.2

Form No. 960690ai

Amendment 61
December 2011

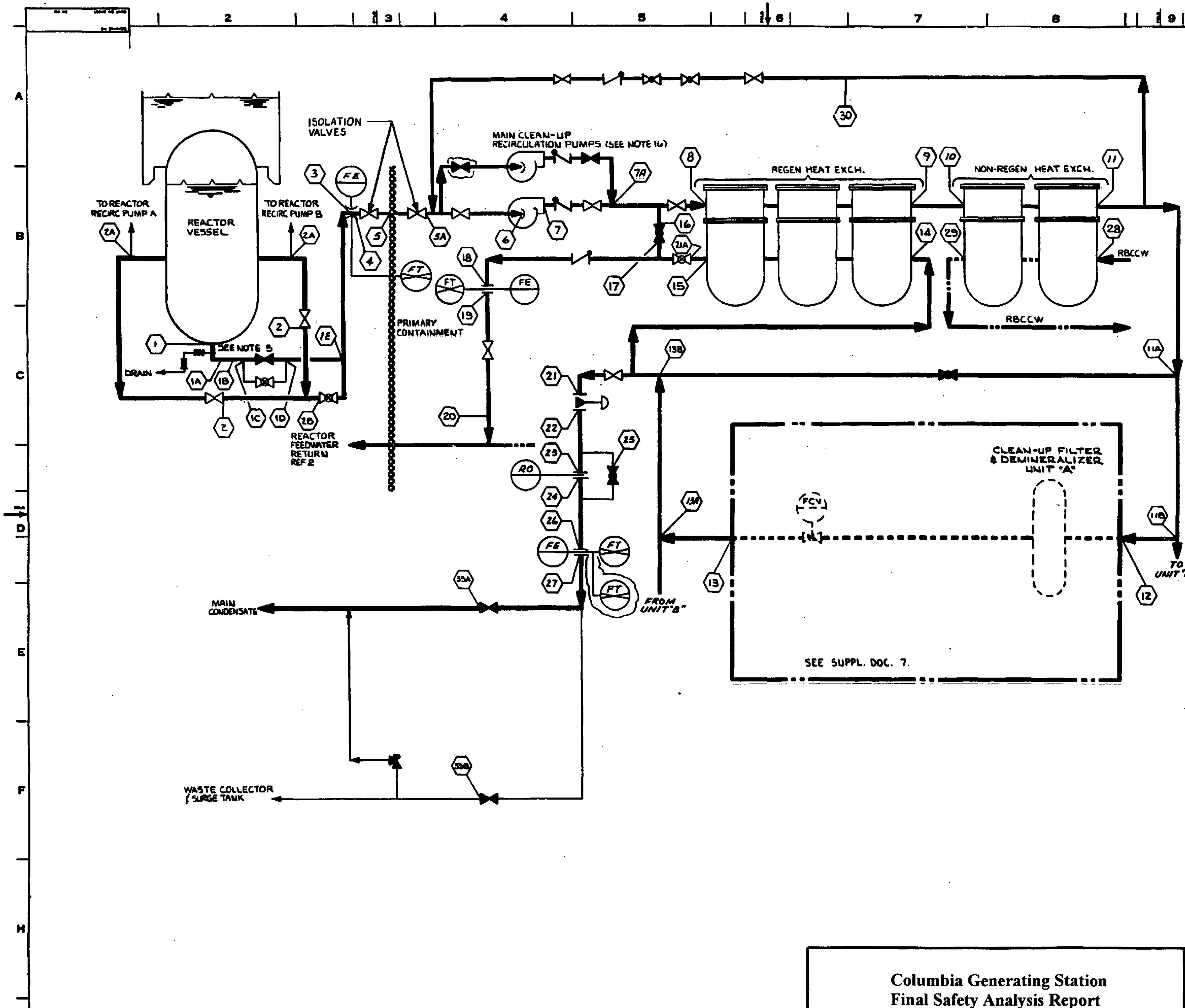


Columbia Generating Station
Final Safety Analysis Report

Reactor Water Cleanup System – P&ID

Draw. No. M523-3	Rev. 4	Figure 5.4-22.3

Form No. 960690ai
LDCN-06-000



NOTES:

