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*The listed drawings are included as "General References" only; i.e., refer to the drawings to obtain additional detail or to obtain background information. These drawings are not part of the UFSAR. They are controlled by the Controlled Documents Program.

5.0 REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

5.1 SUMMARY DESCRIPTION

The equipment and evaluations presented in this chapter are applicable to either unit. [5.1-1]

The reactor coolant system includes those systems and components which contain or transport reactor coolant, in the form of water or steam, to and from the reactor pressure vessel (RPV). These systems form a major portion of the reactor coolant pressure boundary (RCPB). Chapter 5 of this report provides information regarding the reactor coolant system and pressure-containing appendages out to and including the outermost isolation valve in the main steam and feedwater piping. [5.1-2]

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

A. Part of the reactor coolant system (RCS), or

B. Connected to the RCS up to and including any and all of the following:

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

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

    3. The RCS safety relief valve (SRV), relief valves (RVs), and safety valves.

The topics of the RCS and connected systems that are discussed in this chapter include RCPB integrity, the RPV and appurtenances, and the major RCPB allied subsystems. Diagram of Nuclear Boiler and Reactor Recirculation Piping is shown in FSAR Figure 5.1-1. Table 5.1-1 includes information such as overall dimensions, design pressures and temperatures, power ratings, and design codes for the major components of the RCS. The total water and steam volume of the reactor vessel and recirculation system is approximately 15,679 cubic feet at 68°F. Additional parameters of the RCS are summarized in Tables 5.4-1 through 5.4-6. [5.1-3]

5.1.1 Reactor Coolant Pressure Boundary Integrity

Section 5.2 addresses the integrity of the RCPB. This section includes discussions of overpressurization protection, RCPB materials, inservice inspection and testing of the RCPB, and RCPB leakage detection.

To protect against overpressure, relief valves are provided that can discharge steam from the RCS to the suppression pool. The automatic depressurization system (ADS) also acts to automatically depressurize the RCS in the event of a (small break) loss-of-coolant

accident (LOCA) in which the high pressure coolant injection (HPCI) system fails to maintain sufficient reactor vessel water level. Section 6.3 provides more details regarding the ADS and HPCI systems. Depressurization of the RCS allows the low-pressure core cooling systems to function.

5.1.2 Reactor Pressure Vessel and Appurtenances

The RPV and appurtenances are described in Section 5.3. The major safety consideration for the RPV is the ability to function as a radioactive material barrier. Various combinations of loading were considered in the vessel design. The vessel meets the requirements of applicable codes and criteria. The possibility of brittle fracture was considered, and suitable design and operational limits have been established to avoid conditions where brittle fractures are possible. Figures 5.3-4a through 4c of Section 5.3 provide more information regarding the pressure-temperature limits. Refer to Reference 1 for detailed design information on the RPV, the purchase specifications for the RPV, RPV manufacturers data, and the seismic analysis of the RPV.

5.1.3 Reactor Coolant System Subsystems

Section 5.4 deals with subsystems that are closely allied to the RCPB. These include the reactor recirculation system, the hydrogen water chemistry (HWC) system, the main steam line flow restrictors, the reactor core isolation cooling (RCIC) system, the residual heat removal (RHR) system, and the reactor water cleanup (RWCU) system. A brief description of these subsystems is provided in the following paragraphs.

The reactor recirculation system (refer to Section 5.4.1) provides coolant flow through the core. Adjustment of the core coolant flow rate changes reactor power output, thus providing a means of following plant load demand without adjusting control rods. The recirculation system is designed so that no fuel damage will result during operational transients caused by reasonably expected single operator errors or equipment malfunctions. The arrangement of the recirculation system routing plus the appropriate placement of pipe break restraints is such that a piping failure cannot compromise containment integrity.

The HWC system (refer to Section 5.4.3) is used to inject hydrogen into the reactor coolant to limit the dissolved oxygen concentration. Suppression of dissolved oxygen, coupled with high purity reactor coolant, reduces the susceptibility of reactor piping and materials to intergranular stress corrosion cracking.

The main steam line flow restrictors (refer to Section 5.4.4) are venturi-type flow devices that are welded into each steam line between the RPV and the first main steam line isolation valve (MSIV). The restrictors are designed to limit the loss of coolant resulting from a main steam line break outside the primary containment so that reactor vessel water level remains above the top of the core during the time required for the MSIVs to close.

Two isolation valves are installed on each main steam line, one inside and the other outside the primary containment. The MSIVs are discussed in Section 6.2.

The RCIC system (refer to Section 5.4.6) provides makeup water to the core during a reactor shutdown when the reactor becomes isolated from the main condenser and feedwater flow is not available. The system is started automatically upon receipt of a low reactor water signal, or is manually started by the operator. Water is pumped to the core from the contaminated condensate storage tank by a turbine-driven pump using reactor steam. For 10 CFR 50 Appendix R considerations, the electric-driven safe shutdown makeup pump (SSMP) provides backup to the RCIC system of either Unit 1 or Unit 2. For details on the SSMP system refer to Section 5.4.6.5.

The major equipment of the RHR system (refer to Section 5.4.7) includes four main system pumps, two heat exchangers, and four RHR service water pumps. The RHR system can be used to remove heat under a variety of situations. In one of its three modes of operation (shutdown cooling), the RHR system removes decay heat during normal shutdown and reactor servicing. A second mode of RHR system operation (containment cooling) removes heat from the primary containment following a loss-of-coolant accident. The third operational mode of the RHR system is low-pressure coolant injection (LPCI). Low pressure coolant injection is used during a postulated loss-of-coolant accident. LPCI operation is described in Section 6.3.2. Other features of the RHR system include: supplementing the fuel pool cooling system; draining the condenser to the suppression chamber by taking water from a condensate pump; transferring water from the RPV to the main condenser or to the suction of the condensate pumps; transferring water from the suppression chamber to the radwaste system, or via the radwaste system to the main condenser; and delivering and returning reactor water to the fuel pool system demineralizer for cleanup.

The RWCU system (refer to Section 5.4.8) recirculates a portion of the reactor coolant through a filter-demineralizer to remove soluble and insoluble impurities. The RWCU system maintains RCS coolant inventory by returning the same quantity of water that was extracted.

Section 5.4 also discusses: main steam line and feedwater piping (refer to Section 5.4.9), valves (refer to Section 5.4.12), and the safety and relief valves (refer to Section 5.4.13).

5.1.4 Piping and Instrumentation Diagrams

The piping and instrumentation diagrams (P&IDs) applicable to the RCS and connected systems are identified in Table 5.1-2. This table is organized according to the drawing topic first and then the applicable unit. [5.1-4]

5.1.5 General Arrangement

The general arrangement drawings for the reactor coolant system and connected systems are shown on M-6, M-8, and M-9. These drawings are applicable to both Unit 1 and Unit 2.

5.1.6 References

1. Quad Cities Reactor Pressure Vessel Design Report.

Table 5.1-1

REACTOR COOLANT SYSTEM DATA

Reactor Vessel

Internal height	68 ft 7-5/8 in
Internal diameter	251 in
Design pressure and temperature	1250 psig at 575°F
Maximum heatup rate and normal cooldown rate	100°F within a 1-hour period
Base metal material	SA-302 Grade B
Top Head thickness	4 in. (minimum)
Shell thickness	6-1/8 in (minimum)
Bottom Head thickness	6-1/8 in. (minimum)
Design lifetime	40 years
Base metal initial NDT (assumed)	40°F maximum
Cladding material	Weld deposited ER-308 electrode
Cladding thickness	1/8 in (minimum)
Design code	ASME[*] Section III Class A, 1965

Recirculation Loops

Number	2
Material	Stainless steel
Design Pressure and temperature	
Suction	1175 psig at 565°F
Discharge	1325 psig at 580°F
Design Code	ASME Section I, 1965 ASME Section I, 1968 USAS B 31.1, 1967

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* ASME Boiler and Pressure Vessel Code.

Table 5.1-1 (Continued)

REACTOR COOLANT SYSTEM DATA

Recirculation Pumps	
Number	2
Type	Vertical, centrifugal, single stage
Power rating	6000 hp
Speed	1800 rpm
Flow rate	45,000 gal/min
Design pressure and temp.	1450 psig at 575°F
Developed head	570 ft
Design code	ASME Section III Class C, 1965 ASME Section III Class C, 1968
Recirculation Valves	
Number	8
Type	Motor operated gate
Design code	ASME Section I, 1965 ASME Section I, 1968 USAS B 31.1, 1967
Jet Pumps	
Number	20
Material	Stainless steel
Overall height (top of nozzle to diffuser discharge)	18 ft 7 in
Diffuser diameter	20-3/4 in

Table 5.1-1 (Continued)

REACTOR COOLANT SYSTEM DATA

<u>Main Steam Lines</u>	
Number	4
Diameter	20 in
Material	Carbon Steel
Design Code	ASME Section I, 1965 ASME Sections I and III, 1968 USAS B 31.1, 1967
<u>Electromatic Relief Valves</u>	
Number	4
Capacity (each)	558,000 lbm/hr each at 1120 psig
Pressure setting (analytical limit)	≤1115 psig (2) ≤ 1135 psig (2)
Design Code	USAS B 31.1 1967 ASME Section III Code Class 1, 1980 Edition, Winter 1980 Addenda without Code Stamp ASME Section III Code Class 1, 1995 Edition, with 1996 Addenda without Code Stamp
<u>Safety Valves</u>	
Number	8
Capacity (each)	644,543 lbm/hr each at 1240 psig
Pressure Setting	1240 psig (2) 1250 psig (2) 1260 psig (4)
Design Code	ASME Section III, 1965 ASME Section III, 1968 USAS B 31.1, 1967

Table 5.1-1 (Continued)

REACTOR COOLANT SYSTEM DATA

Safety/Relief Valve (Target Rock)

Number	1
Capacity (each)	598,000 lbm/hr at 1080 psig
Pressure Setting (safety function setpoint)	1135 psig
Pressure Setting (analytical limit for relief function)	≤1135 psig
Design Code	ASME Section III, 1965 ASME Section III, 1968 USAS B 31.1, 1967

Reactor Core Isolation Cooling System

TURBINE

Steam Pressure Inlet	150 — 1120 psia
Exhaust	25 psia
Power	80 — 500 hp
Steam Flow	6,000 — 16,500 lb/hr

Pump

Number	1
Type	5 - stage, horizontal, centrifugal
Discharge Developed Head - over a reactor pressure range of 1135 psia - 165 psia	2800 ft at 1135 psia — 525 ft at 165 psia
Flow	400 gal/min
NPSH	20 ft

Control Power

	125 Vdc
	120 Vac

Table 5.1-2

APPLICABLE REACTOR COOLANT SYSTEM P&IDs

Topic	Unit	Drawing
Main steam piping	1	M-13
Main steam piping	2	M-60
Reactor feed piping	1	M-15
Reactor feed piping	2	M-62
Pressure suppression piping	1	M-34
Pressure suppression piping	2	M-76
Reactor recirculation piping	1	M-35
Reactor recirculation pump trip ATWS piping	1	M-35
Reactor recirculation piping	2	M-77
Reactor recirculation pump trip ATWS piping	2	M-77
SSMP diagram	1 & 2	M-70
RCIC piping	2	M-89
RCIC piping	1	M-50
RHR piping	1	M-37
RHR piping	1	M-39
RHR piping	2	M-79
RHR piping	2	M-81
RWCU piping	1	M-47
RWCU piping	2	M-88

5.2 INTEGRITY OF REACTOR COOLANT PRESSURE BOUNDARY

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

5.2.1 Compliance With Codes and Code Cases

The edition of applicable codes, addenda, and code cases for the pressure vessels, piping, valves and pumps of the RCPB components are listed in Section 3.2. [5.2-1]

5.2.2 Overpressurization Protection

The Quad Cities Extended Power Uprate Project included re-evaluating a broad set of most limiting transient events at the power uprated conditions. The Limiting Transient Overpressure Events which are reanalyzed at 2957 MWt included the MSIV closure with direct scram, the single MSIV closure, the load rejection with bypass, the slow recirculation increase, and the fast recirculation increase. In addition, a Turbine Trip without bypass with a high flux scram was performed to reconfirm that the MSIV closure with flux scram was the limiting event for the ASME overpressure analysis. Specific diagrams showing the results of their transient are contained in Reference 5.

Overpressurization of the RCPB during reactor operations other than refueling is prevented by the design of the reactor control systems and the reactor safety system. These design features include: [5.2-2]

A. High reactor pressure scram;

B. High neutron flux scram;

C. Turbine-generator load rejection scram;

D. Operation of the reactor core isolation cooling (RCIC) system;

E. Operation of the turbine bypass system;

F. Operation of the dual-function safety/relief valve (SRV); [5.2-3]

G. Operation of the relief valves;

H. Operation of the safety valves;

I. Operation of the high pressure coolant injection (HPCI) system ;and

J. Main steam isolation valve (MSIV) closure scram.

5.2.2.1 Design Bases

The purpose of the relief and safety valves is to prevent over-pressurizing of the RCPB including the reactor pressure vessel (RPV). The relief valves are also designed to rapidly depressurize the RPV in the event of a small break loss-of-coolant accident (LOCA) where HPCI malfunctions so that core spray

and the low pressure coolant injection (LPCI) mode of the residual heat removal (RHR) system will function to protect the fuel barrier. Position indication of the safety/relief valve and the other four relief valves is required to be obtainable during reactor operation. To achieve these purposes, the relief, safety/relief, and safety valves have the following capacities and setpoints: [5.2-4]

Relief Valves (4) [5.2-5]	
Capacity	558,000 lb_m/hr each at 1120 psig
Pressure Setting	<=1115 psig (2)
(analytical limit)	<=1135 psig (2)
Safety Valves (8)	
Capacity	644,543 lb_m/hr each at 1240 psig
Pressure Setting	1240 psig (2)
	1250 psig (2)
	1260 psig (4)
Safety/Relief Valve (1) (Target Rock) [5.2-6]	
Capacity	598,000 lb_m/hr at 1080 psig
Pressure Setting	1135 psig
(safety function setpoint)	
Pressure Setting	≤1135 psig
(analytical limit for relief function)	

The relief valves, which include the SRV, are sized to rapidly remove the generated steam flow upon closure of the turbine stop valves and coincident with failure of the turbine bypass system.

The safety valves are sized to protect the RPV against overpressure during a MSIV closure without direct scram on valve position event, a turbine trip with a failure of the turbine bypass system and without direct scram on turbine stop valve position event, or a load reject with a failure of the turbine bypass system and without direct scram on turbine control valve fast closure event (see Section 5.2.2.2.3 for further details). The ASME Code requires that each vessel designed to meet Section III be protected from the consequence of pressure and temperature in excess of design conditions. The USAS B 31.1 Code for Pressure Piping also requires overpressure protection. [5.2-7]

The relief and safety valve capacities are taken from the "Transient Protection Parameters Verification for Reload Licensing Analyses." [5.2-8]

The installation of the Acoustic Side Branch (ASB) in the inlet piping to the Electromatic Relief Valves and Safety Valves (see Section 3.9.2.1) has increased the pressure loss coefficient upon valve actuation. The increased loss coefficient results in a reduction in flow through these valves for a given steam line pressure. The impact of flow reduction due to this additional pressure loss is factored into the applicable events as part of each reload analysis. The ASME capacity of the valves is not changed.

5.2.2.2 Design Evaluation

5.2.2.2.1 Loadings and Analyses

Steam generated following reactor isolation must be removed rapidly enough to prevent a large pressure rise. The relief valves are provided to remove sufficient steam from the reactor, following a scram that includes MSIV closure, to prevent the safety valves from lifting. [5.2-9]

In compliance with ASME Section III, the safety valves must be set to open no higher than 105% of design pressure, and at least one safety valve pressure setting shall not be greater than the design pressure of the vessel. The setpoints of the safety valves comply with the ASME Code taking into account static heads and dynamic losses.

Studies have been made on plants that are geometrically similar to Quad Cities on the loadings which the relief and safety valves place on the main steam line. The loadings considered include: [5.2-10]

A. The thermal expansion effects of the main steam and relief valve discharge piping.

B. The earthquake effects of the relief and safety valves and relief valve discharge piping.

C. The jet force exerted on the relief and safety valves during the first millisecond when the valve is open and steady state flow has not yet been established.

D. The dynamic effects of the kinetic energy of the piston disc assembly when it impacts on the base casting of the valve.

The piping system and supports were qualified for the following loading conditions which the relief valves placed on the main steam piping. [5.2-11]

A. thermal and dead weight effects on the main steam and relief valve discharge piping

B. Operating Basis Earthquake (OBE) and Safe Shutdown Earthquake (SSE) effects on the main steam and relief valve discharge piping

C. dynamic effects of the kinetic energy and the jet forces when the relief valves open (SRV) and safety valves open (SV)

These six load cases (dead weight, thermal, OBE, SSE, SRV and SV), reference Section 3.9, are then used in various combinations (i.e., one or more relief valves opening, one or more safety valves opening) to provide maximum piping stress and support loading. This support loading includes the piping restraints inside the pressure suppression chamber (torus).

