5.4 COMPONENT AND SUBSYSTEM DESIGN

5.4.1 REACTOR RECIRCULATION PUMPS

5.4.1.1 Safety Design Bases

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

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

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

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

5.4.1.2 Power Generation Design Bases

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

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

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

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

5.4.1.3 Description

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

The recirculated coolant consists of saturated water from the steam separators and dryers that has been subcooled by incoming feedwater. This water passes down the annulus between the reactor vessel wall and the core shroud. A portion of the coolant flows from the vessel, through the two external recirculation loops, and becomes the driving flow for the jet pumps. Each of the two external recirculation loops discharges high pressure flow into an external manifold from which individual recirculation inlet lines are routed to the jet pump risers within the reactor vessel. The remaining portion of the coolant mixture in the annulus becomes the driven flow for the jet pumps. This flow enters the jet pump at suction inlets and is accelerated by the driving flow. The flows, both driving and driven, are mixed in the jet pump throat section and result in partial pressure recovery. The balance of recovery is obtained in the jet pump diffuser section (see Figure 5.4-5). The adequacy of the total flow to the core is discussed in Section 4.4.

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

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

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

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

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

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

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

The recirculation pump shaft seal assembly consists of two individual seals built into a cartridge or cartridges which can be readily replaced without removing the motor from the pump. The seal assembly is designed to require minimum maintenance over a long period of time, regardless of whether the pump is stopped or is operating at various speeds with water at various pressures and temperatures. Each seal is designed for a life of one year based on a 90 percent probability factor. Each individual seal in the cartridge is capable of sealing against pump design pressure so that any one seal can adequately limit leakage in the event that the other seal should fail. A breakdown orifice is provided in the pump casing to reduce leakage in the event of a gross failure of both shaft seals. Provision is made for monitoring the pressure drop across each individual seal as well as the cavity temperature of each seal. Provision is also made for piping the seal leakage to a flow measuring device.

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

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

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

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

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

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

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

The recirculation piping, valves, and pumps are supported by hangers to avoid the use of piping expansion loops that would be required if the pumps were anchored. In addition, the recirculation loops are provided with a system of restraints designed so that reaction forces associated with any split or circumferential break do not jeopardize primary containment integrity. This restraint system provides adequate clearance for normal thermal expansion movement of the loop. Because possible pipe movement is limited to slightly more than the clearance required for thermal expansion movement, no impact loading on limit stops is considered. A more detailed discussion of the recirculation piping restraints can be found in Section 3.6.

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

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

5.4.1.4 Safety Evaluation

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

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

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

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

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

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

5.4.1.5 Inspection and Testing

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

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

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

5.4.2 STEAM GENERATORS (PWR)

Not applicable to this BWR.

5.4.3 REACTOR COOLANT PIPING

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

5.4.4 MAIN STEAMLINE FLOW RESTRICTORS

5.4.4.1 Safety Design Bases

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

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

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

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

(4) To provide trip signal for MSIV closure.

5.4.4.2 Description

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

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

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

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

5.4.4.3 Safety Evaluation

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

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

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

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

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

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

5.4.4.4 Inspection and Testing

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

5.4.5 MAIN STEAMLINE ISOLATION SYSTEM

5.4.5.1 Safety Design Bases

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

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

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

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

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

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

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

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

5.4.5.2 Description

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

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

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

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

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

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

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

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

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

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

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

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

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

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

5.4.5.3 Safety Evaluation

In a direct cycle nuclear power plant the reactor steam goes to the turbine and to other equipment outside the containment. Radioactive materials in the steam are released to the environs through process openings in the steam system or escape from accidental openings. A large break in the steam system can drain the water from the reactor core faster than it is replaced by feedwater.

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

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

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

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

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

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

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

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

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

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

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

5.4.5.4 Inspection and Testing

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

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

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

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

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

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

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

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

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

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

5.4.6 REACTOR CORE ISOLATION COOLING SYSTEM

5.4.6.1 Design Bases

5.4.6.1.1 Residual Heat and Isolation

5.4.6.1.1.1 Residual Heat

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

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

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

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

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

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

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

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

Other isolation bases are defined in Subsection 5.4.6.1.1.2.

5.4.6.1.1.2 Isolation

Isolation valve arrangements include the following:

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

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

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

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

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

5.4.6.1.2 Reliability, Operability, and Manual Operation

5.4.6.1.2.1 Reliability and Operability (Also see subsection 5.4.6.2.4)

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

A design flow functional test of the RCIC system may be performed during normal plant operation by drawing suction from the condensate storage tank and discharging through a full flow test return line to the condensate storage tank. The discharge valve to the reactor vessel (via feedwater system) remains closed during the test, and reactor operation remains undisturbed. All components of the RCIC System shall be capable of individual functional testing during normal plant operation. Control system decision shall provide automatic return from test to operating mode if system initiation is required. There are three exceptions: 1) Auto/manual initiation on the flow controller. This feature is required for operator flexibility during system operation. 2) Steam inboard/outboard isolation valves. Closure of either or both of these valves requires operator action to properly sequence their opening (see Subsection 5.4.6.2.5.1). An alarm sounds when either of these valves leaves the fully open position. 3) Other bypassed or otherwise deliberately rendered inoperable parts of the system which affect the capability to perform a safety function shall be automatically indicated in the control room at the system level. Capability shall exist to manually initiate indication of system inoperability for manual initiation of system level indication shall exist for items not readily automated.

5.4.6.1.2.2 Manual Operation (Also see Subsection 5.4.6.2.5.2)

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

5.4.6.1.3 Loss of Offsite Power

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

5.4.6.1.4 Physical Damage

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

5.4.6.1.5 Environment

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

5.4.6.2 System Design

5.4.6.2.1 General

5.4.6.2.1.1 Description

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

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

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

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

Following a reactor scram, steam generation will continue at a reduced rate due to the core fission product decay heat. At this time the turbine bypass system will divert the steam to the main condenser, and the feedwater system will supply the make-up water required to maintain reactor vessel inventory. In the event the reactor vessel is isolated, and the feedwater supply is unavailable, relief valves are provided to automatically (or remote manually) maintain vessel pressure within desirable limits. The water level in the reactor vessel will drop due to continued steam generation by decay heat.

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

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

5.4.6.2.1.2 Diagrams

The following diagrams are included for the RCIC Systems.

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

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

5.4.6.2.1.3 Interlocks

The following defines the various electrical interlocks:

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

5.4.6.2.2 Equipment and Component Description

5.4.6.2.2.1 Design Conditions

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

(1) One 100% capacity turbine and accessories

(2) One 100% capacity pump assembly and accessories

(3) Piping, valves, and instrumentation for:

a. Steam supply to the turbine

b. Turbine exhaust to the suppression pool

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

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

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

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

5.4.6.2.2.2 Design Parameters

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

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

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

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

5.4.6.2.3 Applicable Codes and Classifications

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

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

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

5.4.6.2.4 System Reliability Considerations

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

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

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

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

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

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

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

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

5.4.6.2.5 System Operations

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

5.4.6.2.5.1 Automatic Operation

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

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

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

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

(3) Verify power is available to all components.

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

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

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

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

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

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

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

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

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

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

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

(15) Reset the turbine trip throttle valve.

(16) Stop the barometric condenser vacuum pump.

(17) Close the cooling water supply valve F046.

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

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

5.4.6.2.5.2 Test Loop Operation

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

(1) Verification made in steps 1 through 10 of Subsection 5.4.6.2.5.1 shall be completed.

(2) All motor operated valves shall be positioned as shown on Dwgs. M-149, Sh. 1 and M-150, Sh. 1.

(3) Open F059 and F022 fully.

(4) Start barometric condenser vacuum pump.

(5) Open F046.

(6) Open F045.

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

(8) Adjust F022 to obtain a pump discharge pressure of 300 psig.

(9) Observe turbine RPM on speed indicator.

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

(11) Further adjust F022 to simulate reactor pressure plus line losses to reactor pressure at time of test or actual line pressure drop to reactor (if available) plus reactor pressure.

(12) While turbine is running, check and record the following:

a. Pump Suction Pressure

b. Pump Discharge Pressure

c. Turbine Steam Exhaust Pressure

d. Turbine Steam Inlet Pressure

e. Pump Flow

f. Turbine Speed

(13) When the test is completed, manually trip the turbine.

(14) When the turbine speed indicator reaches "0" RPM, close the test bypass valve to the HPCI test return line which then goes to the condensate storage tank F022.

(15) Close redundant shut off valve E41-F011 to the HPCI test return line which then goes to the condensate storage tank.

(16) Follow steps 14 through 19 of Subsection 5.4.6.2.5.1.

5.4.6.2.5.3 Steam Condensing (Hot Standby) Operation

This section has been deleted.

5.4.6.2.5.4 Limiting Single Failure

The most limiting single failure with the RCIC and its HPCI backup system is the failure of HPCI. If the capacity of RCIC System with a HPCI failure is adequate to maintain reactor water level, the operator follows Subsection 5.4.6.2.5.1. If however, the RCIC capacity is inadequate, Subsection 5.4.6.2.5.1 applies, but additionally the operator may also initiate the ADS system described in Subsection 6.3.2.

5.4.6.3 Performance Evaluation

The analytical methods and assumptions in evaluating the RCIC system are presented in Chapter 15 and Appendix 15A. The RCIC system provides the flows required from the analysis (see Dwg. M1-E51-81, Sh. 1) within a 30 second interval based upon considerations noted in Subsection 5.4.6.2.4.