1. POSITIONS 28 & 29 INDICATE CLEANUP CONDITIONS WITH VARIOUS RBCCW INLET TEMPERATURES.
2. FOR BACKWASH & PRECOAT FREQUENCY SEE SUPPL. DOC. 4 OR 8.
3. FOR DETAILS ON FILTER & DEMINERALIZER SYSTEM SEE PROCESS FLOW DIAGRAM FILTER/DEMINERALIZER SUPPL. DOC. 7.
4. VALVES ARE SHOWN IN THEIR NORMAL OPERATING POSITION.
5. FOR DETAILS OF THE REACTOR VESSEL BOTTOM DRAIN CONNECTIONS SEE SUPPL. DOC. 6.
6. THE MAXIMUM ALLOWABLE PIPE FRICTION DROP FOR THE SIZING OF THE CLEANUP RECIRCULATION PUMPS' SUCTION PIPING FROM POSITIONS 1 THROUGH 4 SHALL BE CONTROLLED BY MODE "B" AND THE NPSH SHALL NOT BE BELOW THE MINIMUM AS SHOWN. MODE "A" SHALL CONTROL THE SIZING OF THE DISCHARGE PIPING AND THE MAXIMUM ALLOWABLE PIPE FRICTION DROP AS SHOWN SHOULD NOT BE EXCEEDED.
7. FOR DATA PERTAINING TO NUMBERS WITHIN HEXAGONS, REFER TO PROCESS DATA SUPPL. DOC. 1.
8. MODE "A" DESIGN BASIS FOR HEAT EXCHANGERS AND NO FLOW IS REQUIRED AT POSITION 16, 17, 21 THROUGH 27.
9. MODE "B" DESIGN BASIS FOR MAIN CLEAN-UP PUMP (MAX. CAPACITY AND MIN. NPSH) AND SIZING OF MAIN PUMPS SUCTION PIPING. NO FLOW REQ'D IN MODE "B" POSITION 8 THROUGH 15 & 21 THROUGH 29.
10. DELETED

NOTES CONTINUED ON SHEET NO. 2

SUPPLEMENTAL DOCUMENTS UNDER THE FOLLOWING IDENTITIES ARE TO BE USED IN CONJUNCTION WITH THIS DRAWING.

		REFERENCE DESIGNATOR
1.	REACTOR WATER CLEAN-UP SYS PROCESS DATA	633-1030
2.	REACTOR WATER CLEAN-UP SYSTEM P&ID	633-1010
3.	REACTOR WATER CLEAN-UP SYS DESIGN SPEC.	633-4010
4.	RADWASTE SYSTEM PD	611-1020
5.	REACTOR SYSTEM OUTLINE DWG.	A62-2050
6.	REACTOR VESSEL OUTLINE DWG.	B13-D003
7.	FILTER/DEMIN SUBSYS DEVICE LIST ITEM 201	633-2001
8.	RADIOACTIVE WASTE	A62-4110
9.	DELETED	
10.	DELETED	

Columbia Generating Station
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Reactor Water Cleanup System Process Diagram

Draw. No. 02G33-04,20,1 | Rev. 7 | Figure 5.4-23.1

MODE A NORMAL OPERATION R.S.S. 1020, PSIA SEE NOTES 1,3 AND 9 ON SH. 1

LOCATION	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19
FLOW, GPM	63	145	352.	352.	352.	352	352	358.	287.	287.	276.	134.	134.	269.	318.	/	/	318.	318.
TEMP. F.	533.	533.	533.	533.	533.	523.	524.	524.	230.	230.	120.	120.	120.	120.	437.	/	/	437.	437.
MAXIMUM PRESSURE			1.			165.	TDH	10.		10.		59.		10.				4.	
DROP PSID	ALLOWABLE PIPE FRICTION DROPS (SEE NOTE 6 ON SH.1) 30.																		

(SEE NOTE 14)

LOCATION	20	21	22	23	24	25	26	27	28	29	28	29	28	29	28	29	30.		
FLOW, GPM	318.	/	/	/	/	/	/	/	468.	475.	516.	524.	610.	678.	678.	686.	7.		
TEMP. F.	437	/	/	/	/	/	/	/	85.	150.	95.	154.	100.	150.	105.	150.	120.		
MAXIMUM PRESSURE									10.		10.		10.		10.				
DROP PSID																			

MODE B HOT SHUTDOWN OPERATION (WITH LOSS OF RPV RECIRC PUMPS). R.S.S. 1003, PSIA SEE NOTES 1 AND 9 ON SH. 1.