Thermal expansion analyses were made for several cases with the relief valve piping both cold and hot and with jet forces, and piston disc impact forces, applied simultaneously to all valves. These studies show that the loads due to relief valve operation have only a minor effect on the stress condition of the main steam piping. The greatest stress is found at the branch connection below the valve. In no case has the stress at this point exceeded the maximum stresses allowed by the ASME Code. [5.2-12]

An overview of the analysis performed for the Electromatic relief valves included the following: [5.2-13]

A. Determine the valve nozzle loads and the valve center of gravity (C.G.) accelerations from the piping analysis.

B. Using the loads of part (A), consider the following elements and find the stresses:

1. Calculate the valve body stress for internal steam pressure plus safe shutdown earthquake and relief valve operation $(SSE+RV)_{ABS}$ valve nozzle loads from part (A).

2. Using $(SSE + RV)_{ABS}$ "g" values, find the stresses in the turnbuckle and the pilot valve tube for these "g" values applied to the extended structure C.G. in three directions simultaneously.

   Assume a continuous solid circular cross-section of 1-1/32-inch outside diameter for the turnbuckle. This simplifies the analysis for the loads and moments supported by the turnbuckle and pilot valve tube.

   Include the internal steam pressure for the pilot valve tube.

3. The stresses of the solenoid assembly mounting bracket hold-down bolts are calculated for the $(SSE + RV)_{ABS}$ acceleration of the solenoid switch assembly C.G. in three directions simultaneously.

4. The solenoid assembly mounting bolts (located at the top of the mounting bracket) are analyzed for the $(SSE + RV)_{ABS}$ acceleration of the solenoid switch C.G. in three directions simultaneously.

5. Target Rock Safety Relief Valve Out-of-Service design basis information is addressed in the cycle–specific reload reports.

A summary of these stresses is shown in Table 5.2-1.

Pneumatic supply lines to the Target Rock SRV have been upgraded to seismic design as a result of IE Bulletin 80-01. This was done to ensure a pneumatic supply to the valve in the event of a design basis earthquake. [5.2-14]

5.2.2.2.2 Relief Valve Sizing

The relief valves are sized, based upon the original analysis, by assuming a turbine trip with simultaneous reactor scram and with a failure of the turbine bypass system. This transient is reanalyzed periodically as part of reload licensing analysis. Typical results for this transient from 2957 MWt operating conditions are presented in Reference 5 and are shown in Figure 5.2-1. The sudden closure of the turbine stop valves with no initial bypass flow effectively doubles the initial rate of increase of primary system pressure. Scram is initiated from the stop valve closure. [5.2-15]

The vessel pressure peaks at 1292 psig. Peak pressure in the steam line at the safety valve location is approximately 1253 psig. Core cooling, level, and pressure control are provided by the RCIC system. Analyses performed at an uprated power level of 2957 MWt [5] have shown that the safety valves may lift during this transient if conservative parameters are assumed; however, this postulated event would be a rare occurrence.

5.2.2.2.3 Safety Valve Steam Flow Capacity

For power uprate, the safety valves steam flow capacity is determined by assuming that the reactor is at 2957 Mwt when a MSIV closure occurs, the relief valves fail to open, direct reactor scram (based on MSIV position switches) fails, and the backup scram due to high neutron flux shuts down the reactor. This transient is reanalyzed periodically as part of each reload license analysis. Pressure increases, following this reactor isolation, until limited by the opening of the safety valves. The peak allowable pressure is 1375 psig (according to ASME Section III equal to 110% of the vessel design pressure 1250 psig). The safety valves setpoints are spread in 10 psi increments between 1240 and 1260 psig. This satisfies the ASME Code specifications that the lowest safety valve be set at or below vessel design pressure, and the highest safety valve be set to open at or below 105% of vessel design pressure. [5.2-17]

The total safety valve capacity is equal to approximately 43% of turbine design flow.

Typical resulting transients at 2957 MWt are shown in Reference 5 and in Figure 5.2-3. The rapid pressurization caused by the isolation (about 100 psi/s) reduces the void content of the core and produces a sharp neutron flux spike before scram shuts down the reactor. Peak fuel surface heat flux is significantly slower, reaching a peak of 129% at about 3 seconds. Vessel dome pressure reaches about 1336 psig with the peak at the bottom of the vessel near 1358 psig. Therefore, the 43% capacity safety valves provide adequate margin below the peak allowable vessel pressure of 1375 psig.

Overpressurization protection analysis is performed using the NRC approved transient code(s) each cycle. A description of the overpressurization protection methodology used for Westinghouse reloads can be found in Section 9.3.2 of Reference 6 for Westinghouse analyses and can be found in cycle-specific reload analyses for AREVA analyses. The MSIV closure without direct scram on valve position event, a turbine trip with a failure of the turbine bypass system and without direct scram on turbine stop valve position event, and a load reject with a failure of the turbine bypass system and without direct scram on turbine control valve fast closure event are evaluated each reload to ensure the ASME overpressure limit is not exceeded. Also, for the turbine bypass valves equipment out-of-service option, the Feedwater Controller Failure (FWCF) event and the Inadvertent High Pressure Coolant Injection (IHPCI) event are analyzed. To satisfy the ASME criterion for the maximum vessel peak pressure and the steam dome pressure safety limit for the FWCF and IHPCI events with turbine bypass valves equipment out-of-service option, a power restriction may apply depending on the availability of the safety valves. Results for the limiting event are presented in the reload licensing report. [5.2-18]

5.2.2.2.4 Relief Valve Discharge Line Restraint Analysis

Earlier experience at some BWR plants with Mark I containments revealed inadequacies in the relief valve piping restraints inside the pressure suppression chamber. In some cases, original restraints were replaced. The NRC had requested that the relief valve piping and restraints inside the pressure suppression chamber of Quad Cities Station, Units 1 and 2, be inspected for signs of damage and analyzed to confirm the adequacy of the original design. [5.2-19]

In the original design, each of the units had five main steam Electromatic relief valves and associated discharge lines which were constructed of 8-inch schedule 80 pipe material per ASTM A-106, Grade B. These five lines entered separate bays of the suppression chamber through the drywell-to-suppression chamber vent tubes, and terminated in a ramshead configuration at the suppression chamber centerline approximately two-thirds of the distance below the normal suppression chamber water level. [5.2-20]

To mitigate the pressure spike following certain postulated transients, which related to the projected rate of reactivity insertion following a scram, a modification was planned to replace the Electromatic relief valves with faster-acting Target Rock SRVs. The complete modification was never installed. Instead a partial version, known as the Scram Reactivity-Interim Fix modification, was installed on both units. This modification replaced only one of the original main steam Electromatic relief valves with a Target Rock SRV on each unit. An extensive analysis was performed to justify installation of the faster-acting Target Rock SRV without upgrading the discharge line restraints. [5.2-21]

The analysis, which was quite conservative, showed that the stresses in the restraint members resulting from all loads combined were below yield in all cases. Analysis of the many welded and bolted connections demonstrated that all of them had acceptable stresses. A special visual inspection of the highest stressed connection was made on Unit 2 on October 14, 1975. The inspection revealed no damage to the connection. [5.2-22]

The first part of the analysis consisted of finding the hydraulic forces on the piping that resulted when a relief valve was opened. The computer code SRVA used advanced calculational techniques to model vent clearing phenomena, and the results were inherently very conservative. The calculation revealed that the hydraulic transient force was the largest single force acting on the pipe; all other forces were much smaller.

The original restraint design did not consider the forces related to clearing water from the vent line. Fortunately, the restraints were designed with enough additional strength to prevent their failure under the revised load combination.

The maximum forces predicted to occur in various members of restraint structures are given in Table 5.2-2. The maximum calculated fiber stresses and shear stresses in various members are also shown. The stresses in the Target Rock valve discharge line restraints were predicted to be higher than those in the Electromatic valve discharge line restraints. [5.2-23]

Table 5.2-2 shows that the stresses in some members of the supporting structures were higher than the AISC allowable stresses. However, the fact that about 215 total blowdowns had occurred prior to the inspection without any apparent damage, and also that the major portion of the hydraulic transient load is of a significantly short duration (lasting only about 0.25 seconds, with peak load occurring only for a fraction of this duration), suggest that the hydraulic transient load used in the analysis was conservatively estimated, and a higher allowable stress in the members (due to a high strain rate) was being realized. [5.2-24]

Presently, each unit has four relief valve lines with Electromatic relief valves and one line with a Target Rock dual-function SRV which opens faster and allows marginally greater flow than the Electromatic valves. An extensive history of Electromatic relief valve discharges exists for the plant. Subsequent inspections have revealed only isolated cases of minor damage to pipe supports, and these were attributed to installation deficiencies. The discharge lines equipped with the faster-acting Target Rock SRVs have

shorter discharge histories; however, inspection of one line after two discharges revealed no damage to pipe supports. [5.2-25]

The adequacy of the Electromatic relief valves (including the Target Rock SRV) and their associated discharge lines and restraints for liquid and two-phase flow was demonstrated as part of a post TMI test activity conducted for the BWR Owners Group. The test results were documented in NEDE-24988-P[1] which was provided to the NRC in September, 1981. [5.2-26]

Subsequent to the analysis described above, additional loadings on the discharge lines were defined in the Mark I Containment Program. T-quencher devices were installed on the exits of the five relief valve discharge lines to reduce hydrodynamic loads in the suppression pool and promote stable steam condensation. Discharge line supports were redesigned. In addition, a new larger vacuum breaker valve was installed on each discharge line in the drywell approximately 20 feet upstream of the jet deflector plate which protects the vent tube. Finally, relief valve setpoints were adjusted and a lock-out timer was installed on the low-set valve on each unit to prevent repeated actuation concurrent with an elevated water level in the discharge line. The larger vacuum breaker, set-point adjustments, and lock-out timer were designed to reduce transient loads on the discharge line and its supports, and hydrodynamic loads on the suppression chamber and attached structures. [5.2-27]

The discharge lines and the associated supports were reanalyzed for the new hydrodynamic loads identified in the Mark I Program (see Section 6.2.1.3.4), and were shown to comply with the applicable acceptance criteria. The analysis applicable to the current discharge line configuration is described in the Plant Unique Analysis Report (PUAR).

5.2.2.3 Piping and Instrumentation Diagrams

The piping and instrumentation diagrams (P&IDs) for the nuclear boiler system including the overpressure protection devices and their routing are shown on P& ID M-13, M-35 for Unit 1; and M-60, M-77 for Unit 2.

5.2.2.4 Equipment and Component Description

The Electromatic relief valves are actuated automatically by a high reactor vessel pressure signal, or they can be operated manually from the control room. To add protection in case of a small line break or for certain degraded transients, actuation of the relief valves to depressurize the reactor vessel will occur automatically when certain permissives are satisfied. Logic sequences resulting in automatic actuation are discussed in Sections 6.3.2.4, 7.3.1.4, and 15.6. [5.2-28]

The reactor relief valves are located on the main steam lines upstream of the first isolation valve and discharge directly to the pressure suppression pool. There are two independent sensor systems supplying the signals to all valves to operate, and all valves are powered by the same safety-grade power source which is separate from the HPCI power source. An additional power source is also available and is automatically switched over upon loss of the primary power source. [5.2-29]

The reactor safety valves are located on the main steam lines inside the primary containment. They are balanced, spring-loaded safety valves that discharge directly to the drywell atmosphere. The safety valves are the final protection against overpressurizing the vessel and are sized to prevent reactor pressure from exceeding the pressure limitations specified in the ASME Code. [5.2-30]

The dual-function Target Rock SRV discharges within the primary containment system, to the suppression chamber. This valve operates automatically on high reactor pressure approximately 100 — 135 psi above the operating pressure, but below the setting of the safety valves. It also serves in the automatic relief system and can be actuated on either automatic or manual signal. This valve will pass approximately 7% of turbine design steam flow. [5.2-31]

The relief valves and the dual-function SRV will also function to automatically depressurize the reactor, under certain conditions, following a loss-of-coolant accident as part of the automatic depressurization system (ADS). The ADS is discussed in Section 6.3. These valves normally open automatically on reactor overpressures of between 115 and 135 psi, and then close at lower, preset pressure levels. An additional function of these valves is to open and remain open below their preset closing pressures when signalled to do so following a LOCA that does not pressurize the drywell. This "remain open" signal is based on sustained signals indicating low-low water level simultaneously with pump operation for either the LPCI or the core spray function. [5.2-32]

By remaining open, these valves reduce the reactor pressure to the point where the LPCI system and/or the core spray system can accomplish reflooding of the reactor core. This permits the activation of the reactor core reflooding systems required for the various break sizes. To protect against a faulty "remain open" signal, a short time delay (120 seconds) is provided during which the operator can override the signal. [5.2-33]

To prevent inadvertent ADS actuation from a fire, power to the ADS logic can be removed by turning the ADS inhibit switch to the INHIBIT position. For severe fires which prevent safe shutdown by normal means, the ADS inhibit switch is turned to the INHIBIT position; and, as an additional precautionary measure, the power supply to the ADS logic circuit is subsequently opened in the 125 Vdc distribution panels. [5.2-34]

Another means to prevent inadvertent actuation of relief valves is to pull their fuses. Refer to the Safe Shutdown Report for a detailed description of relief valve actuation and it's affect on time lines for reactor water makeup during an Appendix R scenario. [5.2-35]

The relief valves and their discharge lines are also capable of being used in an alternate shutdown cooling mode. In this mode of core cooling, water would be supplied to the RPV by a core spray or LPCI pump which would fill the vessel above the steam line discharge nozzles. The water would then flow through the steam lines and the open relief valves back to the suppression pool and the suction side of the core spray or LPCI pump. Based on extensive research performed in response to NUREG 0737 item II.D.1, ERV's are capable of being used in the alternate shutdown cooling mode and all pipe

stresses and support loads for this mode of core cooling are within design allowables. [5.2-36]

Each of the four relief valves and the dual-function SRV discharge to the pressure suppression chamber via dedicated (one per valve) discharge lines. Analyses have shown that upon valve closure, steam remaining in the discharge line can condense, thereby creating a vacuum which will draw suppression pool water up into the discharge line. This "elevated water leg" condition is quickly alleviated by operation of the vacuum breaker on the discharge line; however, the condition is of concern since a subsequent actuation in the presence of an elevated water leg can result in unacceptably high thrust loads on the discharge piping. [5.2-37]

To prevent these unacceptable loads, the setpoints and control logic for the relief valves and the SRV have been designed to ensure that each valve which closes will remain closed until the normal water level in the discharge line is restored. This is accomplished by first establishing opening and closing setpoints such that all pressure induced subsequent actuations (after the "first pop") are limited to the two lowest set valves. These two valves are equipped with additional logic which functions in conjunction with the setpoints to inhibit valve reopening (via reactor repressurization or the ADS) for at least 10.5 seconds following each closure. (This compares with a calculated worst case elevated water leg duration time of 6.3 seconds). This combination of setpoint selection and control logic design satisfies the single failure criterion, and is sufficient to ensure that no credible scenario can result in actuation of a relief or SRV in the presence of an elevated water leg.

The relief and safety valves are provided with acoustic monitors. Vibration from steam discharging through the valve triggers an alarm in the control room if the valve is open. Indicating lamps and a test function are also provided for the acoustic monitors.

The relief valves and Target Rock SRV have indicating lamps in the control room which light if an opening signal is present. These inform the operator if the valve is receiving a signal to open.

Both the relief and safety valves are equipped with temperature elements and acoustic monitors that signal alarms in the control room if one of these valves opens. In addition, the safety valves are equipped with a 10-psig rupture disc in the discharge line. In the event that the temperature elements and acoustic monitors failed to detect a leak in the safety valves discharge lines, an inspection during a refueling outage would reveal it.

5.2.2.5 Mounting of Pressure-Relief Devices

The relief valves, safety valves, and the Target Rock SRV are mounted on the main steam lines. FSAR Figure 10.3-1 and P&IDs M-13, M-60, M-34, and M-76 show the distribution of these pressure relieving devices on the four main steam lines. Information on thrust, bending, and other loads plus the resulting stresses is contained in Tables 5.2-1 for Unit 1 and 5.2-2 for Unit 1 and 2.

5.2.2.6 Applicable Codes and Classification

The structural integrity of the reactor coolant pressure boundary is maintained at the level required by ASME Section XI, 2007 Edition through 2008 Addenda for the fifth inservice inspection (ISI) interval. [5.2-38]

5.2.2.7 Material Specification

Reactor coolant pressure boundary materials, including overpressurization protection materials, are discussed in Section 5.2.3.1.