5.4.6.4 Preoperational Testing

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

5.4.6.5 Safety Interfaces

The Balance of Plant-GE Nuclear Steam Supply System safety interfaces for the Reactor Core Isolation Cooling System are: (1) preferred water supply from the condensate storage tank; (2) all associated safety-related wire, cable, piping, sensors, and valves which lie outside the Nuclear Steam Supply System scope of supply. The air supply for solenoid actuated valves is a non-safety related interface.

5.4.7 RESIDUAL HEAT REMOVAL SYSTEM

5.4.7.1 Design Bases

The RHR system is comprised of two independent loops. Each loop contains two motor-driven pumps, a heat exchanger, piping, valves, instrumentation, and controls. The RHR heat exchangers are cooled by RHR service water. Each loop takes suction from the suppression pool and is capable of discharging water to the reactor vessel via a connection to the reactor recirculation loop, or back to the suppression pool via a full flow test line. Each loop can also take suction from a common line from one reactor recirculation loop. Each loop can take suction from and discharge to the fuel pool cooling system. Both loops can discharge to wetwell and drywell spray spargers. Also, one loop can discharge to the reactor vessel head spray line.

5.4.7.1.1 Functional Design Basis

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

5.4.7.1.1.1 Residual Heat Removal Mode (Shutdown Cooling Mode)

1. The function design basis of the shutdown cooling mode is to have the capability to remove heat from the reactor primary system so that the reactor coolant temperature can be reduced to 125°F after reactor shutdown, once the main condenser can no longer be used as effective heat sink.

2. With one loop in service the shutdown cooling mode of the RHR System is capable of cooling the reactor cooling system to 200 within 24 hours after shutdown. If normal shutdown cooling can not be established, the alternative shutdown cooling systems described in section 15.2.9 are capable of acceptable shutdown heat removal.

A cross-tie line exists between the loop ‘A’ and ‘B’ LPCI injection lines and the RHR Shutdown Cooling suction line. The 1 inch line provides a positive pressure drop across the LPCI injection check valves HV151F050A and HV151F050B to maintain the valves in a closed position during normal operation. Should leakage occur past the seats of the LPCI injection check valves, the excess fluid is diverted through the cross-tie line and returned to the Reactor Recirculation System (RRS) via the RHR System Shutdown Cooling suction line.

During Shutdown Cooling (SDC) operation a portion of the flow is diverted from ‘A’ and/or ‘B’ loop injection lines through the cross-tie line. The quantity of flow diverted is less than the excess flow available during SDC operations and does not impede the capabilities of the RHR System from performing its SDC function.

5.4.7.1.1.2 Low Pressure Coolant Injection (LPCI) Mode

The functional design basis of the LPCI mode is to flood the reactor core when reactor pressure is low. One or more of the four motor-driven RHR pumps are used to pump water from the suppression pool into the reactor vessel via the recirculation loop. This ECCS mode is automatically initiated by low reactor water level (Level 1) or high drywell pressure. Equipment characteristics used for the LOCA analysis in FSAR Subsection 6.3.3.7.1 are described in Subsection 6.3.2.2.4. RHRSW can be aligned to an RHR heat exchanger in this mode of operation to ensure that the long term peak suppression pool temperature following a design basis LOCA remains within design limits, as evaluated in the SSES containment analysis described in section 6.2.1 of the FSAR. With one LPCI injection pump in service in an RHR loop, an RHR heat exchanger may be aligned to support containment cooling by limiting LPCI injection flow to 10,000 gpm and directing flow through the RHR heat exchanger by closing the HV151F048A(B) valve. Only one RHR heat exchanger is credited for long term cooling in the SSES containment analysis.

A cross-tie line exists between the loop ‘A’ and ‘B’ LPCI injection lines and the RHR Shutdown Cooling suction line. The 1 inch line provides a positive pressure drop across the LPCI injection check valves HV151F050A and HV151F050B to maintain the valves in a closed position during normal operation. Should leakage occur past the seats of the LPCI injection check valves, the excess fluid is diverted through the cross-tie line and returned to the Reactor Recirculation System (RRS) via the RHR System Shutdown Cooling suction line.

During LPCI operation a portion of the flow is diverted from the ‘A’ and/or ‘B’ loop injection lines through the cross-tie line. The quantity of flow diverted is less than the excess flow available during either one or two pump LPCI operations and does not impede the capabilities of the RHR System from performing its LPCI safety function.

5.4.7.1.1.3 Suppression Pool Cooling Mode

The functional design basis of the suppression pool cooling mode is sufficient cooling capacity to ensure that the long-term peak suppression pool temperature following a design basis LOCA remains within design basis limits. This mode may be used during normal plant operation, during a transient, or after a LOCA to remove heat from the containment. This mode is initiated and terminated via remote manual control from the control room. See Subsection 6.2.1.1.3 and Table 6.2-6.

5.4.7.1.1.4 Containment Spray Cooling Mode

The functional design basis of the containment spray cooling mode is to provide a redundant means of condensing vapor and cooling the drywell and suppression pool vapor space to control the containment pressure within design limits.

5.4.7.1.1.5 Reactor Steam Condensing Mode

This section has been deleted.

5.4.7.1.1.6 Fuel Pool Cooling Mode

The functional design basis for the fuel pool cooling mode is as follows:

a) The RHRFPC mode is designed and operated to provide cooling such that the fuel pool will be maintained at or below 125°F when the Emergency Heat Load (EHL) is resident in an isolated fuel pool. The EHL can be removed with a RHRSW inlet temperature of 89°F with only one RHR pump and heat exchanger. For crosstied fuel pools, one RHR pump and heat exchanger in one unit in combination with the normal Fuel Pool Cooling System from the adjacent unit is sufficient to maintain the fuel pools at or below 125°F with the EHL resident in one fuel pool and fuel at the scheduled offload rate in the other fuel pool. This function is described in Section 9.1.3.1b and 9.1.3.2.

b) The RHRFPC mode is designed and operated to provide sufficient cooling to prevent fuel pool boiling in the event that a seismic event causes an extended loss of both units' normal fuel pool cooling systems. This capability exists for both crosstied and isolated fuel pools.

When one RHR pump is operated in the RHRFPC mode, the spent fuel pool level must be raised to a minimum level above the weirs in order to support the design flowrate of this mode. Additional details describing this mode of RHR are contained in Sections 5.4.7.2.6c, 9.1.3.1c, 9.1.3.2, and 9.1.3.3.

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

The low pressure portions of the RHR system, are isolated from full reactor pressure whenever the primary system pressure is above the RHR system design pressure. See Subsection 5.4.7.1.3 for further details. In addition, automatic isolation may occur for reasons of vessel water inventory retention which is unrelated to line pressure rates. (See Subsection 5.2.5 for an explanation of the Leak Detection System and the isolation signals.) Reactor Coolant pressure boundary valves are subject to inservice inspection leakage testing requirements as provided in 10CFR50.55a (see Subsection 3.9.6).

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

5.4.7.1.3 Design Basis For Pressure Relief Capacity

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

(1) Thermal relief only

(2) Valve bypass leakage only

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

Transients are treated by items (1) and (3); item (2) above has resulted from an excessive leak past isolation valves. RHR System pressure relief valves are set to assure that the maximum expected pressure from the worst case overpressure event does not exceed the ASME code allowable pressure for the ECCS piping.

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

In addition a high pressure check valve will close to prevent reverse flow from the reactor if the pressure should increase. Relief valves in the discharge piping are sized to account for leakage past the check valve.

5.4.7.1.4 Design Basis With Respect to General Design Criteria 5

The RHR system for each unit does not share equipment or structures with the other nuclear unit except for the Spent Fuel Pools as discussed in Subsection 9.1.3.3. They also share the common Emergency Service Water System. Sharing of this system with respect to General Design Criteria 5 is discussed in Section 3.1.2.1.5.

5.4.7.1.5 Design Basis for Reliability and Operability

The design basis for the Shutdown Cooling mode of the RHR system is that this mode is controlled by the operator from the control room. The only operations performed outside of the control room for a normal shutdown are manual operations of local flushing water admission valves, which are the means of providing clean water to the shutdown portions of the RHR system. These portions are flushed with reactor water, suppression pool water or condensate to minimize the input of potentially harmful impurities to the reactor coolant system.

Two separate shutdown cooling loops are provided. The design basis is that, with both loops in operation, reactor coolant temperature will be reduced below 125°F after all control rods are inserted. This includes the time to depressurize the reactor, flush and preheat the RHR System. The reactor coolant can be brought below 212°F within 24 hours with only one loop in operation. With the exception of the shutdown suction, vessel head spray, and steam supply and condensate discharge lines, the entire RHR system is part of the ECCS and containment cooling systems, and is therefore required to be designed with redundancy, flooding protection, piping protection, power separation, etc. required of such systems. (see Section 6.3 for an explanation of the design bases for ECCS systems). Shutdown suction and discharge valves are required to be powered from both offsite and standby emergency power for purposes of isolation and shutdown following a loss of offsite power. In the event either of the two shutdown supply valves fail to operate, and the shutdown supply valves cannot be opened by hand, alternate shutdown cooling is established in accordance with plant procedures. If repairs are required to the shutdown suction valves, the line can be isolated using manual valve F067 provided containment access is possible. Residual heat is absorbed by the main condenser or by the suppression pool with pool cooling by the RHR system while repairs are in process.