LOCATION	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19
FLOW, GPM	162.	98.	358.	358	358.	358	358	/	/	/	/	/	/	/	/	358.	358.	358.	358.
TEMP. F.	540.	545.	543.	543.	543.	543.	544.	/	/	/	/	/	/	/	/	544.	544.	544.	544.
MAXIMUM PRESSURE			1.			165.	TDH									146.		4.	
DROPS PSID	ALLOWABLE PIPE FRICTION DROPS SEE NOTE 6 ON SH. 1 DELTA P																		

LOCATION	20	21	22	23	24	25	26	27	28	29									
FLOW, GPM	358.	/	/	/	/	/	/	/	/	/									
TEMP. F.	544.	/	/	/	/	/	/	/	/	/									
MAXIMUM PRESSURE																			
DROP PSID																			

DESIGN PRESSURE & TEMPERATURE GIVEN BELOW IS FOR INFORMATION ONLY AND IS THE BASIS FOR PIPING DESIGN. ESTIMATED LINE SIZES ARE FOR INFORMATION ONLY, ACTUAL LINE SIZES AS DETERMINED BY THE PIPING DESIGNER SHALL MEET THE PROCESS DATA HYDRAULIC REQUIREMENTS.

LOCATION	1--1A*	1A-1B**	1B-1E	1C-1D	2A-2B	2B-5A	5A-6	7-7A	7A-8	9-10	11-11B	11A-13B	11B-13A	13A-14	13A-33A	15-21A	21A-20	27-33B	
DESIGN PRESS.(PSIG)	1250.	1250.	1250.	1250.	1250.	1250.	1250.	1420.	1420.	1420.	1420.	1420.	1420.	1420.	1420.	1420.	***	1420.	
DESIGN TEMP. (DEG.F)	575.	575.	575.	575.	575.	575.	575.	575.	575.	575.	150.	150.	150.	150.	150.	575.	***	150.	
ESTIMATED LINE SIZE (IN)	2.0	2.5	4.0	1.0	4.0	6.0	3.0	3.0	4.0	4.0	4.0	4.0	3.0	4.0	4.0/6.0	4.0	4.0	4.0	

NOTES:

11. THE MINIMUM REQUIRED NPSH OF THE CLEANUP RECIRC. PUMPS IS 16.0 FEET BASED ON CONDITION SHOWN IN MODE B.
12. SEE PARAGRAPH 4.2.3 OF 25A1878 FOR STARTUP PROCEDURE.
13. DURING HOT STANDBY WITH ONE CLEANUP PUMP IN OPERATION. BLOWDOWN RATE IS APPROXIMATELY 126.0 GPM @ 545 DEGREES F.
14. COOLING WATER FLOW VALUES NOT TO BE EXCEEDED BY MORE THAN 6%.
15. ALL AUXILIARY PIPING IS DESIGNED TO 150. PSIG 150. DEG. F.
16. 100% CAPACITY PUMPS ARE ALTERNATED DURING OPERATIONS.

LEGEND

* LOCATION 1A IS THE POINT WHERE THE BOTTOM DRAIN LINE CONNECTION EXITS FROM THE C.R.D. HOUSING AREA.

** LOCATION 1B IS THE POINT WHERE THE BOTTOM DRAIN LINE CONNECTION EXITS FROM THE REACTOR VESSEL PEDESTAL.

*** TO SAME CONDITIONS AS THE FEEDWATER PIPING (BY OTHERS).

/ INDICATES CONDITIONS FOR 0 FLOW RATE.

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Reactor Water Cleanup System Process Diagram