5.2.2.8 Process Instrumentation

Process instrumentation is shown on P&ID M-13 (Unit 1) and P&ID M-60 (Unit 2).

5.2.2.9 System Reliability

Safety valve sizing (see 5.2.2.2.3) uses very conservative assumptions on relief valve availability and method of scram. Further discussions on failures and their effects are contained in Section 15.2.

5.2.2.10 Testing and Inspection

The relief valves and safety valves are inspected for cyclic strain and thermal stress. The safety valves are bench checked periodically for the proper setpoint. See Section 3.9 for inservice inspection and inservice testing (IST) of valves. [5.2-39]

5.2.3 Reactor Coolant Pressure Boundary Materials

5.2.3.1 Material Specifications

The principal pressure retaining materials and the appropriate material specifications for the reactor coolant pressure boundary components are defined in GE design and purchase specifications, or the specifications of other suppliers of RCPB components. [5.2-40]

5.2.3.2 Compatibility with Reactor Coolant

The importance of establishing and maintaining appropriate water chemistry conditions in the reactor coolant of boiling water reactor (BWR) nuclear power plants is well established. During the past decade, most operating BWRs (including Dresden and Quad Cities Stations) have experienced unanticipated pipe cracking problems that have resulted in a significant loss of availability, and have increased the total personnel radiation exposure associated with inspection and repair. The cause of these problems has been intergranular stress corrosion cracking (IGSCC) which results from the simultaneous occurrence of an aggressive environment, particular materials, and stress conditions. A contributing cause of these problems has been the formation of locally corrosive environments as the result of the ingress of impurities during operation. Exelon Generation Company (EGC) recognizes that if IGSCC is to be controlled, the appropriate water chemistry must be maintained in the primary system of the company's BWR plants. [5.2-41]

5.2.3.2.1 Boiling Water Reactor Water Chemistry

The BWR water chemistry control program establishes achievable ranges for water chemistry parameters where IGSCC is suppressed. Compliance with the program's impurity concentrations has been shown to reduce the rate of IGSCC and reduce the probability of initiating new cracks. Data from Dresden Unit 2 indicates that IGSCC in reactor recirculation piping can be suppressed by controlling impurity concentrations within the achievable ranges combined with injection of approximately one ppm hydrogen into the feedwater to reduce free oxygen. This approach to the prevention of cracking is called hydrogen water chemistry (HWC).

In addition to reducing IGSCC, research shows that the appropriate control of water chemistry will also assist in controlling radiation buildup, minimizing fuel failure, and minimizing damage to the turbine caused by chemistry.

Specific corporate water chemistry control requirements have been implemented at Quad Cities. These requirements reflect the current understanding of the role of chemical transport, impurity concentration, materials of construction, corrosion behavior, chemical analytical methods, and industry practice regarding the operation and integrity of the primary system. The specific requirements were primarily taken from the BWR Owners Group (BWROG) and Electric Power Research Institute (EPRI) Water Chemistry Guidelines, existing GE chemistry guidelines, and known or suspected contaminant concerns at EGC's BWRs. These specific requirements are provided in system chemistry procedures.

Noble Metal Chemical Addition (NMCA) has been developed by General Electric as a method to enhance the effectiveness of Hydrogen Water Chemistry (HWC) in mitigating Intergranular Stress Corrosion Cracking (IGSCC) in vessel internal components. Additionally, use of NMCA allows lower injection rates of HWC which in turn reduces plant radiation exposure over the life of the plant. NMCA process deposits a very thin discontinuous layer of noble metals onto all wetted surfaces during the injection process. The treated surfaces will behave catalytically and promote oxidant-hydrogen recombination. This results in low corrosion potential of components at low hydrogen injection rates. Higher reactor water conductivity is anticipated during the application due to the effect of non-corrosive noble metals on the measured conductivity (reference Table 5.2-4).

5.2.3.2.1.1 Training

A training program for personnel involved in water chemistry control is required to implement the corporate policy. The goal of this program is to improve the overall awareness among station personnel of the need for chemistry control. Personnel required to be trained include all chemistry staff, chemistry technicians, licensed operators, and selected systems engineering and maintenance personnel.

5.2.3.2.2 Compatibility of Construction Materials with Reactor Coolant

The materials of construction exposed to the reactor coolant consist of the following: [5.2-42]

A. Solution-annealed austenitic stainless steels (both wrought and cast) Types 304, 304L, 316, 316L, and XM-19,

B. Nickel base alloys — Inconel 600 and Inconel 750X,

C. Carbon steel and low alloy steel,

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

E. Colmonoy and Stellite hardfacing material.

All of these materials of construction, with the possible exception of Inconel 600, are resistant to stress corrosion cracking in the reactor coolant. General corrosion of these materials, except carbon and low alloy steel, is negligible. Conservative corrosion allowances are provided for all exposed surfaces of carbon and low alloy steels.

Contaminants in the reactor coolant are controlled to very low limits set by the reactor water quality specifications. No detrimental effects will occur on any of these materials from allowable contaminant levels in high purity reactor coolant. Radiolytic products have no adverse effects on these materials.

5.2.3.2.3 Compatibility of Construction Materials with External Insulation and Reactor Coolant

The materials of construction exposed to external insulation are: [5.2-43]

- Solution annealed austenitic stainless steels (Types 304, 304L, and 316), and
- Carbon and low alloy steel.

Two types of external insulation are employed at Quad Cities. The first, reflective metal insulation, does not contribute to any surface contamination and has no effect on construction materials. The second, nonmetallic insulation, is used on stainless steel piping and components and complies with the requirements of the following industry standards:

- ASTM C692-71, Standard Methods for Evaluating Stress Corrosion Effects of Wicking Type Thermal Insulation on Stainless Steel (Dana Test); and
- RDT-M12-1T, Test Requirements for Thermal Insulating Materials for Use on Austenitic Stainless Steel, Section 5 (KAPL Test).

Chemical analyses are required to verify that the leachable sodium, silicate, and chloride in this insulation are within acceptable levels. The insulation is packaged in waterproof containers to avoid damage or contamination during shipment and storage.

Since there are no additives in the reactor coolant, leakage would expose materials to high purity, demineralized water. Exposure to demineralized water would cause no detrimental effects.

5.2.3.3 Fabrication and Processing of Ferritic Materials

This subsection describes how Appendix G requires the determination of pressure - temperature limits for ferritic materials to achieve acceptable stresses, and the adjustment of these limits to account for the effects of accumulated neutron irradiation.

5.2.3.3.1 Fracture Toughness — Reactor Pressure Vessel

Title 10 CFR Part 50, Appendix G, "Fracture Toughness Requirements," requires that pressure-temperature limits be established for reactor coolant system heatup and cooldown operations, inservice leak and hydrostatic tests, and normal reactor operation. These limits are required to ensure that the stresses in the RPV remain within acceptable limits. They are intended to provide adequate margins of safety during any condition of normal operation, including anticipated operational transients. [5.2-44]

Specific pressure-temperature limits are presented and discussed in Section 5.3.2 which indicate EGC's commitments regarding 10 CFR 50, Appendix G and Regulatory Guide 1.99, Revision 2.

5.2.3.3.2 Fracture Toughness — Reactor Coolant Pressure Boundary Minus Reactor Pressure Vessel

The relief valves and the safety valves were exempted from fracture toughness requirements because Section III of the 1965 ASME Code did not require impact testing on valves with inlet connections of 6 inches or less nominal pipe size. [5.2-45]

Main steam isolation valves were also exempted from fracture toughness requirements because Section III of the 1965 ASME Code with Summer 1965 Addenda did not require brittle fracture testing on ferritic pressure boundary components when the system temperature was in excess of 250°F at 20% of the design pressure.

The recirculation pumps were exempted from the ASME Code and the USAS Code for pressure piping because of their classification as machinery. This is more completely discussed in Section 5.4.1.1

5.2.3.3.3 Control of Welding

5.2.3.3.3.1 Control of Electroslag Weld Properties

Electroslag welding of longitudinal seams of the RPV was performed in accordance with ASME Section III, Code Case 1355. This code case and other code cases applying to materials and fabrication are identified in the vessel manufacturer's fabrication report and on the manufacturer's data report, Form N-1A. A detailed description of the electroslag welding process used on Quad Cities is contained in Amendment 13/14, Appendix F, Dresden FSAR[2]. The electroslag welding process was not used in the fabrication of the recirculation pump casings. [5.2-46]

5.2.3.4 Fabrication and Processing of Austenitic Stainless Steels

This section provides information relative to fabrication and processing of austenitic stainless steel for components in the RCPB.

5.2.3.4.1 Avoidance of Stress Corrosion Cracking

The methods and actions regarding stress corrosion cracking are discussed in subsequent sections.

5.2.3.4.1.1 Avoidance of Significant Sensitization

Sensitization of austenitic stainless steels during fabrication can induce residual stresses that promote IGSCC.

The Quad Cities pressure vessels were manufactured with some components of furnace-sensitized stainless steel material. In particular, vessel nozzle safe-ends were sensitized because they had been attached to the vessel prior to furnace annealing. [5.2-47]

All the nozzles having furnace-sensitized safe-ends were replaced with non-sensitized components and are listed below as follows: [5.2-48]

A. Recirculation inlet,

B. Recirculation outlet,

C. Core spray,

D. Six-inch instrumentation (top head),

E. Head vent,

F. Jet pump instrumentation,

G. Control rod drive hydraulic system return (safe-end removed and nozzle capped),

H. Core differential pressure/standby liquid control, and

I. Two-inch instrumentation.

In each case, except for the two-inch instrument nozzles, the replacement process left a portion of the sensitized stainless steel buttering on the nozzle when the sensitized safe-end was cut off. This buttering was overlay clad on the inside of the nozzle. The two-inch instrument nozzles were inconel and the safe-end and entire weld were removed and replaced.

During Q2R21 refueling outage, a leak was discovered from the Unit 2 N-11B two-inch instrument nozzle during the system pressure test. A portion of the old nozzle was removed and a new nozzle was welded to the outside of the vessel. The repair is reconciled to the original code of construction.

The pipe socket on the inboard end of the core differential pressure/standby liquid control nozzle was cut off and a new one formed by weld build-up and machining.

The upper steam dryer guide rod brackets had been furnace-sensitized and were replaced. The weld pads and surrounding cladding were overlay clad. Although the following equipment had not itself been furnace-sensitized, the weld pads and surrounding cladding were overlay clad:

A. Steam dryer lower guide rod brackets,

B. Surveillance specimen holder brackets,

C. Core spray sparger brackets,

D. Steam dryer support brackets,

E. Feedwater sparger brackets,

F. Shroud head guide rod brackets, and

G. Jet pump riser brace.

In order to avoid partial and/or local sensitization of austenitic stainless steel during heat treatments and welding operations for reactor core structural members and reactor coolant system pipe components, the control of heat input was carefully monitored and procedurally controlled with maximum interpass temperature limited to 350°F. Heat treating of reactor core structural members during manufacturing was limited to a maximum of 800°F. No heat treating was permitted during field erection. General Electric quality control inspectors ensured conformance to approved procedures both at the vendor's shop and at the site. The types of weld metal used for safe-ends of components within the RCPB are inconel and stainless steel.

5.2.3.4.1.2 Electroslag Welds (Regulatory Guide 1.34)

Refer to 5.2.3.3.3.1 for information on control of electroslag weld properties.

5.2.3.5 Intergranular Stress Corrosion Cracking

Generic Letter 88-01, "NRC position on IGSCC in BWR Austenitic Steel Piping", dated January 25, 1988, provides the NRC staff's positions and guidelines concerning the piping materials used for the reactor coolant pressure boundary. Subsequently, the final safety evaluation of EPRI Report TR-113932 ("BWR Vessel and Internals Project, Technical Basis for Revisions to Generic Letter 88-01 Inspection Schedules (BWRVIP-75)"), dated May 14, 2002, revised the Generic Letter (GL) 88-01 inspection schedules. The BWRVIP-75 revised inspection schedules were based on consideration of inspection results and service experience gained by the industry since issuance of GL 88-01, and includes additional knowledge regarding the benefits of improved BWR water chemistry. The information that follows is a point-by-point comparison of the requirements of the generic letter and the measures in place at Quad Cities. The numbering corresponds to the generic letter, and exceptions are indicated.

5.2.3.5.1 Programs to Mitigate IGSCC

Exelon Generation Company has an integrated program to mitigate IGSCC which includes the following: [5.2-49]

A. Hydrogen water chemistry (HWC) (see Section 5.4.3 for information on HWC);

B. Stress improvement through induction heat stress improvement (IHSI) and mechanical stress improvement program (MSIP);

C. Weld overlays, including overlays with pipelocks, for flaw indications in excess of ASME Section XI, Subsection IWB-3500 limits (overlays meet NUREG-0313, Rev. 2 requirements); and

D. System modifications, which include removal of the head spray line, removal and capping of the control rod drive return line, and replacement of reactor water cleanup piping with conforming material.

5.2.3.5.2 Augmented Inspection Program

Exelon Generation Company's augmented inspection program conforms basically to positions on inspection schedules, methods and personnel, and sample expansion delineated in GL 88-01 and BWRVIP-75 as approved by the NRC. Exceptions are for welds that are not accessible for non-destructive examination (NDE).

5.2.3.5.3 NRC Notifications

Exelon Generation Company will notify the NRC for flaw indications that exceed IWB-3500 limits or changes in welds with flaw indications following in-house determinations and/or recommendations.

5.2.4 Inservice Inspection and Testing of Reactor Coolant Pressure Boundary

Inservice examination and tests of ISI Class 1, 2, 3, and MC. (See Section 3.9.6 for explanation of ISI Class vs. ASME Class. Only the RPV and skirt are ASME Class 1 in the RCPB). Components will be performed in accordance with Section XI of the ASME Code and applicable Addenda as required by 10 CFR 50.55a(g), except where specific written relief has been granted by the NRC. Certain requirements of later editions or addenda of Section XI are impracticable to perform on Quad Cities because of the design, component geometry, and materials of construction. For this reason, 10 CFR 50.55a(g)(6)(i) authorizes the NRC to grant relief from certain requirements after making the necessary findings. [5.2-50]

During the 1990 refueling outage, cracks were discovered in the Unit 2 vessel head. A supplemental reactor head and upper shell inspection plan has been implemented for Unit 2 in lieu of the successive examination requirements of ASME Section XI.

The inservice testing of pumps and valves is discussed in Section 3.9.6. The ISI/IST of Class 2, 3, and MC components is discussed in Section 6.6. This ISI program for Class 1, 2, and 3 is based on the requirements of Section XI of the ASME Code 2007 Edition through 2008 Addenda. The program for Class MC is based on the requirements of Section XI of the ASME Code 2001 Edition through the 2003 Addenda. Table 5.2-3 lists the systems included in the ISI program. [5.2-51]

5.2.4.1 System Boundary Subject to Inspection

In addition to the RPVs and their support skirts, components and supports within ASME Section III, Class 1 boundaries are subject to the requirements of ISI per ASME Section XI. The P&IDs and IWE (MC) program drawings define the applicability of the ISI Class 1, 2, 3, MC, and NC IST program for systems subject to the requirements of ASME Section XI. These ISI boundaries on the P&IDs are limited to safety-related systems which contain water, steam or radioactive materials and, in accordance with Regulatory Guide 1.26, this includes some non-RCPB components. [5.2-52]

Some pumps and valves not included in the ISI Class 1, 2 or 3 boundaries marked on the P&IDs have been included in the IST program in recognition of their importance to safe plant operation. These pumps and valves are noted as ISI Class "NC" meaning not classified ISI Class 1, 2, or 3 but having augmented quality requirements. Section 3.9 contains a discussion of ISI/IST for pumps and valves.

5.2.4.2 Arrangement and Accessibility

The "as built" Quad Cities design does not permit ready access for volumetric examination of RPV shell welds in accordance with the requirements of ASME Section XI - Rules for Inservice Inspection of Nuclear Power Plant Components. Exelon Generation Company recognizes the importance of inspecting welds that are presently inaccessible and will study and implement, if practicable, new means to include these welds within the ISI program as such means become available. [5.2-53]

5.2.4.3.1 Examination Techniques and Procedures

The methods, techniques, and procedures used for ISI comply with subarticle IWA-2200 of ASME Section XI. [5.2-54]

Liquid penetrant or magnetic particle methods will be used for surface examinations, and radiographic and/or ultrasonic methods will be used for volumetric examinations.