5.4.7.1.6 Design Basis for Protection from Physical Damage

See Section 3.12 for discussion.

5.4.7.2 Systems Design

5.4.7.2.1 System Diagrams

All of the components of the RHR system are shown in the P&ID, Dwgs. M-151, Sh. 1, M-151, Sh. 2, M-151, Sh. 3, and M-151, Sh. 4. A description of the controls and instrumentation is presented in Subsection 7.3.1.1.

A process diagram and process data for the RHR System are shown in Dwgs. M1-E11-3, Sh. 1 and M1-E11-3, Sh. 2. All of the sizing modes of the system are shown in the process data. The FCD for the RHR system is provided in Chapter 7.

When the system is operated from the control room, interlocks are provided: (1) to prevent drawing vessel water to the suppression pool; (2) to prevent opening vessel suction valves above the suction line design pressure or the discharge line design pressure, with the pumps at shutoff head; (3) to prevent inadvertent opening of drywell spray valves while in shutdown; and (4) to prevent pump start (i.e. maintaining a pump trip signal) when suction valve(s) are not open.

5.4.7.2.2 Equipment and Component Description

a. System Main Pumps

The RHR main system pumps are motor-driven deepwell pumps with mechanical seals. The motors are water cooled by the Emergency Service Water System. The pumps are sized on the basis of the LPCI mode (Mode A) and the minimum flow mode (Mode J) on Dwgs. M1-E11-3, Sh. 1 and M1-E11-3, Sh. 2. Design pressure for the pump suction structure is 220 psig with a temperature range from 40 to 360°F. Design pressure for the pump discharge structure is 500 psig. The bases for the design temperature and pressure is maximum shutdown cut-in pressures and temperature, minimum ambient temperature, and maximum shutoff head. The pump pressure vessel is carbon steel, the shaft and impellers are stainless steel. The required pump NPSH can be obtained from the pump characteristic curves provided in Figure 6.3-119. Available NPSH is provided in Section 6.3.2.2.4.1.

b. Heat Exchangers

The RHR system heat exchangers are sized on the basis of the duty for the shutdown cooling mode (Mode F on Dwg. M1-E11-3, Sh. 1). All other uses of these exchangers require less cooling surface. The RHR heat exchanger data provided by GE is as follows:

Flow rates are 10,000 gpm (rated) on the shell side and 9,000 gpm (rated) on the tube side (service water side). Rated inlet temperature is 125°F shell side and 85°F tube side. The overall heat transfer coefficient is 218 BTU per hour per square foot per °F. The exchangers contain 7593 ft^2 of effective surface. Design temperature range of shell side is 40°F to 470°F and tube side is 32°F to 470°F. Design pressure is 450 psig on both sides, fouling factors are 0.0005 shell side and 0.002 tube side. The construction materials are carbon steel for the pressure vessel with 70-30 copper-nickel tubes and carbon steel tube sheet clad with copper nickel.

c. Valves

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

d. RHR Suction Strainers

Each 24" RHR pump suction line penetrates the vertical wall of the suppression pool, leading directly to a vertical "T” arrangement whose centerline is 23" from the pool wall. Two RHR high capacity stacked disk suction strainers are mounted on the 24" "T" for each RHR pump. These strainers replaced the original conical design. Each of the two strainers provides a flow area of 204 ft^2. The strainers have sufficient capacity to filter their design debris source term under worst case conditions while maintaining strainer pressure drop below the maximum value required to provide adequate NPSH and system flow. The design debris source term consists of conservative amounts of insulation, paint chips and other drywell debris that is assumed to be destroyed by LOCA jet forces and transported to the suppression pool through the downcomers. This debris is assumed to be filtered by the strainers along with corrosion products that would exist in the suppression pool prior to a LOCA. The stacked disk strainers are designed for a maximum pressure drop of 2.5 psi at a flow of 13,800 gpm while filtering their design debris source term. Correlations between the amount of debris filtered by the strainers and strainer pressure drop are based on testing performed on one of the Susquehanna strainers and NRC approved methodology outlined in NEDO-32686, “Utility Resolution Guide for ECCS Suction Strainer Blockage”. The suppression pools are cleaned and inspected periodically to maintain corrosion product amounts at acceptable levels and to confirm the absence of miscellaneous debris that would be a strainer blockage threat.

The strainer mesh size is 0.125” +/- 0.005” which screens out all particles greater than 1/8” nominal diameter. This criteria is used in conjunction with the design of the drywell and suppression pool containment spray nozzles, which have a free passage diameter of 0.125." Particles equal to or smaller than 1/8" in size would have no effect on RHR pump operation.

The minimum height of the suppression pool water level above the "T" centerline is 11'-11". The available and required NPSH are provided in Section 6.3.2.2.4.1.

The RHR suction strainers are shown in Figures 5.4-4A and 5.4-4B.

ESF Portions of the RHR System

The ECCS (LPCI) portions of the RHR system include those sections required to operate Modes A, B, and G on Dwg. M1-E11-3, Sh. 1.

The route includes suppression pool suction strainers, suction piping, RHR pumps, RHR heat exchangers, discharge piping injection valves, and drywell piping to the reactor recirc discharge lines.

Pool cooling components include pool suction strainers, suction piping, pumps, heat exchangers and pool return lines required to operate Mode D on Dwg. M1-E11-3, Sh. 1.

Containment spray components required for Modes C-1 and C-2 on Dwg. M1-E11-3, Sh. 1 are the same as pool cooling except that the spray headers replace the pool return lines.

5.4.7.2.3 Controls and Instrumentation

Controls and instrumentation for the RHR system are described in Chapter 7. The RHR system incorporates relief valves to protect the components and piping from inadvertent overpressure conditions. The relief valve set point, capacity and method of collection are shown in Table 5.4-3.

5.4.7.2.4 Applicable Codes and Classifications

Refer to Sections 3.9 and 3.10 for discussion of applicable codes and standards.

5.4.7.2.5 Reliability Considerations

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

A cross-tie line exists between the loop ‘A’ and ‘B’ LPCI injection lines and the RHR Shutdown Cooling suction line. The 1 inch line provides a positive pressure drop across the LPCI injection check valves HV151F050A and HV151F050B to maintain the valves in a closed position during normal operation. Should leakage occur past the seats of the LPCI injection check valves, the excess fluid is diverted through the cross-tie lien and returned to the Reactor Recirculation System (RRS) via the RHR System Shutdown Cooling suction line.

During LPCI or Shutdown Cooling (SDC) operations of a portion of the flow is diverted from the ‘A’ and/or ‘B’ loop injection lines through the cross-tie line. The quantity of flow diverted is less than the excess flow available during LPCI or SDC operations and does not impede the capabilities of the RHR System from performing its SDC or LPCI safety function.

5.4.7.2.6 Manual Action

a. Residual Heat Removal (Shutdown Cooling Mode)

In shutdown cooling operation, when vessel pressure is below the shutdown cooling cut-in permissive, the system is prewarmed and stagnant water is flushed to the condenser hotwell or radwaste via valves F040 and F049 which are operated from the control room. Following verification that acceptable differential temperatures exist between the RPV steam dome and the bottom head drain; between the reactor coolant contained in an idle recirculation loop and the RPV; and that the RHR lines are full, the RHRSW system is placed in service. With the RHR Heat exchanger inlet (F047) and bypass valves (F048) open and the outlet valve (F003) closed, an RHR pump is subsequently started with flow back to the RPV regulated by return valve F017. RHR flow is established through the Heat Exchanger by throttling open outlet valve F003. The cooldown rate of reactor coolant is controlled via valves F017 (total RHR system flow) and the F048 (heat exchanger bypass flow), and/or the F003 (heat exchanger flow) and the F047 (heat exchanger inlet). All operations are performed from the control room except for opening and closing of local flush water valves.

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

b. Steam Condensing

This section has been intentionally deleted.

c. Fuel Pool Cooling Mode

Operation of RHR in the fuel pool cooling mode requires manual actions to be performed both in the control room and locally. The system will also be required to be filled and vented, which will require the manipulation of various small manual valves. The filling operation may also include operation of the ESW system in the event the normal fill systems are unavailable. These actions are described in and controlled by plant procedures.

5.4.7.3 Performance Evaluation

Thermal performance of the RHR heat exchangers is based on the residual heat generated after rod insertion, a 125°F vessel outlet (exchanger inlet) temperature, and the flow of two loops in operation. Because shutdown is usually a controlled operation, maximum service water temperature less 10°F is used as the service water inlet temperature. These are nominal design conditions; if the service water temperature is higher, the exchanger capabilities are reduced and the shutdown time may be longer and vice versa.

5.4.7.3.1 Shutdown With All Components Available

No typical curve is included here to show vessel cooldown temperatures versus time due to the infinite variety of such curves that may be due to: (1) clean steam systems that may allow the main condenser to be used as the heat sink when nuclear steam pressure is insufficient to maintain steam air ejector performance; (2) the condition of fouling of the exchangers; (3) operator use of one or two cooling loops; (4) coolant water temperature; and (5) system flushing time. Since the exchangers are designed for the fouled condition with relatively high service water temperature, the units have excess capability to cool when first cut in at high vessel temperatures. Total flow and mix temperature must be controlled to avoid exceeding 100°F per hour cooldown rate.