Draw. No. 02G33-04,20,2	Rev. 5	Figure 5.4-23.2

Form No. 960690

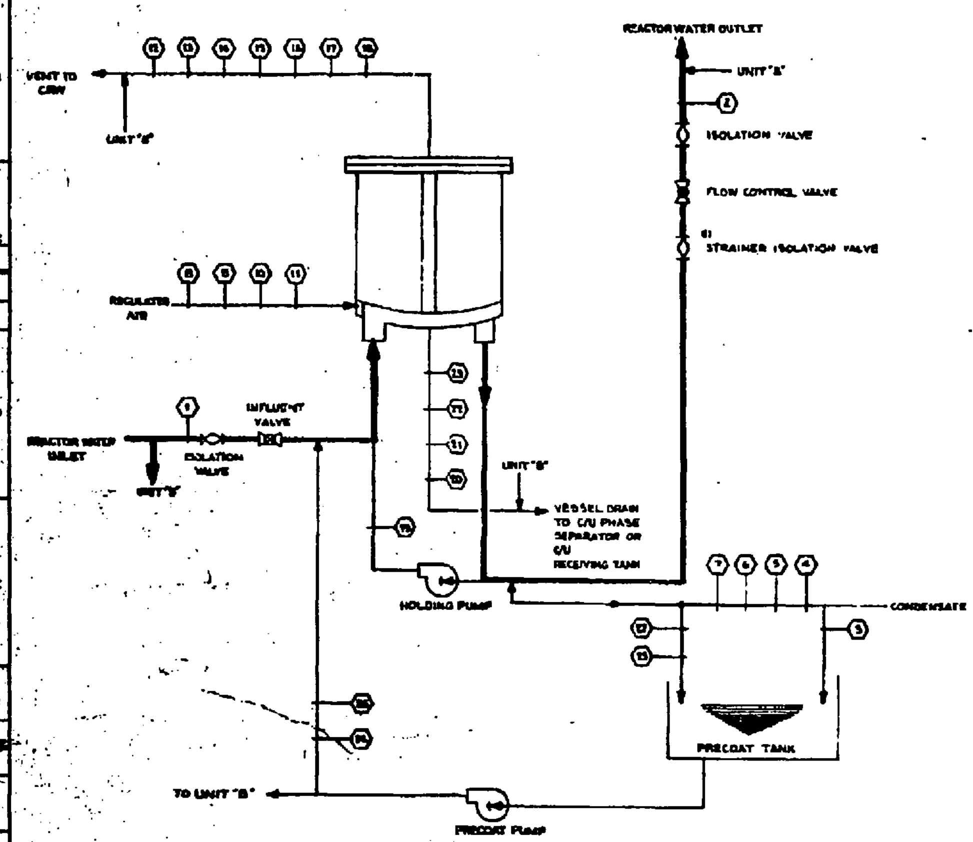

TABLE I (SEE TABLE V) — F/D AREA = 64 FT²

TYPE REQUIREMENT			STATION SERVICES									EXHAUST AIR HANDLING							PROCESS PIPE SIZING								
COMMODITY			CONDENSATE					SERVICE AIR				RADIOACTIVE AIR							REACTOR WATER OR COND	REACTOR WATER + F/D SOLIDS	COND + F/D SOLIDS	COND + F/D SOLIDS	COND + F/D SOLIDS	SLURRY COND + RESIN	CNDS	CNDS	CNDS
POSITION	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	20	21	22	23	24	25	26	27
PROCESS STEP	NORMAL OPER	NORMAL OPER	PRECOAT SLURRY	SLOW BKWSH	FILL & WASH	REWASH	FILL	SLOW BKWSH	SLOW DRAIN	FILL & WASH	REWASH	DRAIN	SLOW BKWSH	SLOW DRAIN	FILL & WASH	REWASH	DRAIN	FILL	HOLD	DRAIN	SLOW DRAIN	REWASH	DRAIN	RESIN PRECOAT	PRECOAT RETURN	PRECOAT RETAIN	PRECOAT RETAIN RETURN
PRESSURE, PSIG	SEE NOTE 1	SEE NOTE 1	25	25	25	25	25	20	20	10	20	5	5	5	5	5	5	5	[illegible]	20	20	20	20	50	50	50	50
TEMPERATURE, °F			80	80	80	80	80	60	60	60	60	60	60	60	60	60	60	60	[illegible]	120	80	80	80	80	80	80	80
FLOW RATE			25 GPM	[illegible]	62 GPM	63 GPM	125 GPM	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	125 GPM	62 GPM	62 GPM	125 GPM	125 GPM	100 GPM	100 GPM
FLOW TIME, MIN.	—	—	1.4	10	2	2	10	10	10	2	2	5	10	10	2	2	5	10	—	5	10	2	8	15	15	1	1
TOTAL FLOW	—	—	35 GAL	125 GAL	125 GAL	125 GAL	125 GAL	900 SCF	900 SCF	[illegible] SCF	[illegible] SCF	46 SCF	540 SCF	900 SCF	900 SCF	[illegible] SCF	46 SCF	16 SCF	—	125 GAL	125 GAL	125 GAL	125 GAL	125 GAL	125 GAL	100 GAL	100 GAL
TOTAL LBS SOLIDS	—	—	11.5 ●	—	—	—	—	—	—	—	—	—	—	—	—	—	—	—	—	16 ●	RESIDUAL	RESIDUAL	RESIDUAL	11.5 ●	—	—	—