Periodic visual inspections are also made of nozzle-to-vessel weld joints to assure that no deterioration is occurring. [5.2-55]

5.2.4.4 Inspection Intervals

As defined in subarticle IWA-2430 of ASME Section XI, the inspection interval for ISI Class I components will be 10 years. The interval may be extended by as much as one year to permit inspections to be concurrent with plant outages. [5.2-56]

The inspection schedules are in accordance with IWB-2400, IWC-2400, IWD-2400, IWE-2400 for Class 1, 2, 3, and MC respectively. It is intended that inservice examinations be performed during normal plant outages such as refueling shutdowns or maintenance shutdowns occurring during the inspection interval. No examinations will be performed which require draining of the RPV or removal of the core solely for the purpose of accomplishing the examinations.

A supplemental reactor head and upper shell inspection plan for Unit 2 has been implemented in lieu of the successive examination requirements of ASME Section XI IWB-2420(b).

5.2.4.5 Examination Categories and Requirements

The extent of the examinations performed and the methods utilized (e.g., volumetric, surface, visual) are in accordance with ASME Section XI, Tables IWB-2500-1, IWC-2500-1, IWD-2500-1, IWE-2500-1, and IWF 2500-1 for Class 1, 2, 3, and MC components and component supports. [5.2-57]

5.2.4.6 Evaluation of Examination Results

The standards for examination evaluation meet the requirements of Section XI, IWB-3000, "Acceptance Standards." The program for flaw evaluation follows Table IWB-3410-1, "Acceptance Standards." [5.2-58]

The program regarding repairs of unacceptable indications or replacement of components containing unacceptable indications meets the requirements of Section XI, IWA-4000, "Repair Procedures." [5.2-59]

5.2.4.7 System Leakage and Hydrostatic Pressure Tests

System leakage and hydrostatic tests are conducted in accordance with IWB-5000, "System Pressure Tests." Section 5.3.2.2 provides additional requirements. [5.2-60]

5.2.5 Detection of Leakage Through Reactor Coolant Pressure Boundary

The reactor vessel head is flanged to the vessel and sealed with two concentric O-rings. The area between the two O-rings is monitored to provide an indication of leakage from the inner O-ring seal. [5.2-61]

The primary reactor coolant leak detection system consists of four different and independent means by which leakage can be detected. These are monitoring the: [5.2-62]

A. Sumps in the containment;

B. Drywell temperature and pressure changes;

C. Drywell air coolers cooling water differential temperature changes; and

D. Drywell atmosphere activity level changes.

There were no startup (preop) tests to verify system operability or sensitivity, nor was the drywell subjected to leaks to determine response.

Once a leak has been detected within the drywell by any of the methods covered in this section, it becomes necessary to locate, and if possible, determine its magnitude and rate of change with time. The smaller the leak the more difficult it becomes in locating its source. For example, through the use of a continuous air sample system, it is possible to detect changes in radioactive nuclides from one 24-hour period to the next. Very small leaks are thus possible to detect. [5.2-63]

The systems described in this section would be used by the operator to find the source of leakage. The systems are remote in nature and provide the operator a method of cross checking to locate the source of leakage or the area in the drywell in which the leak has developed.

5.2.5.1 Containment Sumps

One method of detecting leakage from the primary coolant pressure boundary is monitoring the flow out of the drywell floor drain sump and the equipment drain sump. All free unidentified leakage from the primary coolant pressure boundary will drain to the floor drain sump. [5.2-64]

All controlled identified leakage (seals, etc.) is piped to the equipment drain sump and monitored separately. Therefore, the flow rate from the floor drain sump is the total unidentified leakage of reactor coolant and is the principal leakage of concern from a safety standpoint.

The normal background flow out of the floor drain sump is quite low, approximately 1 gal/min. Therefore, leaks from the primary system on the order of 1 to 3 gal/min, over the period of about 1 day, can be detected and measured even when large allowances are made for variations in background leakage.

5.2.5.2 Drywell Temperature and Pressure

Temperature sensors are located at 34 different points in the drywell. Six points feed into a 12-point recorder in the Control Room, and other 28 points feed into a 30-point recorder on the Drywell Environs Rack. The steady-state temperature in the drywell would be increased about 1°F for a 2 gal/min leak from the primary system. [5.2-65]

The drywell pressure is monitored and indicated in the control room. The steady-state pressure in the drywell would be increased about 0.03 psi for a 1 gal/min leak of reactor coolant.

5.2.5.3 Air Coolers Temperature Differential

Temperature sensors are installed in the reactor building closed cooling water discharge from each of the seven drywell air coolers. The temperatures are recorded. The inlet water temperature of the reactor building closed cooling water header to the coolers is also recorded. An increase in the differential temperature across the coolers of 5°F would typically be equivalent to approximately a 3 gal/min steam leak or a 7.5 gal/min liquid leak.

5.2.5.4 Other Drywell Leakage Monitors

The following systems are used by the operator to determine that leakage exists within the drywell. These various systems, operating together or singly, provide the information to the operator that a possible problem has developed within the drywell. [5.2-66]

5.2.5.4.1 Air Sample System

Each drywell is equipped with 22 air sampling points, and 3 sample lines to the oxygen analyzer. Each suppression chamber has one sampling point. The air sample can be drawn through 1/2 inch tubing from the various sample points. Primary containment isolation is provided by either redundant manual or automatic isolation valves. At the local rack, the samples may be filtered using a filter cartridge holder and the air returned to the drywell, or a grab sample may be taken at the rack for laboratory analysis. A continuous air monitor is provided from one of the oxygen analyzer points. This air monitor will count gross beta activity which will be recorded, and will alarm on an increase. This provides an indication that a leak has occurred.

5.2.5.4.2 Pressure Switches

A pressure switch will alarm if failure of the inner O-ring takes place on the reactor vessel.

5.2.5.4.3 Acoustic Monitors

An acoustic monitor is mounted at each safety and relief valve (including the SRV) in the drywell, 13 in all. Leaks from these valves would cause vibrations to be picked up by the monitor, sounding an alarm in the control room. Each monitor has its own set of indicating lights in the control room.

5.2.5.5 Leakage Rate Limits

The limiting leakage conditions included in the Technical Specifications are that with the plant in MODES 1, 2 and 3: there shall be no pressure boundary leakage; unidentified coolant leakage into the primary containment shall be less than or equal to 5 gal/min; total leakage in the primary containment shall be less than or equal to 25 gal/min averaged over the previous 24 hour period; and unidentified leakage increases shall be less than or equal to 2 gal/min within the previous 24 hour period (in MODE 1). [5.2-67]

In the event that the unidentified leakage limit, or total leakage limit is exceeded, the Technical Specifications allow 4 hours to reduce the leakage to within limits. In addition, if the unidentified leakage increase limit is exceeded, the Technical Specifications allow 4 hours to reduce the leakage increase to within limits or to identify that the source of the unidentified leakage increase is not IGSCC susceptible material. In the event that the required actions cannot be performed within the 4-hours period, or if pressure boundary leakage exists, the affected unit must be in hot shutdown within 12 hours and cold shutdown within 36 hours.

In addition, a limit is procedurally administered for the volume pumped from the drywell equipment drain sump (identified leakage). This limit is 9600 gallons in an 8-hour period (20 gal/min).

5.2.5.6 High/Low Pressure Interfaces

The core spray, (RHR)/LPCI, and the RHR shutdown cooling suction are all monitored for reactor coolant system leakage into the system by pressure switches located in the pump discharge lines. These switches activate a high pressure alarm in the main control room when the line pressure exceeds the alarm setpoint. These lines are protected by relief valves which are bench tested at least every third refueling outage. [5.2-68]

5.2.5.7 Compliance With Regulatory Guide 1.45

The various leak detection systems and capabilities, as described herein, detect RCPB leakage, both identified and unidentified. These sensitive and diverse systems meet the acceptance criteria of Regulatory Guide 1.45. [5.2-69]

5.2.6 Detection of Leakage Beyond the Reactor Coolant Pressure Boundary

Provisions have been made in the design of Quad Cities to detect leakage from vital fluid-carrying systems beyond the limits of the RCPB. Included in these provisions, which are discussed below, are floor drain sumps, area radiation monitoring, area temperature monitoring, and visual inspection. [5.2-70]

5.2.6.1 Floor Drain Sumps

Floor drain sumps and pumps are provided within the secondary containment (reactor building). Leakage from fluid-carrying systems would be detected by an increased frequency of sump pump operation, increased input to the radwaste floor drain collector system, or high water level in the sump with ultimate annunciation in the control room.

5.2.6.2 Area Radiation Monitoring

Area radiation monitors are provided throughout the plant equipment and operating areas. These monitors can detect leakage from radioactive sources. Activity levels are indicated in the control room. Leakage would be detected by an increased level of activity beyond normal operating background with ultimate high activity annunciation in the control room. Further information on area radiation monitoring is contained in Section 12.3.

5.2.6.3 Area Temperature Monitoring

Area temperature monitors are provided in appropriate areas and equipment spaces of the plant. These monitors will detect leakage from high temperature fluid-carrying systems. Temperature indication is provided in the control room. Leakage would be detected by an increase in the normal operating temperature of the area with ultimate high temperature annunciation in the control room and, in some cases, automatic isolation of the system. Systems provided with automatic isolation on detection of high area temperatures are HPCI, RCIC, RWCU and the MSIVs. [5.2-71]

5.2.6.4 Visual Inspection of Equipment and Operating Areas

Access to equipment spaces to permit normal routine visual inspection is provided to those areas of controlled occupancy as well as those of continuous occupancy, radiation levels permitting. [5.2-72]

5.2.7 References

1. NEDE-24988-P, "Analysis of Generic BWR Safety/Relief Valve Operability Test Results," General Electric.

2. Dresden 2 and 3 FSAR, Amendment 13/14, Appendix F.

3. Deleted.

4. Deleted.

5. GE-NE-A22-00103-10-01, “Dresden and Quad Cities Extended Power Uprate, Task T0900: Transient Analysis, Revision 0.”

6. “Reference Safety Report for Boiling Water Reactor Reload Fuel,” Westinghouse Topical Report CENPD-300-P-A (Proprietary), CENPD-300-NP-A (Non-Proprietary), July 1996.

Table 5.2-1

SUMMARY OF STRESSES ON RELIEF VALVE PARTS FOR UNIT ONE

	Relief Valve Part	Allowable Stress at Design Temperature	Maximum Stress
1.	Valve Body	$1.0S = 18{,}760$ psi $1.5S = 28{,}140$ psi	$\sigma_m = 1{,}533$ psi* $\sigma_m + \sigma_b = 22{,}954$ psi*
2.	Turnbuckle	$1.0S_{yield} = 28{,}100$ psi	$\sigma = 15{,}332$ psi
3.	Pilot Valve Tube	$1.0S_{yield} = 23{,}580$ psi	$\sigma = 22{,}115$ psi
4.	Solenoid Assembly Mounting Bracket Hold-down Bolts	$1.0S_{yield} = 105{,}000$ psi	$\sigma = 3{,}036$ psi
5.	Solenoid Assembly Mounting Bolts	$1.0S_{yield} = 105{,}000$ psi	$\sigma = 3{,}584$ psi

* Note that the ERV valve body stresses were originally qualified by enveloping the existing Dresden and Quad Cities ERV pipe loadings. The numbers in this table reflect this original condition. The present ERV valve body stress is qualified based on the qualification of the ERV pipe stress. See UFSAR Section 3.9 for more details on the current pipe stress summary.

Table 5.2-2

FORCES AND STRESSES IN SUPPORTING STRUCTURE AT QUAD CITIES 1 & 2 (HISTORICAL)

Member	Axial (kip)	Shear (kip)	Moment (kip-ft) Major Axis	Moment (kip-ft) Minor Axis	Maximum Fiber Stress (ksi)	Shear Stress (ksi)	A	B
Target Rock SRV Line: **								
Beam 906-903 8WF58	0.5	20.0	123.0	20.5	41.95	4.48	1.226	0.971
Beam 601-610 12FW45	1.5	8.0	57.4	18.2	29.56	1.97	0.864	0.684
Beam 400-406 12WF36	2.1	1.0	17.0	3.7	10.34	0.27	0.302	0.239
Beam Column A900-802 6WF25	17.0	13.0	20.5	0.5	18.11	6.38	0.529	0.419
Electromatic Valve Lines :**								
Beam 906-903 8WF58	0.5	16.0	98.7	17.3	34.2	3.59	1.000	0.792
Beam 601-610 12WF45	2.2	6.5	48.5	15.7	25.4	1.61	0.742	0.588
Beam 400-406 12WF36	2.0	1.5	13.7	3.7	9.5	0.40	0.277	0.220
Beam Column A900-802 6WF25	13.5	10.5	16.7	0.2	14.3	5.15	0.418	0.331

A = Shear Stress/(0.95)(F_y)
B = Shear Stress/F_y*

** These values are being retained for historical purposes only. Refer to UFSAR Section 3.9 for current SRV/ERV pipe support stress summaries.

Table 5.2-3

LIST OF SYSTEMS INCLUDED IN THE ISI PROGRAM

System	Class
Control Rod Drive	1 & 2
Residual Heat Removal (RHR)	1 & 2
RHR Service Water	3
Standby Liquid Control (SBLC)	1 & 2
Reactor Water Cleanup	1
Core Spray	1 & 2
High Pressure Coolant Injection (HPCI)	1 & 2
Main Steam	1
Feedwater	1 & 2
Diesel Generator Cooling Water	3
Reactor Recirculation	1
Reactor Core Isolation Cooling (RCIC)	1
Control Room HVAC	3
Drywell	MC
Suppression Chamber	MC
Vent System	MC

Table 5.2-4

REACTOR COOLANT SYSTEM CHEMISTRY LIMITS

OPERATIONAL MODE(s)	Chlorides	Conductivity (μmhos/cm @ 25°C)
1	$\leq$ 0.2 ppm	$\leq$ 1.0
2 and 3	$\leq$ 0.1 ppm	$\leq$ 2.0**

** During Noble Metal Chemical Addition (NMCA), <=10.0 μmhos/cm @ 25°C is the limit

5.3 REACTOR VESSELS

This section presents pertinent data on the Quad Cities reactor pressure vessels (RPVs). Unless otherwise noted, the information presented applies to both Unit 1 and Unit 2 RPVs.

5.3.1 Reactor Vessel Materials

The RPV materials and fabrication methods conform to the ASME Boiler and Pressure Vessel Code (ASME Code) 1965 Edition and the Summer 1965 Addendum as referenced in Section3.2.8.4. Inservice inspection (ISI) techniques conform to ASME Section XI with approved exceptions as noted in Section 5.2.4. [5.3-1]

5.3.1.1 Material Specifications

Reactor vessel material specifications are discussed in Section 5.2.3.1. Additional information on RPV materials is contained in Section 5.3.3.2.

5.3.1.2 Special Processes Used for Manufacturing and Fabrication

The Quad Cities Unit 1 RPV was fabricated entirely in the United States by Babcock & Wilcox (B&W). The Unit 2 RPV was fabricated by several different vendors, including one in Holland, as noted in the following paragraphs. [5.3-2]

Fabrication work on the Unit 2 bottom head assembly and lower shell course was performed by the Rotterdam Dockyard Company (RDM) in Rotterdam, Holland. These two pieces were seam-welded together and returned to the United States as a fully completed subassembly including control rod drive (CRD) stub tubes, shroud support skirt, and vessel support skirt. [5.3-2a]

The CRD stub tube material is Inconel SB167, Code Case 1336, Paragraph 1. The stub tubes were joined to the vessel bottom by a weld on the Inconel-clad surface which makes a full penetration of the stub tube wall as specified in Figure N-462.4(e) of the ASME Code, 1965, Section III. The toe of this weld was removed by the finished counterbore.

All work on Unit 2 was performed and documented in accordance with ASME Section III. The procedures required by the attachment to the National Board of Boiler and Pressure Vessel Inspectors' letter of July 24, 1968, were implemented by providing the Illinois State Board of Boiler Rules with the required documentation. This documentation included copies of all welder qualification test reports and performance test reports for each welder.

All other components of the Unit 2 core internals and primary system were of domestic manufacture. For example, B&W completed the circumferential seam weld which attached the upper shell course to the RPV flange.

Chicago Bridge and Iron (CB&I), which completed fabrication of the Unit 2 RPV prior to its shipment to the plant site, provided a certification comparable to the ASME Code N-1A form. The following footnote was included in that certification: [5.3-3]

> "This unstamped vessel was built as a `State Special' based on agreements between the State of Illinois and Commonwealth Edison Company. A portion of the vessel was fabricated by Rotterdam Dockyard Company. This vessel was not stamped because Rotterdam Dockyard Company does not hold an ASME certificate of authorization. Procedures equivalent to the requirements of the ASME Code were used."

Electroslag welding of longitudinal seams of the RPV was performed by B&W in accordance with ASME Section III, Code Case 1355 (See Section 5.2.3.3.3.1 for further details). [5.3-4]

5.3.1.3 Special Methods for Nondestructive Examination

Standard methods, in use at that time, were used for nondestructive examinations except for inspection of the CRD stub tubes as explained in the following paragraphs.