5.4.7.3.2 Shutdown With Most Limiting Failure

Shutdown under conditions of the most limiting failure is a loss of the suction path for normal shutdown cooling. Reactor shutdown can be achieved using alternate shutdown cooling as described in Section 15.2.9. The capability of the heat exchanger for any time period is balanced against residual heat, pump heat, and sensible heat. The excess over residual heat and pump heat is used to reduce the sensible heat.

5.4.7.4 Preoperational Testing

The preoperational test program and startup test program as discussed in Chapter 14. will verify the capabilities of the system to provide the flows, pressures, condensing rates, cooldown rates, and reaction times required to perform all system functions as specified for the system or component in the System Data Sheets and Process Data.

5.4.8 REACTOR WATER CLEANUP SYSTEM

The reactor water cleanup system is an auxiliary system, a small part of which is part of the reactor coolant pressure boundary up to and including the outermost containment isolation valve. The other portions of the system are not part of the reactor coolant pressure boundary and are isolatable from the reactor.

5.4.8.1 Design Bases

5.4.8.1.1 Safety Design Bases

The RCPB portion of the RWCU system:

(1) Prevents excessive loss of reactor coolant, and

(2) Prevents the release of radioactive material from the reactor.

5.4.8.1.2 Power Generation Design Bases

The reactor water cleanup system:

(1) Removes solid and dissolved impurities from reactor coolant;

(2) Discharges excess reactor water during startup, shutdown, and hot standby conditions;

(3) Minimizes temperature gradients in the recirculation piping and vessel during periods when the main recirculation pumps are unavailable.

(4) Minimizes RWCU System heat loss; and

(5) Enables the major portion of the RWCU system to be serviced during reactor operation.

5.4.8.2 System Description

The reactor water cleanup system (see Dwgs. M-144, Sh. 1, M-144, Sh. 2, M-144, Sh. 3, Dwg. M1-G33-16, Sh. 1 and Dwg. M1-G33-18, Sh. 1 continuously purifies the reactor water. The system takes its suction from the inlet of the reactor recirculation pump and from the reactor pressure vessel bottom head. The processed water is returned to the reactor pressure vessel via the feedwater system, to the main condenser or radwaste.

The cleanup system can be operated at any time during planned operations, or it may be shut down. The cleanup system is classified as a primary Power Generation System. The cleanup system is not an Engineered Safety Feature.

The major equipment of the reactor water cleanup system is located in the reactor building. This equipment includes regenerative and nonregenerative heat exchangers, filter-demineralizers with regeneration equipment and cleanup pumps. The entire system is connected by associated valves and piping; controls and instrumentation provide proper system operation. Design data for the major pieces of equipment are presented in Table 5.4-2.

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

The temperature of the filter-demineralizer units is limited by the resin operating temperature. Therefore the reactor coolant must be cooled before being processed in the filter-demineralizer units. The regenerative heat exchanger transfers heat from the tubeside (hot process) to the shellside (cold process). The shellside flow returns to the reactor. The nonregenerative heat exchanger cools the process further by transferring heat to the Reactor Building Closed Cooling Water System. The nonregenerative heat exchanger is sized to maintain the required filter-demineralizer temperature, even when the effectiveness of the regenerative heat exchanger is partially reduced by diversion of a portion of the process to either the main condenser or the radwaste system.

The filter-demineralizer units (see Dwg. M-145, Sh. 1 are pressure precoat type filters using filter aid and finely ground, mixed ion-exchange medium. Spent resins are not regenerable and are sluiced from the filter-demineralizer unit to a backwash receiving tank from which they are transferred to the radwaste system for processing and disposal. To prevent resins from entering the reactor recirculation system in the event of failure of a filter-demineralizer resin support, a strainer is installed on the outlet of each filter-demineralizer unit. Each strainer has a control room alarm that is energized by high differential pressure. A bypass line is provided around the filter-demineralizer units for bypassing the units.

The cleanup recirc pumps are vertical sealless units, top-hung, with the motor below the pump. Two (2) pumps are provided, each with a capacity of 100% of system design flow. To prevent contaminants in the reactor water from reaching the cleanup recirc pump motors, a purge water subsystem (see Dwg. M-144, Sh. 3 is provided. Clean water of reactor quality is taken from the control rod drive hydraulic system and injected into the cleanup recirc pump motors. Monitoring instrumentation for the cleanup recirc pumps, as well as instrumentation and controls for the purge water subsystem are located on a local control panel.

In the event of low flow or loss of flow in the system, or outboard isolation valve G33-F004 is not fully open, when filter demineralizer is available, the RWCU holding pumps are automatically started and flow is maintained through each filter-demineralizer by its own holding pump. Sample points are provided in the common influent header and in each effluent line of the filter-demineralizer units for continuous indication and recording of system conductivity. High conductivity is annunciated in the control room. The influent sample point is also used as the normal source of reactor coolant grab samples. Sample analysis also indicates the effectiveness of the filter-demineralizer units.

The suction line of the RCPB portion of the RWCU system contains two motor-operated isolation valves which automatically close in response to high flow signals. (Sections 7.6 and 5.2 describe the Leak Detection System and it is summarized in Table 5.2-8.) This action prevents the loss of reactor coolant and release of radioactive material from the reactor.

The outboard isolation valve will automatically close to prevent damage of the filter-demineralizer resins if the outlet temperature of the non-regenerative heat exchanger is high. The outboard isolation valve will also close upon manual initiation of the Standby Liquid Control System. This prevents removal of liquid poison by the cleanup system should the SLCS be in operation. The RCPB isolation valves may be remote manually operated to isolate the system equipment for maintenance or servicing.

The outboard isolation valve of Unit 1 and the inboard isolation valve of Unit 2 will close upon placing the transfer switches in 'Emergency' position. The transfer switches are located on the respective Units Remote Shutdown Panels. This action also isolates the control and indication circuits located in the Control Room Panels for these valves.

Two remote manual-operated gate valves on the return lines to the reactor provide long term leakage control. Instantaneous reverse flow isolation is provided by check valves in the RWCU piping.

Operation of the reactor water cleanup system is controlled from the main control room. Resin-changing operations, which include backwashing and precoating, are controlled from a local control panel in the reactor building. The time required to remove a unit from the line, backwash and precoat is typically less than one hour.

A functional control drawing is provided in Section 7.7.

5.4.8.3 System Evaluation

The RCPB isolation valves and piping are designed to the requirements defined in Section 3.2 and the requirements of Subsection 7.3.1.1a.2.

5.4.9 MAIN STEAM LINES AND FEEDWATER PIPING

5.4.9.1 Safety Design Bases

In order to satisfy the safety design bases, the main steam and feedwater lines have been designed:

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

(2) With suitable accesses to permit in-service testing and inspections.

5.4.9.2 Power Generation Design Bases

In order to satisfy the design bases:

(1) The main steamlines have been designed to conduct steam from the reactor vessel over the full range of reactor power operation.

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

5.4.9.3 Description

The main steam piping is described in Section 10.3. The main steam and feedwater piping are shown in Dwgs. M-141, Sh. 1 and M-141, Sh. 2.

The feedwater piping consists of two 24-in. outside diameter lines each of which penetrate the containment and drywell and branch into three 12 inch lines which connect to the reactor vessel. Feedwater containment isolation valves are described in Table 6.2-12. The design pressure and temperature of the feedwater piping between the reactor and maintenance valve F011 are 1250 psig and 575°F. The Seismic Category I design requirements are placed on the feedwater piping from the reactor through the inboard isolation valve up to and including the outboard isolation valve in the connected piping.

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

The general requirements of the feedwater system are described in Subsections 7.1.2b.1, 7.7.1, 7.7.2 and 10.4.7.

5.4.9.4 Safety Evaluation

Differential pressure on reactor internals under the assumed accident condition of a ruptured steamline is limited by the use of flow restrictors and by the use of four main steamlines. All main steam and feedwater piping is designed in accordance with the requirements defined in Section 3.2. Design of the piping in accordance with these requirements ensures meeting the safety design bases.

5.4.9.5 Inspection and Testing

Inspection and testing are carried out in accordance with Subsection 3.9.1 and Chapter 14. In-service inspection is considered in the design of the main steam and feedwater piping. This consideration assures adequate working space and access for the inspection of selected components.

5.4.10 PRESSURIZER

Not applicable to BWR.

5.4.11 PRESSURIZER RELIEF DISCHARGE SYSTEM

Not applicable to BWR.

5.4.12 VALVES

5.4.12.1 Safety Design Bases

Valves are designed to operate under the internal pressure/temperature loading as well as the external loading experienced during the various system transient operating conditions.

The design loading combinations, design criteria and stress limits associated with ASME Section III valves are presented in Tables 3.9-6, 3.9-12 and 3.9-25.

The functional testing of active valves is covered in Subsection 3.9.3.2.b.2.

The materials for valves are covered in Subsections 5.2.3 and 6.1.1.

Inservice inspection requirements for ASME Section III valves are covered in Subsection 3.9.6.

5.4.12.2 Description

Line valves furnished are manufactured standard types, designed and constructed in accordance with the requirements of ASME Section III. All materials, exclusive of seals and packing, will endure the 40-year plant life under the environmental conditions applicable to the particular system (see Section 3.11). For service beyond 40 years, aging of equipment is managed to ensure it continues to perform its intended function. Power operators are sized to operate successfully under design-basis conditions. Furthermore, active safety related MOVs are sized to Generic Letter 89-10 requirements committed to the NRC by SSES in PLA-3311.