● SEE NOTE 2

TABLE II (SEE TABLE V) — F/D AREA = 33 FT²

TYPE REQUIREMENT			STATION SERVICES									EXHAUST AIR HANDLING							PROCESS PIPE SIZING								
COMMODITY			CONDENSATE					SERVICE AIR				RADIOACTIVE AIR							REACTOR WATER OR COND	REACTOR WATER + F/D SOLIDS	COND + F/D SOLIDS	COND + F/D SOLIDS	COND + F/D SOLIDS	SLURRY COND + RESIN	CNDS	CNDS	CNDS
POSITION	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	20	21	22	23	24	25	26	27
PROCESS STEP	NORMAL OPER.	NORMAL OPER.	PRECOAT SLURRY	SLOW BKWSH	FILL & WASH	REWASH	FILL	SLOW BKWSH	SLOW DRAIN	FILL & WASH	REWASH	DRAIN	SLOW BKWSH	SLOW DRAIN	FILL & WASH	REWASH	DRAIN	FILL	HOLD	DRAIN	SLOW DRAIN	REWASH	DRAIN	RESIN PRECOAT	PRECOAT RETURN	PRECOAT RETAIN	PRECOAT RETAIN RETURN
PRESSURE, PSIG	SEE NOTE 1	SEE NOTE 1	25	15	15	25	25	20	20	10	20	0	5	5	5	5	10	5	[illegible]	20	20	20	20	35	35	35	35
TEMPERATURE, °F			80 TO 120	80	80	80	80	60	60	60	60	60	60	60	60	60	60	60	[illegible]	120	80	80	80	80	80	80	80
FLOW RATE			40 GPM	[illegible]	[illegible]	100 GPM	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	2.1 SCFM	10 GPM	5.5 GPM	6 GPM	[illegible]	30 GPM	[illegible]	260 GPM	130 GPM	[illegible]
FLOW TIME, MIN.	—	—	1.5	10	2	2	10	10	10	2	2	3	10	10	2	2	3	10	—	8	10	2	3	15	130	1	1
TOTAL FLOW	—	—	80 GAL	160 GAL	210 GAL	210 GAL	360 GAL	[illegible] SCF	1560 SCF	340 SCF	340 SCF	21 SCF	1540 SCF	1560 SCF	510 SCF	340 SCF	21 SCF	21 SCF	—	160 GAL	160 GAL	100 GAL	360 GAL	[illegible]	[illegible]	[illegible]	136 GAL
TOTAL LBS SOLIDS	—	—	18 ●	—	—	—	—	—	—	—	—	—	—	—	—	—	—	—	—	23 ●	RESIDUAL	RESIDUAL	RESIDUAL	18 ●	—	—	—

● SEE NOTE 2

TABLE III (SEE TABLE V) — F/D AREA = 126 FT²

TYPE REQUIREMENT			STATION SERVICES									EXHAUST AIR HANDLING							PROCESS PIPE SIZING								
COMMODITY			CONDENSATE					SERVICE AIR				RADIOACTIVE AIR							REACTOR WATER OR COND	REACTOR WATER + F/D SOLIDS	COND + F/D SOLIDS	COND + F/D SOLIDS	COND + F/D SOLIDS	SLURRY COND + RESIN	CNDS	CNDS	CNDS
POSITION	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	20	21	22	23	24	25	26	27
PROCESS STEP	NORMAL OPER	NORMAL OPER	PRECOAT SLURRY	SLOW BKWSH	FILL & WASH	REWASH	FILL	SLOW BKWSH	SLOW DRAIN	FILL & WASH	REWASH	DRAIN	SLOW BKWSH	SLOW DRAIN	FILL & WASH	REWASH	DRAIN	FILL	HOLD	DRAIN	SLOW DRAIN	REWASH	DRAIN	RESIN PRECOAT	PRECOAT RETURN	PRECOAT RETAIN	PRECOAT RETAIN RETURN
PRESSURE, PSIG	SEE NOTE 1	SEE NOTE 1	25	25	25	25	25	20	10	10	20	0	5	5	5	5	0	5	[illegible]	20	20	20	20	35	35	35	35
TEMPERATURE, °F			80 TO 120	80	80	80	80	60	60	60	60	60	60	60	60	60	60	60	[illegible]	120	80	80	80	80	80	80	80
FLOW RATE			50 GPM	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
FLOW TIME, MIN.	—	—	1.8	10	2	2	10	10	10	2	2	3	10	10	2	2	3	10	—	5	10	2	3	15	15	1	1
TOTAL FLOW	—	—	90 GAL	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	—	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
TOTAL LBS SOLIDS	—	—	15 ●	—	—	—	—	—	—	—	—	—	—	—	—	—	—	—	—	21 ●	RESIDUAL	RESIDUAL	RESIDUAL	25 ●	—	—	—