Inspection of the CRD stub tube shop welds was accomplished by progressive and final dye penetrant inspection and by ultrasonic (UT) inspection from the finished counterbore, all as required by ASME Section III, Paragraph N-462.4(e). The UT inspection exceeded the ASME Code requirements in that it covered the weld metal in addition to the base metal, heat-affected zone, and weld cladding. Also, the sensitivity used for UT testing was "high gain" and exceeded the ASME Code requirements. [5.3-5]

A similar "high gain" UT test was applied to the CRD stub tube field welds in addition to progressive dye penetrant testing.

5.3.1.4 Special Controls for Ferritic and Austenitic Stainless Steels

Regulatory Guides, as such, did not exist at the time the Quad Cities RPVs were fabricated. Information related to specific Regulatory Guides (as requested in Regulatory Guide 1.70, Rev. 3) is provided below to correlate actual past practice with current requirements. Unless otherwise stated, there has been no commitment to the Regulatory Guides.

Regulatory Guide 1.34

Electroslag welding of longitudinal seams was performed in accordance with ASME Section III, Code Case 1355 as discussed in Section 5.2.3.3.3.1. [5.3-6]

Regulatory Guide 1.44

Section 5.2.3.4.1.1 discusses the use of sensitized stainless steel and the actions to remove/control sensitized components.

Regulatory Guide 1.50

Preheat temperatures used when welding low alloy steel components (shells, flanges, plates) met applicable requirements or had contract variations approved by GE, the vendor responsible for supplying the RPV. [5.3-7]

Regulatory Guide 1.99

Section 5.3.2.1 contains information on compliance with the methodology in Regulatory Guide 1.99, Rev. 2.

Regulatory Guide 1.190

Regulatory Guide (RG) 1.190 provides state of the art calculation and measurement procedures that are acceptable to the NRC for determining Reactor Pressure Vessel neutron fluence. RPV fluence has been evaluated using a method in accordance with the recommendations of RG 1.190. Future evaluations of RPV fluence will be completed using a method in accordance with the recommendations of RG 1.190 (as noted in Reference 3).

5.3.1.5 Fracture Toughness

Sections 5.2.3.3.1 and 5.3.2.1 describe fracture toughness provisions for the Quad Cities Units 1 and 2 RPVs.

5.3.1.6 Material Surveillance

Vessel material surveillance samples are located within the reactor vessel to enable periodic monitoring of changes in material properties with exposure. The samples include specimens of the base metal, weld zone metal, heat affected zone metal, and standard specimens. These specimens receive neutron exposures more rapidly than the vessel wall material of interest (i.e., the innermost 25% of vessel wall thickness) and therefore lead it in integrated neutron flux. The neutron exposure rate of the average specimen at the core midplane is approximately 1.2 times the exposure rate of the adjacent inside surface of the vessel wall. There were 401 samples initially inserted in the vessel. Table 5.3-1 provides the location and status of the material specimens. [5.3-8]

In 2003, the NRC approved Quad Cities participation in the BWR Vessel and Internals Project (BWRVIP) Integrated Surveillance Program (ISP) as described in BWRVIP-78 and BWRVIP-86 in Reference 2. The NRC approved the ISP for the industry in Reference 2 and approved Quad Cities participation in Reference 3. The ISP meets the requirements of 10 CFR 50 Appendix H and provides several advantages over the original program. The surveillance materials in many plant-specific programs do not represent the best match with the limiting vessel beltline materials since some were established prior to 10 CFR 50 Appendix H requirements. Also, the ISP allows for better comparison to unirradiated material data to determine actual shifts in toughness. Finally, for many plants, ISP data will be available sooner to factor into plant operations since there are more sources of data.

The current withdrawal schedule for both units is based on the NRC-approved revision of BWRVIP-86 (Reference 2). Based on this schedule, Quad Cities is not scheduled to withdraw an additional material specimen. [5.3-9]

5.3.1.7 Reactor Pressure Vessel Fasteners

The top head of the RPV is secured to the vessel with studs, nuts, and spherical washers. Nut torquing and detorquing is accomplished using a stud tensioner. Technical Specifications require that the RPV head bolting studs (or closure studs) not be under tension unless the metal temperature of the vessel shell immediately below the vessel flange is at or above 83°F. This value (83°F) comes from the reference temperature (RT_{NDT}) and ASME Code considerations as discussed in Section 5.3.2. [5.3-10]

A fatigue usage analysis dated May 1999, demonstrated that the cumulative fatigue usage factor (CFUF) for the RPV closure studs would remain below 1.0 for current forty-year design life. A previous fatigue usage analysis dated March 1990, using then current duty cycle values, originally predicted that the RPV closure studs would reach the CFUF limit of 1.0 in 1998. This prediction was recalculated using actual cycle data through November 1997, to demonstrate that the CFUF limit would be reached in 2002. The purpose of the May 1999 analysis was to reduce conservatism used in the original vessel closure stud analysis and to qualify the studs for a forty-year design life using an updated fatigue evaluation. The primary means of reducing the fatigue usage was to use the actual number of operating cycles and perform new cycle pairing based on stress ranges and number of occurrences. Further reduction in fatigue usage was accomplished by using the appropriate ASME Code fatigue curve of 2.7Sm versus 3Sm. The results of this analysis show that the CFUF for the vessel closure studs is less than 1.0 for both Units 1 and 2 at the end of the forty-year design life. Further analysis was performed in 2003 in support of flood-up of the RPV using Feedwater/Condensate resulting in additional limitation on the number of several vessel stress cycles as described in Table 3.9-1A. The results of these analyses show that the usage factor meets the allowable limit of 1.0 established in the ASME Section III Code and as a result justifies forty years of operation. [5.3-11]

5.3.2 Pressure - Temperature Limits

Fast (>1 MeV) neutron irradiation above 10^{17} nvt begins to affect the mechanical properties of ferritic steel. The most important consideration is that of the change in the temperature at which ferritic steel breaks in a brittle rather than a ductile mode (referred to as the Nil Ductility Transition Temperature or NDTT). The NDTT increases with increasing irradiation. ASME Section III, N-446 specifies the design conditions for determination of the NDTT. Extensive tests have established the magnitude of changes in the NDTT as a function of the integrated neutron dosage. [5.3-12]

The SA 302B steel, with fabrication procedures specified by the ASME Code and by GE, is relatively insensitive to neutron irradiation. In fact, no change in the Adjusted Reference Temperature (ART) is expected to occur at neutron exposures less than 4.0×10^{17} nvt.

Originally, the flux levels were calculated using a modified Albert-Welton point kernel[1] which was originally developed as an approximate method of calculating the attenuation from a point fission source in water. The method represents fast neutron attenuation in water by a function which was experimentally fitted to data obtained for neutron attenuation by hydrogen in water. The nonhydrogenous portion of the attenuation was approximated by energy independent removal cross sections. The attenuation coefficients were obtained by fitting the kernel to the Oak Ridge Bulk Shielding Reactor water centerline data. This

method was incorporated into a computer program that integrated the contribution of discrete source points in the reactor volume to each point where flux was to be calculated.

The form of the Albert-Welton point kernel was developed to calculate dose rate. To obtain flux densities using this kernel, the normalization constant,

$$\alpha = 7.29 \text{ x } 10^9 \frac{\text{m rep}}{\text{hr- watt}} \tag{5.3-1}$$

was converted to

$$\alpha' = 1.67 \text{ x } 10^{11} \frac{\text{neutrons}}{\text{cm}^2\text{- sec- watt}} \tag{5.3-2}$$

The conversion was made by normalizing the number of fission neutrons above 1 MeV per watt in the fission spectrum to rep per watt of thermal power using the Hurst dosimeter response curve to obtain dose rates from neutron flux.

The intensity of the discrete source points used to describe the reactor volume was determined by a computer program using power functions fitted to the gross radial and axial fission distributions. The absolute power yielded by integrating these points was normalized to the peak reactor thermal power of 2511 MWt.

The nonhydrogenous removal cross sections used in the calculations were taken from "Effective Removal Cross Sections for Shielding," G. T. Chapman and C. L. Storrs, Oak Ridge National Laboratory AECD 3978. The values used were:

UO_2 = 0.100 cm^{-1}
Zr = 0.100 cm^{-1}
Fe = 0.168 cm^{-1}

The value for UO_2 was reduced from 0.110 to 0.100 which results in some conservatism. Attenuation in the water region was included in the point kernel. This attenuation is controlled by the nonhydrogenous oxygen removal cross section and the relative density of the water regions. The oxygen removal cross section was taken as Σ_r = 0.033 cm^{-1}. The water densities used for the core and shield regions were consistent with core and coolant flow analysis.

The projected end-of-life fluences include data from the 10 CFR 50 Appendix H metal surveillance capsules removed from the RPVs with neutron fluences representative of approximately 1/4 of RPV life. These projected peak fluences for the Quad Cities RPVs range from 3.5 x 10^{17} to 4.9 x 10^{17} nvt. [5.3-13]

More recently, the NRC issued Regulatory Guide (RG) 1.190, which provides state of the art calculation and measurement procedures that are acceptable to the NRC for determining Reactor Pressure Vessel neutron fluence. Quad Cities RPV fluence has been evaluated using a method in accordance with the recommendations of RG 1.190. Future evaluations of RPV fluence will be completed using a method in accordance with the recommendations of RG 1.190 (as noted in Reference 3).

5.3.2.1 Limit Curves

The reactor vessel is a primary barrier against the release of fission products to the environs. In order to provide assurance that this barrier is maintained at a high degree of integrity, pressure-temperature (P-T) limits have been established for the operating conditions to which the reactor vessel can be subjected. Figures 5.3-4a through 4c present the P-T curves for those operating conditions: Pressure Testing (Curve A), Non-Nuclear Heatup/Cooldown (Curve B), and Core Critical Operation (Curve C). These curves have been established to be in conformance with Appendix G to 10 CFR 50 and Regulatory Guide 1.99, Revision 2, and take into account the change in NDTT as a result of neutron embrittlement. The adjusted reference temperature (ART) of the limiting vessel material is used to account for irradiation effects. In addition, the NRC has approved an exemption to 10CFR10.60(a), "Acceptance criteria for fracture prevention measures for lightwater nuclear power reactors for normal operation." The approved exemption allows the application of ASME Code Case N-588 and ASME Code Case N-640 in the development of the P-T curves described below. [5.3-14]

5.3.2.1.1 Beltline, Nonbeltline, and Closure Flange Regions

Four vessel regions are considered for the development of the P-T curves: 1) the core beltline region, 2) the nonbeltline region (other than the closure flange region and the bottom head region), 3) the closure flange region, and 4) the bottom head region. The beltline region is defined as that region of the reactor vessel that directly surrounds the effective height of the reactor core, and is subject to an RT_{NDT} adjustment to account for radiation embrittlement. The nonbeltline, closure flange, and bottom head regions receive insufficient fluence to necessitate an RT_{NDT}, adjustment. These regions contain components which include the reactor vessel nozzles, closure flanges, top and bottom head plates, control rod drive penetrations, and shell plates that do not directly surround the reactor core. Although the closure flange and bottom head regions are nonbeltline regions, they are treated separately for the development of the P-T curves to address 10 CFR 50 Appendix G requirements.

Boltup Temperature

The limiting initial RT_{NDT} of the main closure flanges, the shell and head materials connecting to these flanges, connecting welds, and the vertical electroslag welds which terminate immediately below the vessel flange, is 23°F. Therefore, the minimum allowable boltup temperature is established as 83°F (RT_{NDT} + 60°F), which includes a 60°F conservatism required by the original ASME Code of Construction.

Curve A— Pressure Testing

As indicated in Curve A of (Figure 5.3-4a) for pressure testing, the minimum metal temperature of the reactor vessel shell is 83°F for reactor pressures less than 312 psig. This 83°F minimum boltup temperature is based on an RT_{NDT} of 23°F for the electroslag weld immediately below the vessel flange and a 60°F conservatism required by the original ASME Code of Construction. The bottom head region limit is established as 68°F, based on moderator temperature assumptions for shutdown margin analyses.

At reactor pressures greater than 312 psig, the minimum vessel metal temperature is established as 113°F. The 113°F minimum temperature is based on a closure flange region RT_{NDT} of 23°F and a 90°F conservatism required by 10 CFR 50 Appendix G. The P-T limits for pressure testing are valid to 54 effective full power years (EFPY).

Figure 5.3-4a is governing for applicable pressure testing with a maximum heatup/cooldown rate of 20°F/hour.

Curve B — Non-nuclear Heatup/Cooldown

Curve B of Figure 5.3-4b applies during heatups with non-nuclear heat (e.g., recirculation pump heat), and during cooldowns when the reactor is not critical (e.g., following a scram). The curve provides the minimum reactor vessel metal temperatures based on the most limiting vessel stress. The maximum heatup/cooldown rate of 100°F/hour is applicable.

Curve C — Core Critical Operation

CurveC, the core critical operation curve shown in Figure 5.3-4c, is generated in accordance with 10 CFR 50 Appendix G which requires core critical P-T limits to be 40°F above any pressure testing or non-nuclear heatup/cooldown limits.

The actual shift in $RT_{\{NDT\}}$ of the vessel material will be established periodically during operation by removing and evaluating, in accordance with ASTM E185-82 and 10 CFR Part 50, Appendix H, irradiated reactor vessel material specimens installed near the inside wall of the reactor vessel in the core area. The irradiated specimens are used in predicting reactor vessel material embrittlement. The operating limit curves of Figures 5.3-4a through 4c, Curves A throughC, shall be adjusted, as required, on the basis of the specimen data and recommendations of Regulatory Guide 1.99, Revision 2.

5.3.2.2 Operating Procedures

Pressure-temperature limit curves in the Technical Specifications are established to the requirements of 10 CFR 50, Appendix G, to assure that brittle fracture of the RPV is prevented. Further description of these limit curves is in Section 5.3.2.1. [5.3-15]

10 CFR 50 Appendix G stipulates, "Pressure tests and leak tests of the reactor vessel that are required by Section XI of the ASME Code must be completed before the core is critical." However, pressure testing of Class 1 piping components following non-welded repair/replacement of non-RPV related Class 1 piping components may be conducted with the reactor core critical when the following conditions are met.

- A valid Class 1 periodic pressure test which meets the requirements of ASME Section XI Table IWB-2500-1, Examination Category B-P exists for the current operating cycle.

- The replacement activities are performed in accordance with a controlled work process.

A system leakage test at operating pressure is performed on the primary system following each removal and replacement of the RPV head. The system is checked for leaks and abnormal conditions which are then corrected before reactor startup. The minimum RPV temperature during the system leakage test is in accordance with Figure 5.3-4a.

The reactor coolant system was given a system hydrostatic test in accordance with ASME Code requirements prior to initial reactor startup. Before pressurization the system was heated to NDTT +60°F. Piping and support hangers were checked while thermal expansion was in progress. Recirculation pump operation was also checked.

5.3.3 Reactor Vessel Integrity

Section 5.3.3.1 summarizes the RPV's purpose and the factors that contribute to RPV integrity.

The following vendors participated in the design and/or fabrication of the Quad Cities Unit 2 RPV: [5.3-16]

A. Babcock and Wilcox was a supplier to GE for:

   1. All fabrication of the bottom head, including the first cylindrical shell course with nozzles and CRD housings, vessel skirt, and internal shroud support.

   2. All remaining shell courses with nozzles attached. No circumferential seams were welded by B&W except the upper shell course to the vessel flange.

   3. The complete closure head.

B. Rotterdam Dockyard Company, Rotterdam, Holland as a subcontractor to B&W, completed the work under A.1. above for B&W. Rotterdam Dockyard Company installed all stub tubes and welded the bottom head to the lower cylindrical shell course.

C. Chicago Bridge and Iron, Memphis, Tennessee completed all the remaining work as a contractor to GE. This work included such items as welding and post weld stress relief of circumferential seams, hydrotesting and post hydro examination.

Babcock and Wilcox, RDM and CB&I's quality assurance organizations were engaged in their respective work. In addition, B&W audited RDM's performance. General Electric Company audited all three vendors' performance, engaging their Quality Control Procured Equipment organization for all work performed within the U.S.A., and using the European based GE Technical Services Company (GETSCO) quality control organization as an agent for the RDM fabrication. Hartford had the responsibility for third party inspection at B&W, RDM and CB&I and signed both the partial data reports and the N-1A form.

Documentation was provided by B&W to direct RDM as to the remaining fabrication and testing operations to be performed. General Electric Company's quality control organization audited this documentation.

Documentation regarding material and status was also provided by B&W to GE for all components shipped to CB&I. After review, GE forwarded this information to CB&I. Records for the B&W and RDM fabrication are located at B&W. Records for the CB&I fabrication are located at CB&I.