Motor operated valves are essentially the same type of equipment as used on currently operating plants.

5.4.12.3 Safety Evaluation

Line valves are shop tested by the manufacturer for operability. Pressure retaining parts are subject to the testing and examination requirements of Section III of the ASME Code. To minimize internal and external leakage past seating surfaces, maximum allowable leakage rates are stated in the design specifications for both back seat as well as the main seat for gate and globe valves. Valve construction materials are compatible with the maximum anticipated radiation dosage, which is listed in Section 3.11, for the service life of the valves.

Implementation of operational requirements specified in equipment specifications is ensured by document verification procedures.

The Quality Assurance programs instituted to ensure that valves meet specifications are the same as those described in Subsection 5.2.2.2.5.

Programs to establish qualification of the motor operated valve for the intended service are performed in the following areas:

Stroking time

Opening/closing at maximum differential pressure

Seismic performance

Emergency environmental condition.

In addition, active safety related MOVs meet the GL 89-10 program requirements committed to in PLA-3311.

5.4.12.4 Inspection and Testing

Valves serving as containment isolation valves are operationally tested as discussed in Subsection 6.2.4. Preoperational testing of valves is discussed in Chapter 14. In service testing of valves is discussed in Subsection 3.9.6.

Control valves 4 inches and larger and block valves 2-1/2 inches and larger were originally provided with double-packed stuffing boxes with an intermediate lantern leakoff connection to enable detection and measurement of leakage rates for valves located outside of the primary containment. Valves in the turbine building were originally provided with valve stem packing leakoff connections. Research and testing has shown that improved packing provides an effective seal to prevent leakage into the Turbine Building. As a result, these leakoff connections are in the process of being removed and packing configurations changed, as appropriate, to conform with the new requirements. As part of this effort, leakoff isolation valves and piping will be removed (or abandoned in place) and the leakoff collection header piping will be removed or abandoned in place.

Prior to installation, each motor actuator was assembled, factory tested and adjusted on the valve for proper operation, position, and torque switch setting, position transmitter function (where applicable) and speed requirements. Installed active safety related MOVs meet the GL 89-10 recommendations through current in-house programs as committed to by SSES Letter PLA-3311. These programs for the testing, inspection, and maintenance of MOVs assure that they will function when subjected to the design-basis conditions that are considered during both normal operation and abnormal events within the design basis of the plant.

5.4.13 SAFETY AND RELIEF VALVES

5.4.13.1 Safety Design Bases

Overpressure protection has been provided at isolatable portions of systems in accordance with the rules set forth in the ASME Code, Section III for Class 1, 2 and 3 components.

5.4.13.2 Description

Pressure relief valves have been designed and constructed in accordance with the same code class as that of the line valves in the system.

Table 3.9-1 lists the applicable code classes for valves and system design pressures and temperatures. The design criteria, design loading and design procedure are described in Subsection 3.9.3.

5.4.13.3 Safety Evaluation

The use of pressure relieving devices will assure that overpressure will not exceed 10% above the design pressure of the system. The number of relieving devices on a system or portion of a system have been determined on an individual component basis.

5.4.13.4 Inspection and Testing

Valves are stamped with factory ring (NR, GR) settings and applicable service back pressure range. Other design information is included on the nameplates attached on the valves. Further examinations would necessitate removal of the component. Refer to Subsection 5.2.4 for discussion of Inservice Inspection.

5.4.14 COMPONENT SUPPORTS

Support elements are provided for those components included in the RCPB and the connected systems.

5.4.14.1 Safety Design Bases

Design loading combinations, design procedures, and acceptability criteria are described in Subsection 3.9.3. Flexibility calculations and seismic analysis for piping components conform with the appropriate requirements of ASME Section III.

Spacing and size of pipe support elements were based on the piping analysis performed in accordance with ASME Section III and further described in Section 3.7.

Materials, fabrication, and inspection of pipe supporting elements for nuclear piping are in accordance with the ASME Boiler and Pressure Vessel Code, Section III. See Subsection 3.2 for applicable Code Edition.

5.4.14.2 Description

The use and location of rigid-type supports, variable or constant spring-type supports, and anchors or guides were determined by flexibility, stress, and seismic analysis.

5.4.14.3 Safety Evaluation

Design loadings used for flexibility and seismic analysis toward the determination of adequate component support systems included all transient loading conditions expected by each component. Provisions were made to provide travel stops for spring-type supports for the initial deadweight loading due to hydrostatic testing of steam systems to prevent damage to this type of support.

5.4.14.4 Inspection and Testing

After completion of the installation of a support system, all hanger elements are examined in accordance with the requirements of Chapter 14. Final adjustment capability is provided on all hanger or support types.

5.4.15 HIGH PRESSURE COOLANT INJECTION (HPCI) SYSTEM

Refer to Section 6.3 for discussion.

5.4.16 CORE SPRAY (CS) SYSTEM

Refer to Section 6.3 for discussion.

5.4.17 STANDBY LIQUID CONTROL (SLC) SYSTEM

Refer to Subsection 9.3.5 for discussion.

5.4.18 REFERENCES

5.4-1 Ianni, P.W., "Effectiveness of Core Standby Cooling Systems for General Electric Boiling Water Reactors," APED-5458, March, 1968.

5.4-2 "Design and Performance of General Electric Boiling Water Reactor Main Steam Line Isolation Valves," APED-5750, General Electric Co., Atomic Power Equipment Department, March, 1969.

5.4-3 Reference Deleted

5.4-4 NEDO-32686, “Utility Resolution Guide for ECCS Suction Strainer Blockage”, GE Nuclear Energy, October, 1998.

TABLE 5.4-2

REACTOR WATER CLEANUP SYSTEM EQUIPMENT DESIGN DATA		
System Flow Rate (lbs/hr)	146,300	
MAIN CLEANUP RECIRCULATION PUMPS		
Number Required –	2	
Capacity - % (each)	100%	
Design Temperature – (°F)	575	
Design Pressure – (psig)	1,450	
Discharge Heat at Shutoff – (ft)	705	
Minimum Available NPSH – (ft)	15.5	
HEAT EXCHANGERS		
	Regenerative	Non-Regenerative
Rated Capacity – (%)	100	100
Shell Side Pressure – (psig)	1,425	150
Shell Side temperature – (°F)	575	370
Tube Side Pressure (psig)	1,425	1,425
Tube Side Temperature - (°F)	575	575
FILTER-DEMINERALIZERS		
Number Required –	2	
Capacity -% (each)	50	
Flow/Unit (lb/hr)	73,150	
Design temperature – (°F)	150	
Design Pressure – (psig)	1,400	

TABLE 5.4-3

RHR RELIEF VALVE DATA

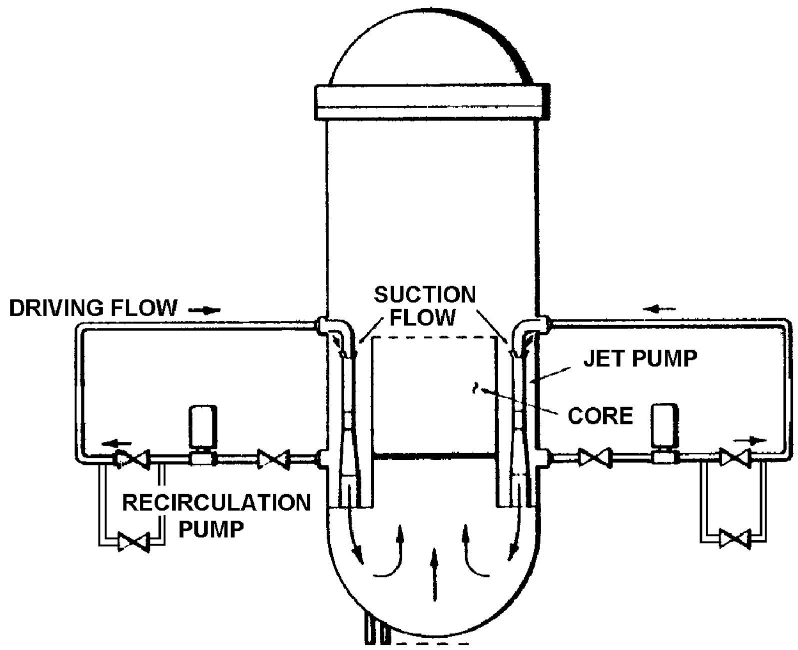
Valve Location	Valve No.	Set Point PSIG	Capacity [1] (gpm)	Method of Collection
Pump Suction Line	PSV-1F030 A, B, C & D	165	72	Liquid Radwaste
Pump Discharge Line	PSV-1F025 A&B	450	20.8	Liquid Radwaste
Head Spray Line	PSV-15113	450	38.1	Liquid Radwaste
Shutdown Supply Line (inside containment)	PSV-1F126	1250	34.6	Drywell Sump
Shutdown Supply Line (outside containment)	PSV-1F029	150	12	Liquid Radwaste
Heat Exchanger (shell side)	PSV-15106 A&B	450	20	Suppression Pool
Heat Exchanger (tube side)	PSV-11213 A&B	180	20	Liquid Radwaste
Heat Exchanger S.W. Line	PSV-11212 A&B	200	1620 gpm	Liquid Radwaste
Loop A B Cross Connect Line	PSV-15193	450	20.8	Liquid Radwaste