● SEE NOTE 2

TABLE IV (SEE TABLE V) — F/D AREA = 157 FT²

TYPE REQUIREMENT			STATION SERVICES									EXHAUST AIR HANDLING							PROCESS PIPE SIZING								
COMMODITY			CONDENSATE					SERVICE AIR				RADIOACTIVE AIR							REACTOR WATER OR COND	REACTOR WATER + F/D SOLIDS	COND + F/D SOLIDS	COND + F/D SOLIDS	COND + F/D SOLIDS	SLURRY COND + RESIN	CNDS	CNDS	CNDS
POSITION	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	20	21	22	23	24	25	26	27
PROCESS STEP	NORMAL OPER.	NORMAL OPER.	PRECOAT SLURRY	SLOW BKWSH	FILL & WASH	REWASH	FILL	SLOW BKWSH	SLOW DRAIN	FILL & WASH	REWASH	DRAIN	SLOW BKWSH	SLOW DRAIN	FILL & WASH	REWASH	DRAIN	FILL	HOLD	DRAIN	SLOW DRAIN	REWASH	DRAIN	RESIN PRECOAT	PRECOAT RETURN	PRECOAT RETURN	PRECOAT RETAIN RETURN
PRESSURE, PSIG	SEE NOTE 1	SEE NOTE 1	25	25	25	25	25	20	20	20	20	5	5	5	5	5	5	5	[illegible]	20	20	20	20	50	[illegible]	[illegible]	[illegible]
TEMPERATURE, °F			80	60	80	80	80	60	60	60	60	60	60	60	60	60	60	60	[illegible]	120	80	80	80	80	80	80	80
FLOW RATE			90 GPM	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	30 GPM	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
FLOW TIME, MIN.	—	—	2	4	2	2	4	4	2.5	2	2	2.9	4	2.9	2	2	2.9	4	—	2.5	2.9	2	7.5	17	15	[illegible]	[illegible]
TOTAL FLOW	—	—	100 GAL	310 GAL	170 GAL	310 GAL	300 GAL	[illegible]	600 SCF	[illegible]	[illegible] SCF	[illegible]	950 SCF	600 SCF	[illegible]	[illegible] SCF	[illegible] SCF	42 SCF	—	340 GAL	340 GAL	310 GAL	340 GAL	[illegible]	[illegible]	[illegible]	[illegible]
TOTAL LBS SOLIDS	—	—	32 ●	—	—	—	—	—	—	—	—	—	—	—	—	—	—	—	—	42 ●	RESIDUAL	RESIDUAL	RESIDUAL	30 ●	[illegible]	[illegible]	[illegible]

● SEE NOTE 2

NOTES:

1. FOR THESE VALUES AND REMAINDER OF SYSTEM VALUES: SEE REACTOR WATER CLEAN-UP SYSTEM PD.
2. DRY WEIGHT VALUES: AS SHIPPED RESINS CONTAIN 50% MOISTURE (APPROXIMATE).

TABLE V

TABLE	FCP
I	7B1E397 GROUP 1 / 7B2E397 GROUP 1
II	[illegible] GROUP 2 / [illegible] GROUP 2 / [illegible] GROUP 3
III	[illegible] GROUP 3 / [illegible] GROUP 3 / [illegible] GROUP 3 / [illegible] GROUP 3
IV	[illegible] GROUP 3

FILTER/DEMINERALIZER BACKWASH & PRECOATING SEQUENCE

FUNCTIONS:

- SERVICE
- ISOLATE & HOLD
- DRAIN
- SLOW BACKWASH
- SLOW DRAIN
- FILL & WASH
- REWASH
- DRAIN
- FILL
- PRECOAT
- PRECOAT RETURN
- HOLD UNTIL RETURN TO SERVICE
- SERVICE