5.3.3.1 Design

5.3.3.1.1 General Parameters

The purpose of the RPV is to support and contain the reactor core, the reactor internals, and the reactor core coolant-moderator and to serve as a high integrity barrier against leakage of radioactive materials to the drywell. To achieve these purposes, the RPV was designed using the following general parameters: [5.3-17]

A.	Design pressure	1250 psig
B.	Nominal operating pressure	1005 psig
C.	Base metal	SA-302 Grade B in accordance with Code Case 1339 (RPV SHELL)
D.	Cladding	Weld deposited Type ER308L electrode
E.	Design codes	ASME B & PV Code Sec. III, Class A 1965 Edition and Summer 1954 Addendum

The nominal operating pressure of 1005 psig was based upon economic analyses for boiling water reactors. The design pressure of 1250 psig was determined by an analysis of margins required to provide a reasonable range for maneuvering during operation, with additional allowances to accommodate transients above the operating pressure without causing operation of the safety valves.

The strength required to withstand external and internal loads, while maintaining a high degree of corrosion resistance, dictated the use of a high-strength low alloy steel SA-302, Grade B, with an internal cladding of Type ER308 stainless steel applied by weld overlay. The reactor vessel was designed for a 40-year operational life. During this period, it will not be exposed to more than 10^{19} nvt of neutrons with energies exceeding 1 MeV.

ASME Section III, Class A, pressure vessel design criteria provide assurance that a vessel designed, built, and operated within its design limits has an extremely low probability of failure due to any known failure mechanism.

5.3.3.1.2 Specific Criteria

The specific stress limit criteria of the reactor coolant pressure boundary for loading combinations of operating loads plus maximum earthquake load, and operating loads plus maximum earthquake load plus loads resulting from a design basis accident (DBA), are discussed below. [5.3-18]

For the RPVs:

A. Stress intensities do not exceed ASME Section III (1965 Edition and Summer 1965 Addendum) allowable stress intensity limits for Design Loads.

B. Primary membrane stresses do not exceed 90% of the yield strength of the material.

For the CRD housings and the incore monitor housings, the additional stresses caused by the DBA are very small. The stress limits for the combination of operating loads plus maximum earthquake loads are therefore controlling. For this condition, the stress limit does not exceed 1.5 times the hot allowable stress (1.5 S_m). [5.3-19]

For the jet pump instrumentation penetration seal, the stresses caused by the maximum earthquake and the DBA are very small. The stress limits for operating conditions, as specified in ASME Section III, "Nuclear Vessels," are therefore used as the limits for these accident conditions. [5.3-20]

5.3.3.1.3 Temperature and Pressure Cycles

See Section 3.9 for RPV temperature and pressure cycles information.

5.3.3.1.4 Dynamic Loads

For the loading case consisting of operating loads plus maximum earthquake loads, the stresses in the reactor vessel, the support skirt, and the internal components which support and position the core are within limits which assure essentially elastic behavior. Use of these stress limits results in no gross deformation of parts which could affect control blade insertability. [5.3-21]

5.3.3.2 Materials of Construction

The reactor vessel is a vertical cylindrical pressure vessel as shown in Figure 5.3-5. The RPV shell base plate material is high-strength low alloy steel SA-302, Grade B, in accordance with Code Case 1339. The vessel interior is clad with weld deposited ER308L stainless steel electrode. The main steam outlet lines are from the vessel body, below the reactor vessel flange. [5.3-22]

The reactor vessel was designed and built in accordance with ASME Section III, Class A. General Electric Company specified additional requirements. Records of material properties were developed and are retained for future evaluation of the RPV over its operating lifetime. [5.3-23]

The CRD housings and the incore instrumentation thimbles are welded to the bottom head of the reactor vessel. The incore flux monitor housings are made of Type 304 stainless steel and designed to ASME Section III. [5.3-24]

The RPV is supported by a steel skirt welded to the bottom of the vessel.

A preservice inspection of the components was conducted after site erection to assure that the RPVs were free of gross defects and to provide a reference base for later inspections. The ISI programs provide for continuing inspections during refueling outages. See Section 5.2.4 for discussion of the ISI of the RPVs. [5.3-25]

5.3.3.3 Fabrication Methods

Sections 5.3.1.2 and 5.3.3 provide information on the fabrication methods for the Quad Cities RPVs.

5.3.3.4 Inspection Requirements

As required by the ASME Code, the reactor coolant system was hydrostatically tested prior to initial criticality. Hydrostatic tests are also performed after any modification to the system. The hydrostatic test pressure and testing conditions are detailed in ASME Section XI, and the USAS B 31.1 Code for Pressure Piping. [5.3-26]

The reactor vessel for Unit 1 is stamped with a Code N symbol which signifies that the hydrostatic test and all other required inspection and testing has been satisfactorily completed, final certification has been issued, and all applicable ASME Code requirements have been met.

5.3.3.5 Shipment and Installation

Several shipments of vessel components, as well as shipment of the completed RPV, occurred during the fabrication process. As noted in Section 5.3.3, several vendors in various locations participated in RPV fabrication. General Electric QA assured that all shipments and installation met appropriate regulations and requirements. [5.3-27]

5.3.3.6 Operating Conditions

Section 5.3.2.1 specifies the operating conditions used to show conformance to Regulatory Guide 1.99, Rev. 2.

5.3.3.7 Inservice Surveillance

Section 5.2.4 summarizes the inservice surveillance or ISI for the Quad Cities Units 1 and 2 RPVs.

5.3.4 References

1. "Reactor Handbook," 2nd Edition, Vol. III Part B, Shielding, pages 72 and 80.

2. BWRVIP-86-A: "BWR Vessel and Internals Project, Updated BWR Integrated Surveillance Program (ISP)," Final Report, October 2002.

3. C. F. Lyon letter to J. L. Skolds, "Quad Cites Nuclear Power Station, Units 1 and 2 – Issuance of Amendments Re: Reactor Vessel Specimen Removal Schedule," dated August 28, 2003.

Table 5.3-1

REACTOR VESSEL MATERIAL SURVEILLANCE
WITHDRAWAL SCHEDULE

UNIT 1

HOLD NUMBER	LOCATION	AZIMUTH	REMOVAL YEAR	STATUS
NEUTRON DOSIMETER	MOUNTED SIDE OF PART #7	95°	1974	REMOVED
2	TOP GUIDE	0°	1974	REMOVED
3	WALL	35°	1974	REMOVED
4	TOP GUIDE	90°	1979	REMOVED
5	WALL	65°	STANDBY	
6	TOP GUIDE	180°	1982	REMOVED
7	WALL	95°	STANDBY	
8	WALL	215°	1982	REMOVED
9	WALL	245°	STANDBY	
10	WALL	275°	STANDBY	

UNIT 2

HOLD NUMBER	LOCATION	AZIMUTH	REMOVAL YEAR	STATUS
NEUTRON DOSIMETER	MOUNTED SIDE OF PART #7	95°	1975	-------
12	TOP GUIDE	0°	1975	REMOVED
13	WALL	35°	1975	REMOVED
14	TOP GUIDE	90°	1979	REMOVED
15	WALL	65°	STANDBY	
16	TOP GUIDE	180°	1981	REMOVED
17	WALL	95°	STANDBY	
18	WALL	215°	1981	REMOVED
19	WALL	245°	STANDBY	
20	WALL	275°	STANDBY	

NOTE: 0° IS DUE WEST.

5.4 COMPONENT AND SUBSYSTEM DESIGN

5.4.1 Reactor Recirculation System

Cooling water is forced through the reactor core by the recirculation system which has components both internal and external to the reactor vessel. The system consists of two external loops, together with associated pumps, valves, and piping, plus internal jet pumps and associated flow channeling components. [5.4-1]

5.4.1.1 Design Bases

5.4.1.1.1 General

The recirculation system provides forced convection cooling of the reactor core. The reactor coolant system is designed and shall be maintained in accordance with the code requirements in Section 5 of the UFSAR with allowance for normal degradation pursuant to the applicable surveillance requirements as follows: [5.4-2]

Design pressure, suction	1175 psig at 565°F
Design pressure, discharge	1325 psig at 580°F
Design pressure, discharge to the outlet side of the discharge shutoff valve	1450 psig at 575°F
Recirculation System Design Codes	ASME B & PV Code (ASME Code) Section I USAS B31.1

Design suction pressure is based upon the peak dome pressure that would accompany the limiting transient, plus the static head from the top of the vessel to the recirculation pump suction inlet. [5.4-3]

Design discharge pressure is established at a nominal 150 psi above the suction pressure to accommodate the pressure output of the recirculation pumps.

5.4.1.1.2 Recirculation Pumps

The design codes for the recirculation pumps are provided in Section 3.2.8.2.

At the time of their design, the recirculation pumps were exempt from the requirements of the ASME Code and the USAS Code for Pressure Piping because of their machinery classification. The Hydraulic Institute Standards were the only applicable standards; however, they were more pertinent to the testing and performance of the pump and consequently provided little or no guidance in the areas of casing quality and structural integrity. Therefore, to assure that the pump casings would sustain pressures of at least reactor vessel pressure, the pump casing was designed to meet the requirements of ASME Section III, Class C.

5.4.1.1.3 Jet Pumps

The jet pump assemblies are capable of withstanding, without failure or loss of required functional integrity, the forces, loads, and stresses calculated to be encountered during normal, transient, and accident conditions. [5.4-4]

Each component is able to withstand the combined loadings due to differential pressure and temperature, dead weight, fluid movement, seismic acceleration, and vibration. Allowable stresses defined by ASME Section III are not exceeded for normal operation. Allowance is made for thermal expansion, corrosion, and crud buildup.

The jet pump assemblies form part of the flow channeling components internal to the vessel that serve to limit post-accident decrease in vessel water level. They are designed to be sufficiently leak tight, despite thermal expansion, to permit reflooding of the reactor core to approximately two-thirds of core height following a design basis loss-of-coolant accident (LOCA). Stresses occurring during accident conditions could exceed code limits, and distortion may occur in some parts, but structural integrity would be maintained, particularly with respect to ensuring the core flooding capability. Demonstration that the design bases and performance criteria of the jet pump system have been satisfied is described in GE Topical Report APED 5460, September 1968.

The key components which govern jet pump performance and which experience high fluid velocities are designed to be removable for inspection and/or replacement.

The jet pumps, as components of the reactor, are designed to provide stable, controlled coolant flow rates to the reactor core for forced convection cooling. The core flow supplied by multiple jet pumps operating in parallel is designed to be uniform and predictable under all flow conditions encountered during normal steady-state and transient reactor operation with no flow discontinuities.

The hydraulic characteristics of the jet pumps, in combination with other plant characteristics, nuclear instrumentation, and the reactor protection system, are designed to not deter safe operation of the plant under normal conditions, and to ensure that no fuel damage will result during operational transients caused by reasonably expected single operator errors or equipment malfunctions.

The jet pumps are designed to be capable of performing their intended function subsequent to a hypothetical LOCA, with particular emphasis on their required function during the core reflooding process. They are designed to perform adequately for the duration of plant life in the reactor environment for all design ranges and operating conditions.

5.4.1.1.4 Other Components

Flow-induced vibrations are possible in the recirculation system under abnormal operating conditions. System support structures are designed to withstand flow-induced vibrational forces. [5.4-5]

5.4.1.2 Description

5.4.1.2.1 Reactor Recirculation System

The recirculation system consists of two recirculation pump loops external to the reactor vessel and 20 jet pumps internal to the vessel. Each external loop consists of a variable-speed, motor-driven recirculation pump, two motor-operated gate valves for pump isolation, piping, and required recirculation flow measurement and control devices. The two external recirculation loops supply high pressure flow to piping systems which connect ultimately to the jet pump nozzles. [5.4-6]

Inside the vessel, saturated water rejected from the steam dryer and steam separator is mixed with incoming subcooled feedwater above the core. The resulting subcooled mixture passes down the annulus between the vessel and the core shroud. Approximately 35% of this flow is passed out of the vessel and through the recirculation loops, and becomes the driving flow for the jet pumps. The remaining 65% of the flow enters the jet pump suction inlets and is accelerated by momentum transfer from the driving flow through the jet pump nozzles.

Some static pressure recovery occurs in the mixing section, and the balance occurs in the jet pump diffuser section. Water flows out of the diffuser at sufficient pressure to recirculate through the core.

5.4.1.2.2 Recirculation Pumps, Valves and Piping

The main recirculating pumps are single-stage centrifugal units with mechanical shaft seals, and a seal purge system. Each pump is rated to deliver 570 feet of head at 45,000 gal/min. The pumps are arranged within the drywell to facilitate inspection, maintenance, and/or removal during plant shutdown conditions. The pumps are driven by variable-speed induction motors, which receive electrical power from adjustable speed drives (ASDs). [5.4-7]

The 1A, 1B, and 2B pump internals (shaft, impeller, cover and heat exchanger) were replaced with a fourth generation design rotating assembly. The replacement assembly is designed to improve the reliability and to minimize the potential for shaft and cover cracking due to thermal fatigue as described in GE SIL 459.

The recirculation pumps and motors are located at a lower elevation than the vessel in order to provide adequate net positive suction head (NPSH). An equalizer line connects the two recirculation loops, and consists of a stainless steel pipe with two pairs of manually operated valves. Three of the four valves are closed during normal operation with one valve left open for thermal expansion of the water.

Originally, the recirculation pump discharge valves had bypass lines with 4-inch valves to avoid sudden core flow increases when starting the recirculation system. However, pipe cracking occurred in the bypass loops of both Quad Cities units due to intergranular stress corrosion cracking (IGSCC) in the heat-affected zones of the welds joining the 4-inch bypass piping to the main 28-inch recirculation piping. Therefore, the bypass lines were removed and jogging circuitry was added as a substitute for the bypass function. [5.4-8]

The recirculation lines are provided with pipe restraints to limit pipe motion so that reaction forces associated with a pipe split or circumferential break will not jeopardize containment integrity. These restraints allow unrestricted expansion and contraction of the piping over the design pressure range of 0 – 1260 psig and the design temperature range of 70 – 575°F. Positioning of the restraints assures that strength of the pipe

would be maintained on both sides of a postulated circumferential break and over the entire length of a postulated split pipe. [5.4-9]

5.4.1.2.3 Jet Pumps

The jet pumps, which have no moving parts, are constructed of Type 304 austenitic stainless steel. They are located inside the reactor vessel between the core shroud and the vessel wall. See Figure 5.4-1. Each pair of jet pumps receives driving flow from a single riser pipe. Each riser has a dedicated vessel penetration and receives flow from one of the two recirculation inlet manifolds. [5.4-10]

The jet pump consists of a diffuser, a throat section, and a nozzle section, as illustrated in Figure 5.4-2. The jet pump diffuser is a gradual conical section terminating in a straight cylindrical section at the lower end which is welded to the core shroud support. The throat section is a straight section of tubing with a short diffuser entrance section at the lower end, which is clamped to the nozzle section. The throat and nozzle sections are attached to the riser and diffuser with brackets which provide structural rigidity, yet permit differential expansion between the carbon steel vessel and the stainless steel jet pump. The overall height from the top of the inlet nozzle assembly to the diffuser discharge is 18 feet 7 inches, and each diffuser has an outside diameter of 20-3/4 inches. Replacement of the throat and nozzle section of a jet pump is possible. Additional descriptive information for the jet pumps is provided in Section 3.9.5.1.

The principle of operation of the jet pump is the conversion of momentum to pressure. The fluid emerging from the nozzle, called the driving fluid, has high velocity and high momentum, but low static pressure. The low-energy downcomer fluid is drawn into the throat section by the pressure difference between the downcomer fluid and the driving fluid. [5.4-11]

In the throat, the two fluid streams combine and undergo momentum transfer. During this process there is some static pressure recovery. However, the main function of the mixing chamber is to provide complete combination of the high-and-low energy streams so that a single high velocity stream enters the diffuser. For optimum operation, the velocity profile at the exit of the throat should be as flat as possible; i.e., the boundary layer entering the diffuser should be as thin as possible. This flat velocity profile ensures maximum performance of the diffuser. In the diffuser, the relatively high velocity of the combined streams is converted to high static pressure. The resulting exit flow has the pressure required to provide the necessary recirculation flow through the core.