[1] Capacity is based on setpoint plus 10% accumulation

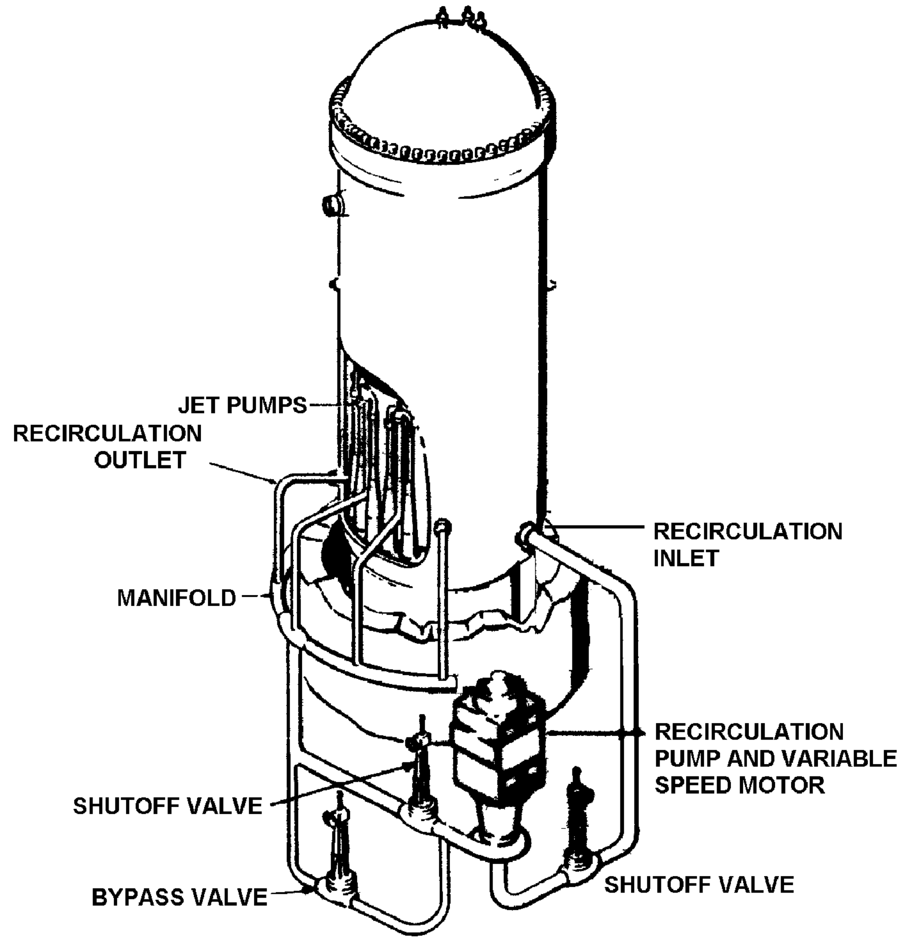
Table 5.4-4

RCPB COMPONENT DESCRIPTION

Security-Related Information
Table Withheld Under 10 CFR 2.390



ELEVATION



ISOMETRIC

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SUSQUEHANNA STEAM ELECTRIC STATION
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RECIRCULATION SYSTEM
ELEVATION AND ISOMETRIC

FIGURE 5.4-1, Rev.55

AutoCAD: Figure Fsar 5_4_1.dwg

THIS FIGURE HAS BEEN
REPLACED BY DWG.
M-143, Sh. 1

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SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
Figure 5.4-2B-1 replaced by dwg. M-143, Sh. 1 FIGURE 5.4-2B-1, Rev. 57

AutoCAD Figure 5_4_2B_1.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
M-143, Sh. 2

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SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
Figure 5.4-2B-2 replaced by dwg. M-143, Sh. 2 FIGURE 5.4-2B-2, Rev. 55

AutoCAD Figure 5_4_2B_2.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
M1-B31-13, Sh. 3

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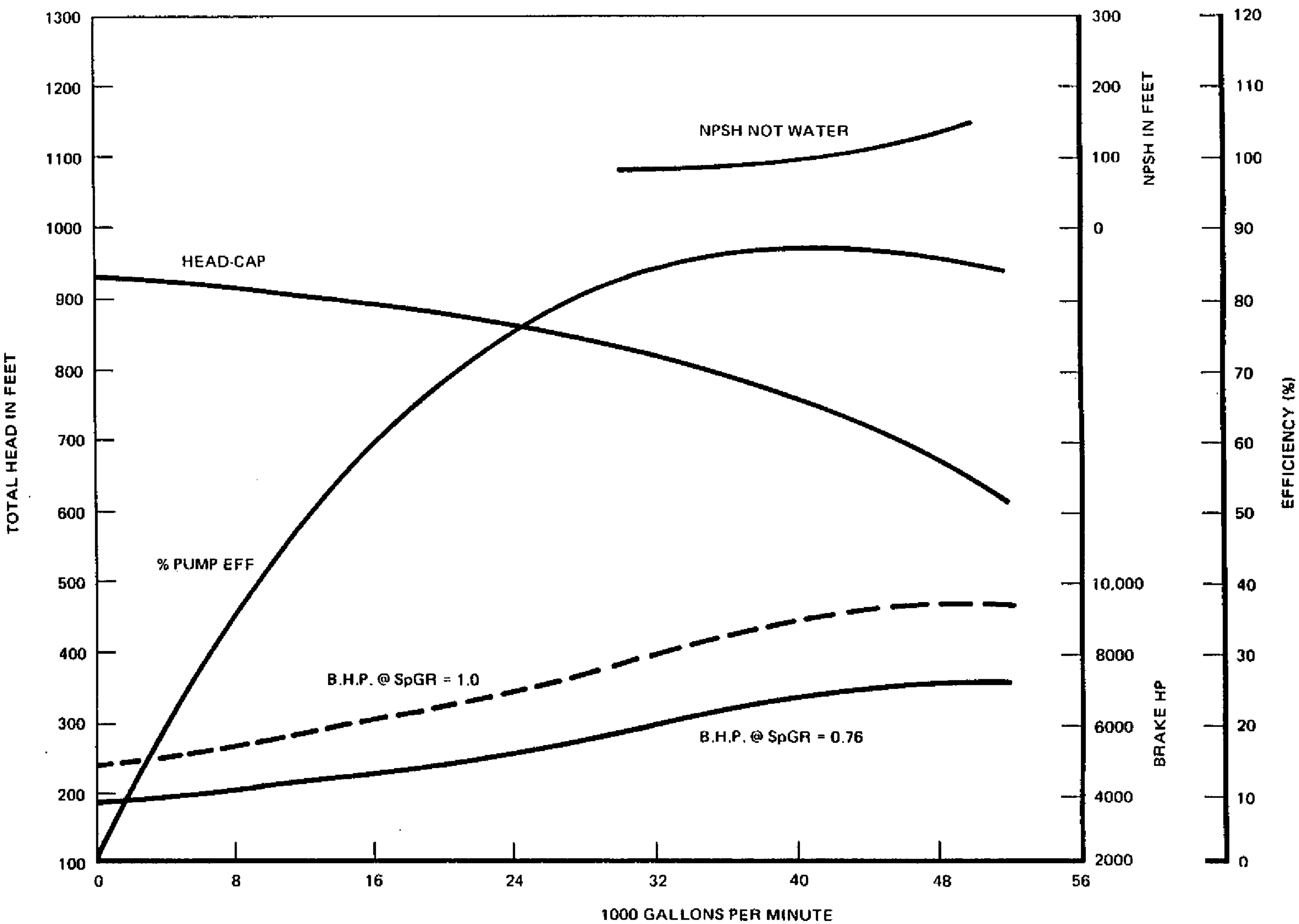
Figure 5.4-2C replaced by dwg.
M1-B31-13, Sh. 3