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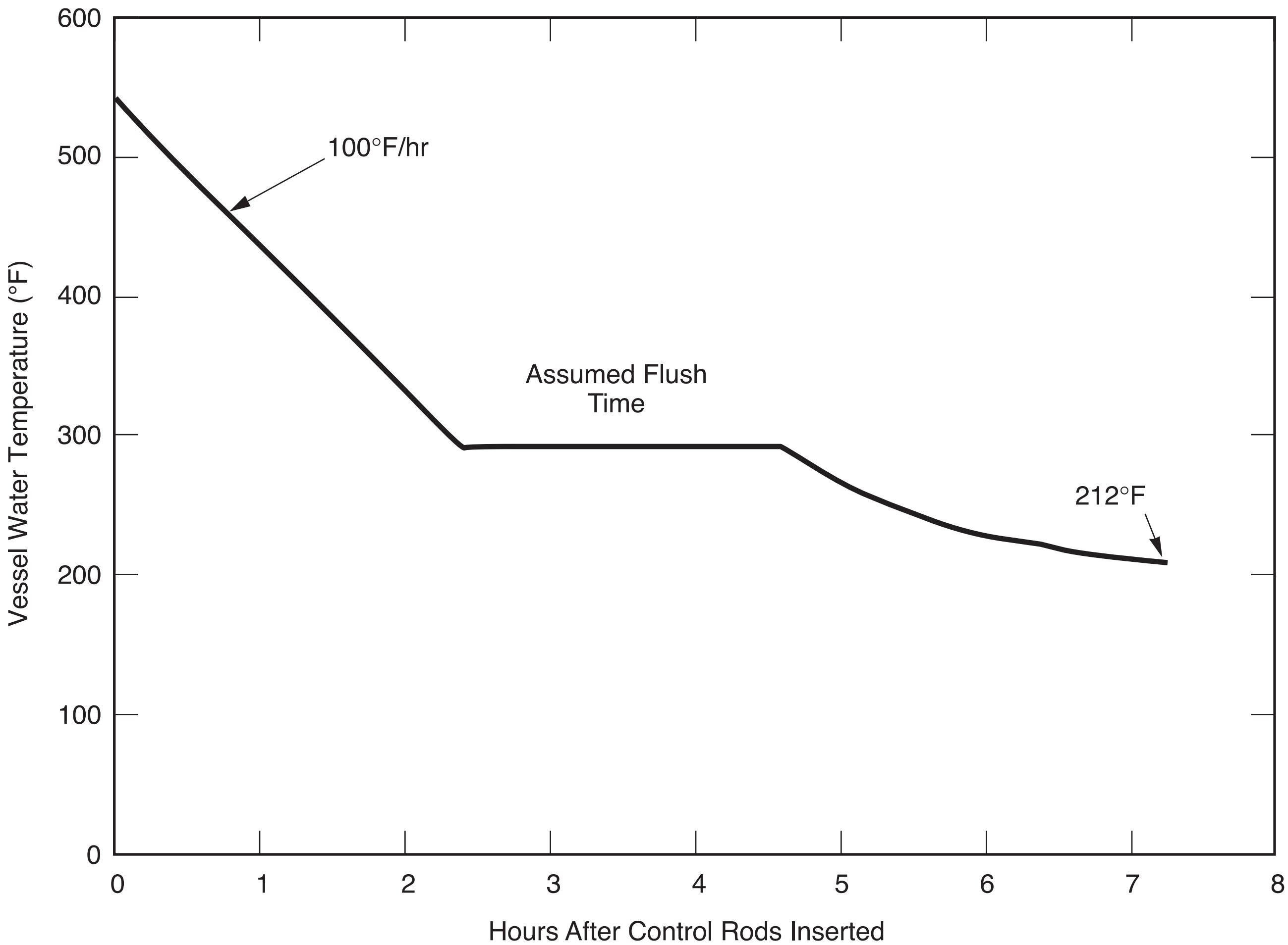
Filter/Demineralization System - P&ID

Draw. No.	02G33-04,14,1	Rev.	5	Figure	5.4-24



Vessel Coolant Temperature Versus Time
(One Heat Exchanger Available)

Draw. No.	960690.92	Rev.		Figure	5.4-25