5.4.1.3 Performance Evaluation

5.4.1.3.1 System Design Test Data

Several series of tests were conducted to study the performance of jet pumps under simulated reactor conditions. The tests were designed to verify analytic performance predictions and to supply additional design information regarding the effects of several jet pump variables. The areas investigated included: mixing chamber or "throat" length, nozzle-to-throat spacing, nozzle eccentricity, nozzle size and configuration, simulated

nozzle erosion, and diffuser configuration. Tests were conducted with quarter-scale models and actual full-scale pumps. The quarter-scale pumps were tested using groups of four. The actual full-scale pumps were tested individually. The tests covered pressure, temperature, and subcooling ranges expected during reactor operation. This program validated the theory and identified particular information required to verify expected jet pump performance under specific operating conditions. In addition, the startup test program for Dresden Unit 2, which preceded that of Quad Cities, provided comprehensive data for both the steady-state and dynamic performance of a recirculation system essentially identical to Quad Cities. A complete history of the jet pump testing is given in APED 5460. [5.4-12]

5.4.1.3.1.1 Performance Efficiency Tests

Early in the program it was verified that the maximum efficiency to be expected from any jet pump is a function of the design flow ratio. This ratio, designated M, is defined as the ratio of the driven mass flow (drawn from the downcomer annulus) to the driving mass flow through the nozzle; hence the efficiency is a function of the drive nozzle size. A jet pump was tested with nozzles designed for three flow ratios (M = 1.00, 1.25 and 2.50). Figures 5.4-3 and 5.4-4 show the results for the nozzles designed to provide maximum efficiency at an M ratio of 2.50. On Figure 5.4-3, the upper curve shows the calculated performance of an idealized jet pump with only mixing losses (no friction losses). This represents a maximum attainable efficiency for a simple jet pump. The second curve shows the calculated jet pump performance based on initial estimates of friction losses. The third curve shows the observed performance. Figure 5.4-4 shows the same data plotted as head ratio N, versus flow ratio M, where head ratio is defined as the ratio of the specific energy increase of the downcomer stream to the specific energy decrease of the driving stream. [5.4-13]

From data generated with the three different nozzles, the following values of maximum efficiency were observed during the cold water phase of tests:

Design Flow Ratio (M)	Efficiency (%)
1.00	38.5
1.25	36.5
2.50	33.5

These data are plotted on Figures 5.4-5 and 5.4-6, which show calculated curves of the peak head ratio $N_{\{p\}}$, and flow ratio at peak efficiency $M_{\{p\}}$, as functions of the area ratio R.

The solid curves are calculated values and are the same curves as those shown on Figures 5.4-3 and 5.4-4 for "reasonable friction." The dash curves through the data points show the experimental performance.

The performance tests provided information regarding operation of the pumping system in the reactor both directly and indirectly. Direct study of the effect of fluid temperature variations and two-phase flow (carryunder), among other effects, was possible through Moss Landing Test simulation of these conditions. In addition, application of the performance data in various analytic models has made it possible to accurately evaluate the performance of the jet pump system under both normal and abnormal modes of reactor operation.

The final jet pump system design characteristics, developed from analyses and application of these test data, are summarized in Table 5.4-1.

A typical jet pump head-capacity characteristic curve for normal operating conditions is shown in Figure 5.4-7.

5.4.1.3.1.2 Cavitation Tests

Cavitation, caused by carryunder or insufficient subcooling, was also tested. Tests were performed at off-rated conditions in order to subject the jet pump to cavitation. This was done by reducing the subcooling and injecting superheated steam into the recirculation flow upstream of the jet pump suction inlet. These tests showed that when the jet pumps were caused to cavitate, the efficiency was decreased, but in spite of the presence of cavitation, further increases in suction flow rate were still possible. [5.4-14]

One test was designed to describe operation with superheated steam injected into the suction inlet. The objective was to determine jet pump performance with simulated carryunder introduced at the rate of 0.06% by weight (maximum available from the test facility). In addition, this test was performed with 0.0 subcooling. The resulting efficiency, 31.7% at M = 1.53, does not represent a serious degradation in jet pump performance. Even though the operating conditions were particularly adverse, the jet pump was still capable of achieving a discharge head of 16.3 psi at full rated flow.
Data collected from this and other tests show the influence of subcooling on total flow. The data indicate that subcooling must be reduced to approximately 3 — 4 Btu/lb before cavitation begins to affect flow rate. This also reaffirmed the observation that the loss of subcooling does not bring about an abrupt loss in pumping capability. This cavitation threshold may be compared to normally available subcooling of 20 Btu/lb.

5.4.1.3.1.3 Erosion-Corrosion Tests

The steel used in the jet pump assemblies has satisfactory corrosion resisting properties which are adequate for the 40-year design life. However, because of the high velocities existing in some regions of the jet pump, the potential problem of erosion-corrosion was investigated. The highest velocities occurring in the jet pump assembly are those at the nozzle, where velocities may be as high as 180 ft/s. Erosion-corrosion tests have been performed on a small-scale nozzle configuration made from Type 304 stainless steel. The test consisted of subjecting the nozzle to three consecutive 1000-hour periods of high-temperature, high-pressure flowing water simulating reactor conditions. Specifically, the driving pressure was approximately 200 psi with the temperature ranging between 240 - 415°F; the nozzle velocity was maintained at 460 — 470 ft/s. At the conclusion of each 1000-hour period, the internal diameter experiencing the high velocity flow was measured to determine the resulting enlargement from which the erosion rate was computed. The results of this test, and earlier tests conducted in substantially the same manner, indicated that the selected design value used in the jet pump design of 0.001 in./yr is conservative. Using this design value, the nozzle inside diameter (ID) will increase by no more than 0.08 inches during the 40-year design lifetime of the BWR.

The effect of a slightly increased nozzle ID is to force the system to run at a slightly lower M ratio. If the nozzle size originally were equal to or greater than the optimum nozzle size required for maximum jet pump efficiency, then an increased recirculation pump

speed would correspond to design core flow. The design of the recirculation pumps includes sufficient margin to accommodate this lifetime effect. If the nozzle size originally were less than the optimum nozzle size, then a decreased recirculation pump speed would correspond to design core flow.

Cavitation could accelerate erosion and corrosion in local areas. Cavitation is not expected to occur, however, because sufficient pressure or subcooling is available to suppress vaporization of the flowing liquid. Taking a typical example, the pressure in the 180° elbow (inlet subassembly) of the jet pump is more than 195 psi higher than the equivalent saturation pressure. At the nozzle, where the highest velocity occurs, the pressure is still more than 120 psi above the saturation pressure. In the throat region (mixer), where the second highest velocity occurs, approximately 20.4 Btu/lb subcooling is available for suppression of cavitation.

Other conditions which tend to promote erosion and corrosion are similarly absent from the jet pump-reactor system:

A. Measurements taken from a full-size prototype jet pump tested under reactor operating conditions indicated no high surface vibration in jet pump components;

B. There are no points of direct impingement of coolant flow on jet pump subassemblies (the inlet subassembly contains an internal vane, installed to reduce flow losses, which further reduces the possibility of impingement in the one subassembly where impingement could be considered most likely to occur);

C. The jet pump-reactor system contains no contaminants or particulate matter that could intensify erosion and corrosion; and

D. The mating surfaces of the throat-diffuser joint contain Stellite-6 to minimize erosion (wire drawing) and corrosion due to leakage.

5.4.1.3.1.4 Stability Tests

Prior to the construction of Quad Cities station, loop tests of a jet pump recirculation system were conducted at the Moss Landing steam plant, where reactor hydraulic conditions were simulated. The tests assessed the operational hydraulic stability of the system as constructed, and provided data for evaluating the behavior of full-scale jet pump recirculation systems installed in reactors. [5.4-15]

5.4.1.3.2 Normal Operation

5.4.1.3.2.1 Reactor Recirculation Flow Monitoring

The reactor core flow rate is monitored by control room readout of the total discharge flow from each group of jet pumps which are driven by an individual external drive loop. Flows are measured by measuring the differential pressure between the diffuser throat and the lower plenum for each jet pump unit. These measurements are provided on the flat panel display at panel 901(2)-38 in the auxiliary electrical equipment room and on the Operator Station in the Main Control Room. For flow measurement purposes, the 20 jet pumps

are divided into four groups of five. A typical group is shown in Figure 5.4-8. In each group, one jet pump has a diffuser with two static pressure taps. The remaining four units have only one pressure tap. The double-tap units were calibrated prior to installation to determine the relationship between flow and differential pressure. This information is used to perform in-reactor calibration of the single-tap pressure difference for all 20 jet pumps. The procedure can be summarized as follows: [5.4-16]

A. Read the static pressure difference from each double tapped unit.

B. Relate this pressure difference to flow rate by using the calibration information collected before installation of the double tapped units.

C. Knowing the flow through each double tapped unit, determine the relationship between flow and the single tap to lower plenum pressure difference (see Figure 5.4-8).

D. Apply the single tap to lower plenum relationship found in the four calibrated double tap units to the remaining 16 single tap units.

After this calibration procedure has been completed, the total core flow is measured by electrically analyzing the signals from the single tap to lower plenum pressure transducers on all twenty jet pumps. The resulting total core flow rate output signal is displayed on the reactor control board.

Analysis shows that plant operation with loss of flow indication in up to three jet pumps is acceptable, as long as each jet pump with failed flow indication is on a separate riser and no more than one double-tap (calibrated) jet pump per loop is affected. This assures that adequate information is available to verify the core is in an analyzed condition during single-loop and double-loop operation. [5.4-17]

Additional considerations apply to operation with failed jet pump differential pressure taps:

A. Core Flow Accuracy — by utilizing the companion jet pump to derive flow through the jet pump with failed flow indication, an accurate value of core flow can still be calculated. In addition, the core flow uncertainty is maintained within the limits assumed for the derivation of the minimum critical power ratio (MCPR) Safety Limit.

B. Jet Pump Integrity — jet pump integrity must be maintained to ensure the core can be reflooded following a design basis accident. Requiring at least one jet pump to have operable flow indication on each riser assures that surveillance testing will detect a jet pump failure. The Technical Specifications (Bases) detail the restrictions on the number, and combination, of jet pumps which can have inoperable flow indication.

C. Emergency Core Cooling System Performance — with up to three jet pump instrument lines severed, the maximum leakage would be 3 gal/min. This amounts to much less than 1% of the total ECCS flow. Sensitivity studies performed by GE have shown leakage of this magnitude has no effect on ECCS performance.

5.4.1.3.2.2 External Recirculation Loop Flow Monitoring

The flow through the two external recirculation loops is continuously monitored by flow elements. The differential pressure signals from these elements are converted to flow signals and transmitted to the control room. In the control room, flow through each loop is both indicated and recorded. The flow signals are used as inputs to the digital Reactor Recirculation Flow Control System (RRCS) and APRM flow units. An indirect measure of external loop flow is available from the recirculation pump motor amperage, which is indicated in the control room. As might be expected, the core recirculation flow is essentially proportional to the total external loop flow rate during normal "flow control" manipulation of plant power output. [5.4-18]

The relationship between recirculation pump speed and flow rate in the associated recirculation loop reflects the hydraulic characteristics of the recirculation loop flow path. Since the inter-loop equalizer line is closed during operation, and for pump speeds greater than 60%, this relationship can serve as a reliable indicator for equipment problems, such as a displaced or damaged jet pump, that would cause hydraulic aberrations. [5.4-19]

5.4.1.3.2.3 Recirculation Pump Performance

The recirculation pumps are approximately 60 feet below normal reactor water level. This static head alone provides sufficient NPSH to prevent cavitation in the drive pumps at low speeds. At the higher speeds of normal operation, cavitation is prevented by the subcooling of the pump suction fluid. Returning feedwater (steam condensate) combines with the saturated fluid leaving the separators and the resulting mixture has more than enough subcooling to prevent cavitation. During initial operation with no feedwater flow, the recirculation pumps are started and brought up to minimum speed until operational pressure and steam production is established. As soon as steam is flowing, the downcomer flow is subcooled by the feedwater flow, and the recirculation pump speed may be increased to rated speed.

During normal operation, both the recirculation pumps and the jet pumps require approximately 35 psi (110 feet) of NPSH. At rated conditions the NPSH available (see Figure 5.4-9) is at least twice this amount.

Reduction of plant output by flow control maneuvering will cause a slight increase in this NPSH margin. The only time NPSH requirements approach the available NPSH is during high core flow and low thermal power operation. There is, however, no incentive to operate under high-flow, low-power conditions for sustained periods of time. Operation in extreme conditions is prevented procedurally as well as by interlocks.

When the recirculation pumps are started, the recirculation pump discharge valve is jogged open to slowly introduce water into the core. The jogging circuitry in the recirculation pump discharge valve opening logic provides for automatic 1-second and 1/2-second opening jogs which prevent sudden increases in reactor coolant flow when starting the recirculation system. During operation, a recirculation pump trips if its discharge valve is closed more than 10% from the full open position. [5.4-20]

During idle loop startup, asymmetric speed operation of the recirculation pumps induces levels of jet pump riser vibration that are higher than normal. A limitation of less than or equal to 45% of rated pump speed for the operating recirculation pump prior to the start of the idle recirculation pump ensures that the recirculation pump speed mismatch requirements are maintained. The pump speed of the operating recirculation pump is determined within limits (</= 45% of rated) within 15 minutes prior to the startup of an idle pump. [5.4-20a]

Single loop operation (SLO) is a mode of operation during which the reactor may operate and produce power with only one recirculation loop in service. One of the requirements for SLO is that the recirculation pump in the idle loop is electrically prohibited from starting except to permit testing in preparation for return to service. Automatic closure of the discharge valve in the idle loop by the LPCI loop selection logic prevents the loss of low pressure coolant injection (LPCI) flow [5.4-21]

through the idle recirculation pump into the downcomer during a postulated accident. Operation in the SLO mode is permissible with the suction and discharge valve in the idle loop open provided the LPCI loop select logic is operable. See Section 15.3.6 for discussion of transients during SLO.

5.4.1.3.2.4 Recirculation Loop Integrity Monitoring

Individual sensors monitor the differential pressure between the jet pump inlets of the two recirculation loops. Excessive pressure differential, which is an indication of a failed recirculation line, is used to properly sequence the operation of the LPCI subsystem described in Section 6.3.2.2. [5.4-22]

Jet pump integrity and operability is checked regularly by monitoring recirculation pump speed, recirculation loop flows, and individual jet pump flows as necessary. Jet pump integrity is required to demonstrate that the core can be reflooded to two-thirds core height following a large recirculation line break LOCA. [5.4-23]

5.4.1.3.2.5 Digital Control System

All jet pump and core flow related sensors are inputs to the Reactor Recirculation Control System (RRCS) and are processed by a digital control system (DCS). The digital control system provides the analog signal filtering, conversions, and summing for determining total core flow. The main function of the DCS is to provide the control interface for recirculation pump speed to control recirculation loop drive flow. The DCS provides the control logic and interfaces with the FWLC and process computer as well.

A RRCS DCS Operator Station is common with the Feedwater Level Control System and is provided in the Main Control Room. The RRCS DCS displays jet pump and core flow related indications, and provides operator interface control display dialog for ASD operation.

Operation of the RRCS DCS is described further in Section 7.7.3.1.1.

5.4.1.3.3 Equipment Malfunctions and System Transients

In addition to the normal operation evaluation, the Moss Landing test data was used to analyze abnormal operating conditions. The most significant equipment malfunctions and system transients from that analysis are discussed in the following sections: [5.4-24]

Malfunction/Transient	UFSAR Section [5.4-25]
Jet Pump Malfunction	15.3.5
Flow Control Malfunctions	
— Zero Speed Demand	15.3.2
— Full Speed Demand	15.4.5
Pump Trips	
— Trip of Both Drive Motors	15.3.1.1 (historical – M-G set related)
— One Drive Motor Trip	15.3.1.2 (historical – M-G set related)
— One Pump Motor Trip	15.3.1.3
Recirculation Pump Seizure	15.3.3
Cold Recirculation Loop Startup	15.4.4
Inadvertent Injection of HPCI	15.5.1

Section 15.8 describes anticipated transients without scram (ATWS) which cause both recirculation pumps to trip.

5.4.1.4 Tests and Inspections

5.4.1.4.1 Jet Pumps

Extensive testing to verify and determine the jet pump performance were performed during the preoperational test program. During reactor operation the system is in continuous use, and the instrumentation provided assures adequate monitoring of system performance.

Visual inspection of the uppermost jet pump components, including holddown beams, is performed each refueling outage using underwater television equipment. [5.4-26]

5.4.1.4.2 Piping

The recirculation piping can be inspected by removing the thermal insulation. Periodic random visual inspection of areas of highest stress concentration, or areas of more importance from a leak standpoint, are made during regularly scheduled refueling outages. [5.4-27]

The criteria for inspection of recirculation piping are based on the probability of a defect occurring or enlarging at a given location. These include areas of known stress concentration and locations where cyclic strain or thermal stress might occur. A statistically significant portion of the system is inspected. The type of inspection at each location is dependent on the type and location of defects expected. Direct visual examination is utilized, wherever possible, since it is sensitive, fast, and positive. Magnetic particle and liquid penetrant inspection are employed wherever practical, and added sensitivity is required. Ultrasonic testing (UT) and radiography are considered where defects can occur on concealed surfaces.