FIGURE 5.4-2C, Rev. 50

AutoCAD Figure 5_4_2C.doc



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SUSQUEHANNA STEAM ELECTRIC STATION
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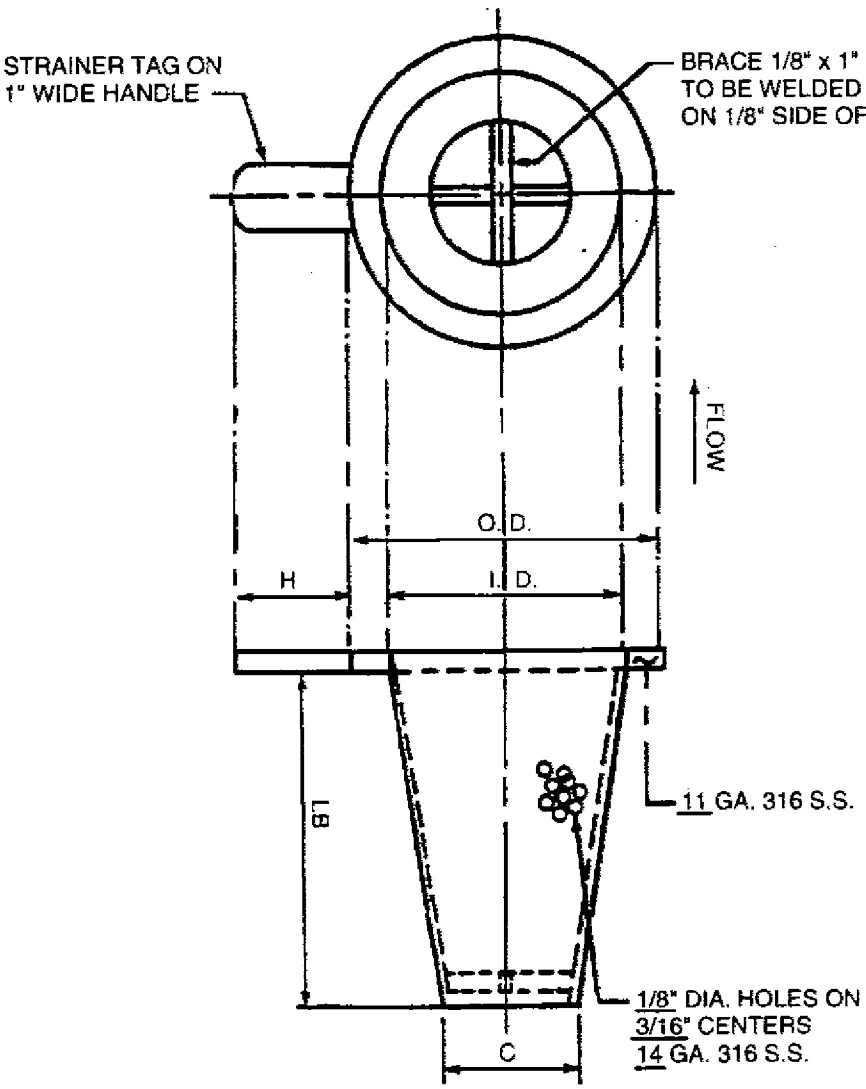
FINAL SAFETY ANALYSIS REPORT

RECIRCULATION PUMP HEAD,
NPSH, FLOW AND
EFFICIENCY CURVES

FIGURE 5.4-3, Rev.49

AutoCAD: Figure Fsar 5_4_3.dwg



NOTES:

1) **MATERIAL OF CONSTRUCTION ALL STAINLESS STEEL. SA-240-316**
2) **HANDLE MATERIAL TO BE 11 GA. SA-240-316**
3) **BOTTOM TO BE PERFORATED**

TYPE 5IP BASKET STRAINER							
QUAN	SIZE	RATING	ID	OD	LB	H	C
16	24"	150#RF	22%	28 1/8	48 1/2	4	16

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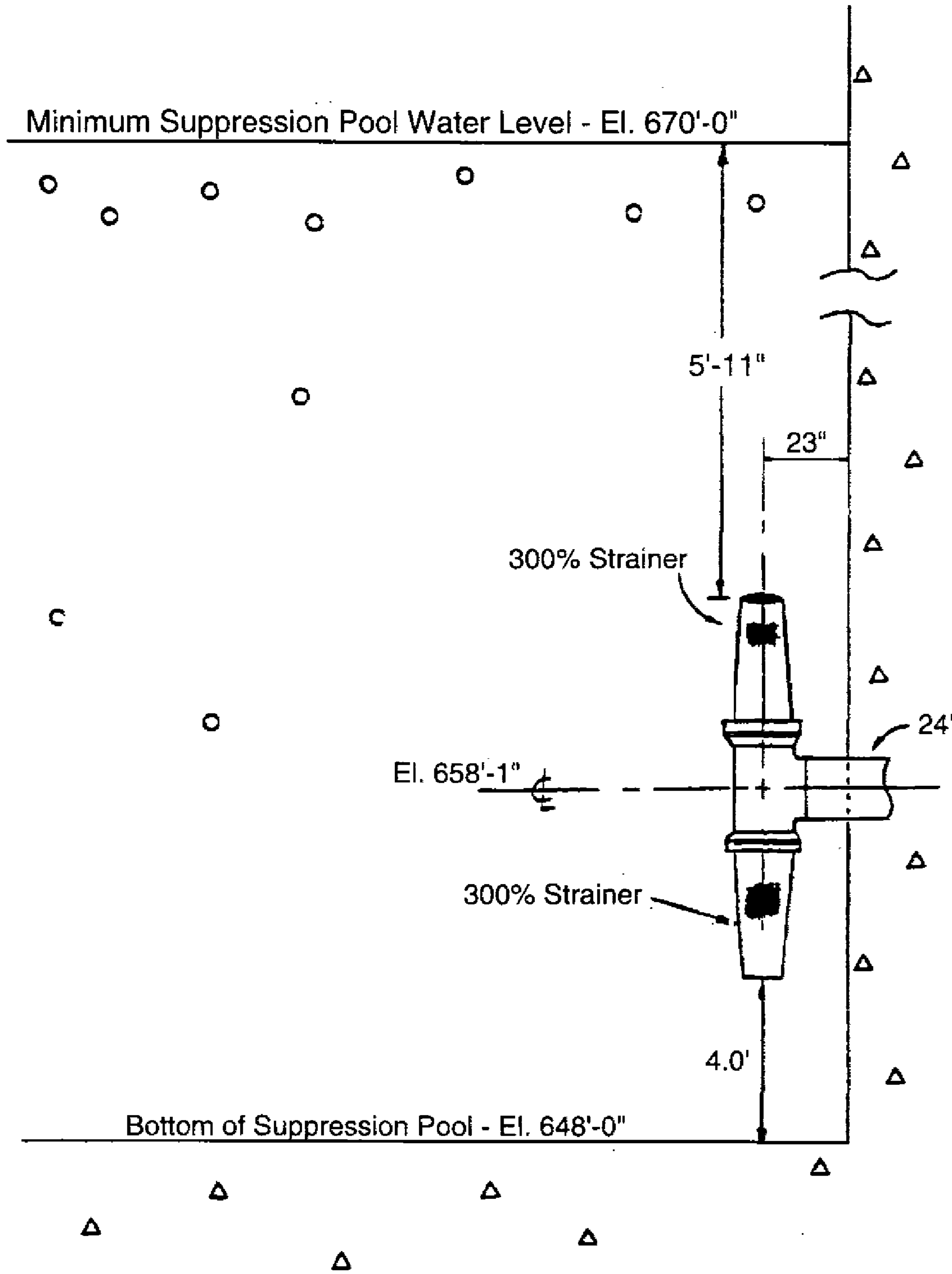
RHR SUCTION STRAINER DETAILS

FIGURE 5.4-4A, Rev.50

AutoCAD: Figure Fsar 5_4_4A.dwg



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SUSQUEHANNA STEAM ELECTRIC STATION
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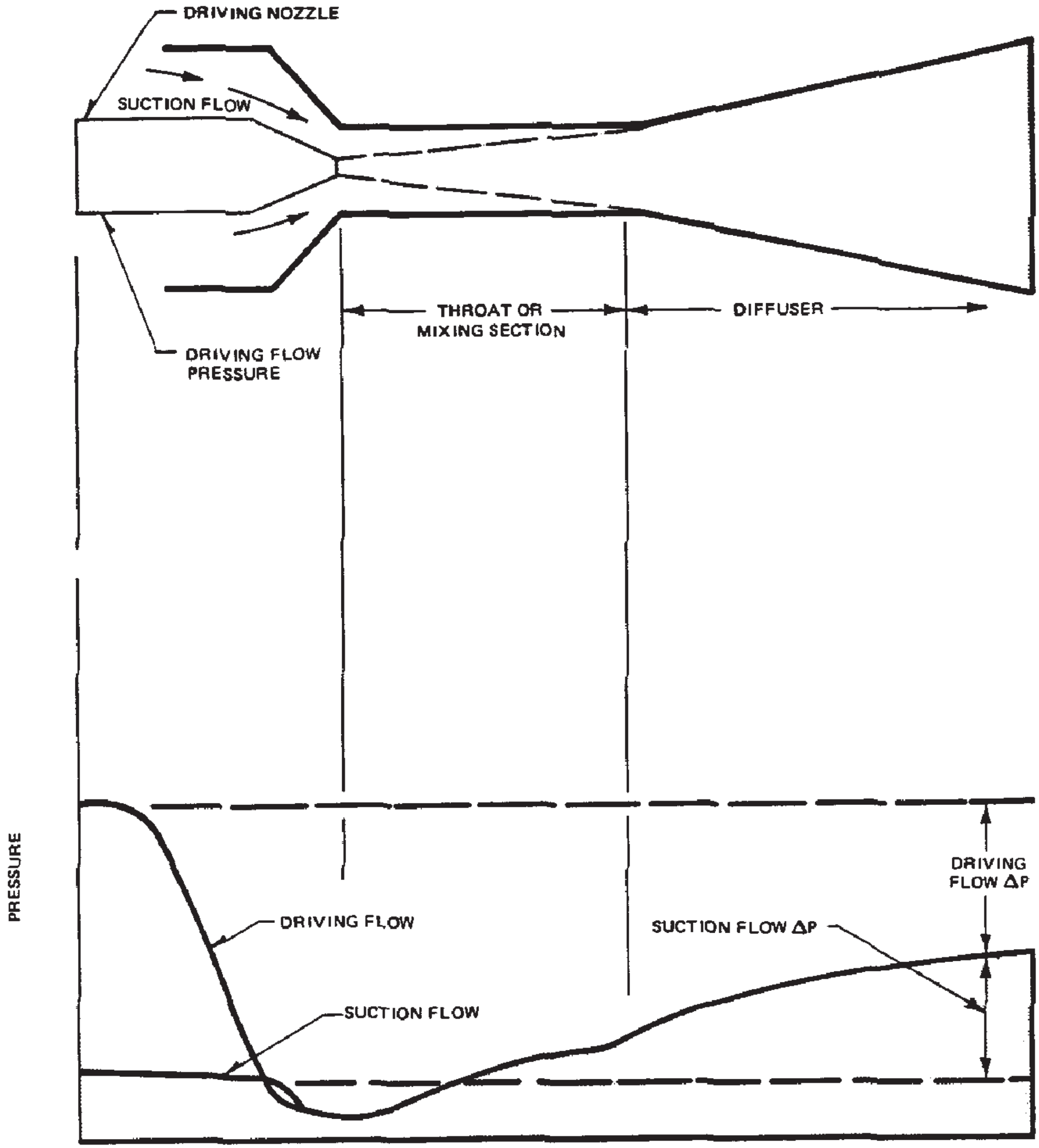
FINAL SAFETY ANALYSIS REPORT

RHR SUCTION STRAINER DETAILS

FIGURE 5.4-4B, Rev.50

AutoCAD: Figure Fsar 5_4_4B.dwg



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SUSQUEHANNA STEAM ELECTRIC STATION
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FINAL SAFETY ANALYSIS REPORT

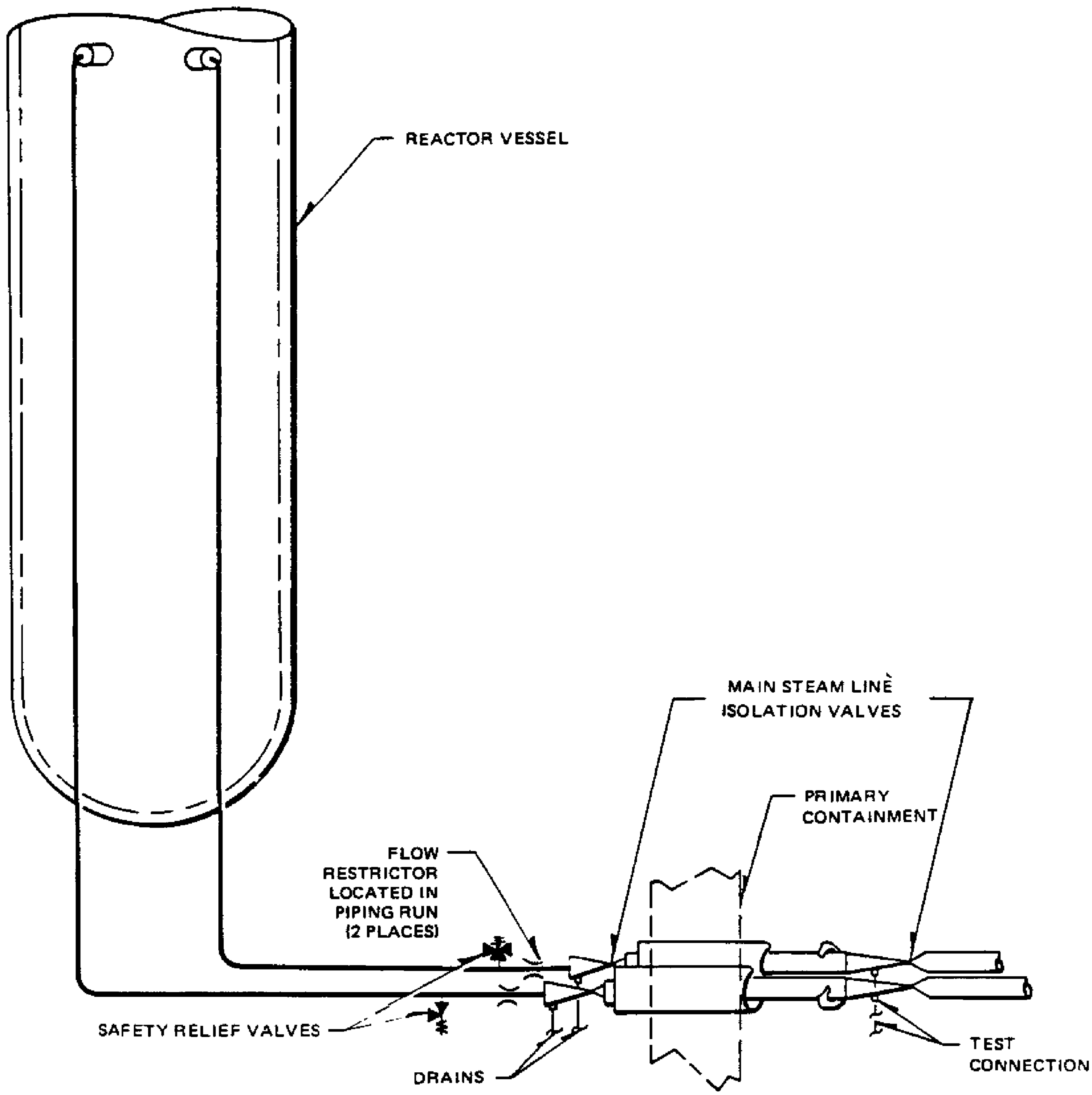
OPERATING PRINCIPLE
OF JET PUMP

FIGURE 5.4-5, Rev.49

AutoCAD: Figure Fsar 5_4_5.dwg

Security-Related Information
Figure Withheld Under 10 CFR 2.390

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
CORE FLOODING CAPABILITY OF RECIRCULATION SYSTEM FIGURE 5.4-6



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SUSQUEHANNA STEAM ELECTRIC STATION
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FINAL SAFETY ANALYSIS REPORT

MAIN STEAM LINE FLOW
RESTRICTOR LOCATION

FIGURE 5.4-7, Rev.49

AutoCAD: Figure Fsar 5_4_7.dwg

THIS FIGURE HAS BEEN
REPLACED BY DWG.
M-149, Sh. 1

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SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
Figure 5.4-9A replaced by dwg. M-149, Sh. 1 FIGURE 5.4-9A, Rev. 55

AutoCAD Figure 5_4_9A.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
M-150, Sh. 1

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Figure 5.4-9B replaced by dwg.
M-150, Sh. 1

FIGURE 5.4-9B, Rev. 55

AutoCAD Figure 5_4_9B.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
M1-E51-81, Sh. 1

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SUSQUEHANNA STEAM ELECTRIC STATION
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Figure 5.4-10 replaced by dwg.
M1-E51-81, Sh. 1

FIGURE 5.4-10, Rev. 52

AutoCAD Figure 5_4_10.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
M-151, Sh. 1

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SUSQUEHANNA STEAM ELECTRIC STATION
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Figure 5.4-13-1 replaced by dwg.
M-151, Sh. 1

FIGURE 5.4-13-1, Rev. 53

AutoCAD Figure 5_4_13_1.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
M-151, Sh. 2

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SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
Figure 5.4-13-2 replaced by dwg. M-151, Sh. 2 FIGURE 5.4-13-2, Rev. 54

AutoCAD Figure 5_4_13_2.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
M-151, Sh. 3

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SUSQUEHANNA STEAM ELECTRIC STATION
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Figure 5.4-13-3 replaced by dwg.
M-151, Sh. 3

FIGURE 5.4-13-3, Rev. 53

AutoCAD Figure 5_4_13_3.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
M-151, Sh. 4

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SUSQUEHANNA STEAM ELECTRIC STATION
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Figure 5.4-13-4 replaced by dwg.
M-151, Sh. 4

FIGURE 5.4-13-4, Rev. 54

AutoCAD Figure 5_4_13_4.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
M1-E11-3, Sh. 1

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SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
Figure 5.4-14-1 replaced by dwg. M1-E11-3, Sh. 1 FIGURE 5.4-14-1, Rev. 52

AutoCAD Figure 5_4_14_1.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
M1-E11-3, Sh. 2

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SUSQUEHANNA STEAM ELECTRIC STATION
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Figure 5.4-14-2 replaced by dwg.
M1-E11-3, Sh. 2

FIGURE 5.4-14-2, Rev. 52

AutoCAD Figure 5_4_14_2.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
M1-G33-16, Sh. 1

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Figure 5.4-15A replaced by dwg.
M1-G33-16, Sh. 1

FIGURE 5.4-15A, Rev. 52

AutoCAD Figure 5_4_15A.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
M1-G33-18, Sh. 1

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Figure 5.4-15B replaced by dwg.
M1-G33-18, Sh. 1

FIGURE 5.4-15B, Rev. 52

AutoCAD Figure 5_4_15B.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
M-144, Sh. 1

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SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
Figure 5.4-16-1 replaced by dwg. M-144, Sh. 1 FIGURE 5.4-16-1, Rev. 57

AutoCAD Figure 5_4_16_1.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
M-144, Sh. 2

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SUSQUEHANNA STEAM ELECTRIC STATION
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Figure 5.4-16-2 replaced by dwg.
M-144, Sh. 2

FIGURE 5.4-16-2, Rev. 55

AutoCAD Figure 5_4_16_2.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
M-144, Sh. 3

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SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
Figure 5.4-16-3 replaced by dwg. M-144, Sh. 3 FIGURE 5.4-16-3, Rev. 56

AutoCAD Figure 5_4_16_3.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
M-145, Sh. 1

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SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
Figure 5.4-18 replaced by dwg. M-145, Sh. 1 FIGURE 5.4-18, Rev. 55

AutoCAD Figure 5_4_18.doc