A critical defect on the order of feet in length is required to cause a running crack in the materials used to construct the primary system. Considering the wall thicknesses of these materials, a crack cannot grow to such a length before it penetrates the wall and causes a leak. The inspection procedures are geared to the detection of leaks and small defects. While these will have a minor effect on plant safety, they may affect plant availability; thus, there is a high incentive to detect them. Experience with operating plants to date shows that the incidence of leaks is very small and the probability of finding them at an early period is very high. Thus, a small defect will not grow to a critical length prior to being detected.

Special inspection requirements have been implemented as a result of industry experience with IGSCC of certain recirculation piping. These requirements are discussed in Section 5.2.3.5.

5.4.1.4.3 Other Components

The following requirements were applied to the procurement of the valves and pumps for Quad Cities Units 1 and 2 recirculation system. In addition, the acceptance standards for

the recirculation system components conform to the draft AEC nondestructive testing standards in effect at the time "Table A". [5.4-28]

5.4.1.4.3.1 Valves

5.4.1.4.3.1.1 Nondestructive Testing

All nondestructive testing was specified to be in accordance with the specified paragraphs of ASME Section III, as listed in the following. [5.4-29]

5.4.1.4.3.1.1.1 Castings

All pressure-containing castings (including discs if cast) were specified to be radiographed and liquid penetrant examined in accordance with, and to meet the acceptance requirements of, Paragraph N323. The technique for radiography was specified to be in accordance with Paragraphs N624.2 through N624.7. Final radiograph and liquid penetrant examination were specified to be performed after at least one solution heat treatment.

5.4.1.4.3.1.1.2 Forgings

All pressure-containing forgings over 4 inches in thickness were specified to be UT and liquid penetrant examined in accordance with, and to meet the acceptance requirements of, Paragraph N322.

5.4.1.4.3.1.1.3 Welds

All pressure-containing butt welds were specified to be radiographed in accordance with, and to meet the acceptance requirements of, Paragraph N624.

The final surface of all welds was specified to be liquid penetrant examined in accordance with, and to meet the acceptance requirements of, Paragraph N627.

5.4.1.4.3.1.1.4 Bolting

All bolting was specified to be either liquid penetrant examined or wet magnetic particle examined in accordance with, and to meet the requirements of Paragraph N325.

5.4.1.4.3.1.1.5 Valve Stems

All valve stems were specified to be UT and liquid penetrant examined in accordance with, and to meet the acceptance requirements, of Paragraph N322.

5.4.1.4.3.2 Pumps

5.4.1.4.3.2.1 Nondestructive Testing

All nondestructive testing was specified to be in accordance with the specified paragraphs of ASME Section III, as in the following.

5.4.1.4.3.2.1.1 Castings

Castings were specified to be radiographed in accordance with paragraph N323.1 including the Code Committee action which limited all defects to severity level 2, and which did not permit defects of Type D, E, F, or G for E-71-34, or of Type D or E for E-186-65T and E-280-65T. The technique for radiography was specified to be in accordance with paragraphs N624.2 through N624.7. Final radiography and liquid penetrant examination of pressure containing castings was specified to be performed after at least one solution heat treatment.

5.4.1.4.3.2.1.2 Forgings

All pressure-containing forgings over 4 inches in thickness, including pump shafts, were specified to be UT and liquid penetrant examined in accordance with Paragraph N322.

5.4.1.4.3.2.1.3 Welds

All pressure-containing butt welds were specified to be radiographed in accordance with Paragraph N624.

The final surface of all welds was specified to be liquid penetrant examined in accordance with Paragraph N627.

5.4.1.4.3.2.1.4 Bolting

All bolting was specified to be either liquid penetrant examined or wet magnetic particle examined in accordance with Paragraph N325.

5.4.2 Steam Generators

This section is not applicable to Quad Cities Station.

5.4.3 Hydrogen Water Chemistry System

The purpose of the hydrogen water chemistry (HWC) system is to inject hydrogen into the reactor coolant to limit the dissolved oxygen concentration. Suppression of dissolved oxygen, coupled with high purity reactor coolant, reduces the susceptibility of reactor piping and materials to IGSCC. A hydrogen injection system injects hydrogen into the condensate pump discharge line through gas saver lance assemblies. An air/oxygen injection system injects air or oxygen into the off-gas system to ensure that the excess hydrogen in the off-gas system is safely recombined. The air/oxygen addition prevents the hydrogen from reaching combustible concentrations within the off-gas system. The air/oxygen is injected upstream of the first stage of the steam jet air ejector. [5.4-30]

Information on the reactor coolant piping is contained in Section 3.9. Intergranular stress corrosion cracking is discussed in Section 5.2.3.5.

5.4.3.1 Hydrogen Injection System

5.4.3.1.1 Design Basis

The hydrogen injection system is designed to be capable of attaining and maintaining water chemistry in the reactor coolant to mitigate the potential for IGSCC. [5.4-31]

The hydrogen injection system does not support safe shutdown or perform any reactor safety function nor does it mitigate any accidents or transients. The HWC system is designed to respond to, or trip off line, when conditions exist that could result in hydrogen concentration achieving explosive levels. The trip setpoints are set at proven parameter levels that will indicate adverse conditions while avoiding nuisance tripping. The station Technical Requirements Manual provides the requirements for explosive gas monitoring and control for the off-gas system.

5.4.3.1.2 Description

The hydrogen supply site for the hydrogen injection system is located 1500 feet from the nearest safety-related structure, the Unit 1 and Unit 2 control room. It is surrounded by a lighted security fence, and truck barrier posts are installed at the fence perimeter to protect it from mobile equipment. The hydrogen is stored as a liquid in a cryogenic storage tank and as a high pressure gas in transportable tube trailers which are used as a back-up supply of hydrogen gas. [5.4-32]

Liquid hydrogen is supplied from a 20,000-gallon cryogenic storage tank and transferred through a cryogenic pump to a vaporization station. The vaporization station consists of a parallel array of vaporizers, with either of two vaporizer legs being capable of supplying the maximum required hydrogen flow rate for each unit. The vaporizers provide high pressure gas to a permanent tube rack. The tube rack provides gaseous hydrogen to the injection system through a pressure control station. Downstream of the

pressure control station is an isolation valve, a pressure regulator, an excess flow check valve, and a nitrogen purge connection.

Tube trailers are used as a reserve supply of hydrogen gas. The hydrogen gas flows from the tube trailers through a close-coupled shutoff valve and a parallel array of pressure-reducing regulators. The trailers are connected, via a flexible pigtail, to a discharge stanchion. The discharge stanchion consists of a shutoff valve, check valve, bleed valve, and a grounding strap. Hydrogen gas piping runs from the discharge stanchion to an isolation valve, and then connects to the pressure control station upstream of the excess flow check valve.

A branch line leads to the Unit 1 and Unit 2 generator hydrogen control cabinet while the main hydrogen supply line continues to an additional excess flow check valve, a nitrogen purge connection, and a manual isolation valve. This line then branches to the Unit 1 or Unit 2 side of the turbine building. Inside the building, each branch line leads to a solenoid-operated isolation valve which closes upon an area hydrogen concentration high signal. The line then leads to a parallel array of flow control stations and then to a purge line flame arrestor. It then branches into four lines leading to the individual condensate pumps. Each of these lines passes through a manual isolation valve, a solenoid-operated isolation valve (which closes if the associated condensate pump is not activated or if the HWC system is tripped), a check valve, and a second manual isolation valve before it connects to the condensate pump discharge line.

Unit 1 HWC system control circuitry has a bypass switch installed. This switch will allow the HWC system trip (Hydrogen Area Trouble) and closure of the hydrogen injection solenoid valves to be bypassed during abnormal operating conditions involving maintenance and testing.

5.4.3.2 Air/Oxygen Injection System

5.4.3.2.1 Design Basis

The air and oxygen injection system is designed to inject a sufficient amount of oxygen into the off-gas system to ensure that the excess hydrogen in the off-gas stream is recombined (air is approximately 20% oxygen). This prevents the hydrogen concentration from reaching the combustibility limit of hydrogen in air. [5.4-33]

The air/oxygen injection system is not credited to support safe shutdown or to perform any reactor safety function.

5.4.3.2.2 Description

The oxygen supply site is located 1000 feet from the control room and 500 feet from the hydrogen supply site. It is surrounded by a lighted security fence, and truck barriers are installed at the fence perimeter to protect it from mobile equipment.

Liquid oxygen is supplied from an 11,000-gallon cryogenic storage tank. The oxygen flows from the tank to an oxygen vaporization station consisting of two pairs of air vaporizers installed in parallel. Each pair of vaporizers is capable of supplying the maximum required oxygen flow rate for each unit. Downstream of the vaporizer station, the oxygen line leads to a parallel array of pressure-reducing regulators, a temperature shutoff valve, and then to an excess flow check valve. The oxygen line then proceeds underground, alongside the hydrogen supply line, to a point near the west wall of the Unit 1 turbine building, where it continues aboveground. The pipe then branches to the Unit 1 or Unit 2 side of the turbine building. Each branch enters the turbine building and leads to a flow control station. The oxygen and air piping is finally connected to the off-gas system piping upstream of the first stage of the steam jet air ejector.

The oxygen piping also has a connection to either the "A" or "B" off-gas pre-heater, upstream of the pre-heater (Unit 2 ONLY).

The air injection line draws building air through a similar flow control station to the off-gas system near the oxygen injection point. This line provides air (at approximately 20% oxygen) as one of the means of controlling oxygen concentration in the off-gas system.

Air (at approximately 20% oxygen) or pure oxygen or both is used at the discretion of the Operations personnel in the control room.

Redundant oxygen analyzers, located downstream of the off-gas recombiners, allow monitoring of the residual oxygen concentrations in the off-gas stream.

5.4.3.3 Condensate Oxygen Injection System

5.4.3.3.1 Design Bases

The condensate oxygen injection system is designed to inject oxygen into the condensate system to maintain a minimum level of dissolved oxygen in the feedwater system. This prevents increased corrosion in the feedwater piping and components which has been shown to occur under extremely low oxygen conditions. [5.4-34]

5.4.3.3.2 System Description

The condensate oxygen injection system is part of hydrogen water chemistry system. The system consists of a cylinder rack for cylinders of compressed oxygen, a pressure regulator, shutoff valves, pressure gauges, metering valve, flow meter, and associated tubing and supports. The oxygen is injected via the turbine building sample panel recovery header which connects to the condensate pump suction header below the hotwell. The installation is located on the turbine building mezzanine (el. 611'6") in the area of the turbine building sample panels.

5.4.3.4 Control and Instrumentation

The instrumentation and controls for the HWC system include all sensing elements, equipment and valve operating hand switches, equipment and valve status switches, process information instruments, and Programmable Logic Controllers (PLCs) necessary to ensure safe and reliable operation. [5.4-35]

All flow control valves for injection are designed to fail closed on loss of air or control signal. A list of the designed HWC trips is provided in Table 5.4-2.

All control room instrumentation and controls for the HWC system are located on a seismically designed control panel. This panel also contains annunciators for local panel trouble alarms.

The HWC system supplies hydrogen to each unit's condensate system. The hydrogen addition rate can be adjusted either automatically, with the addition rate based on steam flow, or manually. Air and/or oxygen flow to the off-gas system is based upon the hydrogen and oxygen concentration downstream of the off-gas recombiners. Additionally, the system is designed so the air/oxygen injection system remains operating after hydrogen injection has been terminated so that all free hydrogen in the condensate and off-gas will be recombined.

5.4.3.5 Performance Analysis

The sample line for HWC performance analysis on Units 1 and 2 originate at the Reactor Water Cleanup system. Electrochemical Corrosion Potential (ECP) measurements and noble metal durability coupons provide data for measuring performance of HWC and Noble Metals. The Noble Metal Chemical Addition (NMCA) is discussed in section 5.2.3.2.1.1.

5.4.3.6 Inspection and Testing

The functional operability of the HWC system was initially tested at the time of system installation.

The plant preventive maintenance program includes inspections of the HWC system. Retesting requirements for the system are based upon manufacturer's recommendations, and consider extended HWC system shutdown periods and other factors not consistent with normal system operation. A retest of the hydrogen supply system integrity is performed following modifications to the hydrogen piping which may affect the pressure boundary of the system.

5.4.4 Main Steam Line Flow Restrictors

5.4.4.1 Design Bases

Main steam line flow restrictors are an engineered safety feature. [5.4-36]

The purpose of the main steam line flow restrictors is to limit the quantity of steam which would be discharged from the reactor vessel in the event of a steam line break. To achieve this purpose, the design basis of the steam line flow restrictors is to limit steam flow through a ruptured steam line to 145% of rated steam line flow.

Limiting the flow through a severed steam line would:

A. Limit the loss of coolant inventory from the reactor vessel;

B. Minimize the amount of moisture carryover that would occur prior to closure of the main steam isolation valves (MSIVs);

C. Minimize the probability of forming high velocity water slugs in the steam line.

5.4.4.2 System Description

A main steam line flow restrictor is a simple venturi welded into each steam line between the reactor vessel and the first MSIV. The restrictors have no moving parts and are located as close to the reactor vessel as practical. The restrictors also serve as flow elements to provide flow monitoring. They were designed and fabricated in accordance with USAS B31.1. [5.4-37]

5.4.4.3 Design Evaluation

The accident for which the main steam line flow restrictors are evaluated is a postulated complete severance of a main steam line outside the primary containment. The rapid depressurization that would accompany this event would result in a steam-water mixture leaving the reactor vessel. The steam-water flow would choke in the decreased area of the flow restrictor by a two-phase mechanism similar to critical flow in gas dynamics. This would limit the fluid flow rate, and therefore the rate of reactor coolant blowdown, sufficiently to permit closure of the MSIVs before the coolant level in the reactor vessel fell below the top of the reactor core (see Section 15.6). [5.4-38]

The restrictors are capable of withstanding the forces produced by saturated steam with a 1300 psi driving head. Downstream of the restrictors the velocities would be reduced, and pressure surges would be of no consequence.

5.4.4.4 Tests and Inspections

Initial differential pressure measurements were obtained over the range of flows expected. These measurements are repeated periodically. Steam flow readings are monitored during reactor operation in the control room. [5.4-39]

5.4.5 Main Steam Line Isolation System

Information on the main steam line isolation system is contained in Sections 6.2.4 and 7.3.2.

5.4.6 Reactor Core Isolation Cooling System

5.4.6.1 Design Bases

The purpose of the reactor core isolation cooling (RCIC) system is to provide cooling water to the reactor core in the event of a postulated isolation of the reactor from the main condenser with a loss of reactor feedwater. To achieve this purpose, the RCIC system is designed to supply 400 gal/min of makeup water to the reactor core over a reactor pressure range of 1135 — 165 psia. [5.4-40]

All components necessary for initiating operation of the RCIC system are completely independent of auxiliary ac power, plant service air, and external cooling water. The system requires only dc power from the station battery to operate the valves.

5.4.6.2 Description

The RCIC system consists of a steam turbine-driven pump unit and associated valves and piping capable of delivering makeup water to the reactor vessel. A summary of design requirements for the turbine-driven pump unit is shown on Table 5.4-3. The RCIC system is shown in P&IDs M-50 for Unit 1 and M-82 for Unit 2. [5.4-41]

The RCIC turbine-driven pump units are located in the Unit 1 north and Unit 2 south core spray equipment rooms. These rooms are provided with coolers to maintain compartment temperatures below the qualification temperatures of the components required for safe shutdown of the plant. The individual component qualification temperatures are identified in the station equipment qualification binders. The room coolers are water-cooled heat exchanger fan units and are supplied by emergency buses. [5.4-42]

The turbine-driven pump supplies demineralized makeup water to the reactor from the contaminated condensate storage tank (CCST). An alternate source of makeup water is the suppression pool. The pump discharge is delivered to the reactor vessel through a connection to the feedwater line, and is distributed within the vessel through the feedwater sparger. Cooling water for the RCIC turbine lubrication oil cooler and gland seal condenser is supplied from the discharge of the RCIC pump.

The steam supply to the RCIC turbine comes from the reactor vessel, and the spent steam from the turbine exhaust is discharged to the suppression pool.

The RCIC system is automatically initiated upon receipt of a reactor vessel low-low water level signal utilizing level sensors and outputs arranged in a one-out-of-two taken twice logic. The system will start automatically and deliver design flow within 30 seconds. A minimum flow bypass line to the suppression pool is provided for pump protection (note: the automatic minimum flow protection is only enabled when a reactor vessel low-low water level signal is present). Provisions are available to manually start and control the RCIC system from the control room and to start and control the RCIC system from the appropriate equipment room. The system delivers full flow until the reactor high water level is reached at which time the RCIC system automatically shuts down. The control logic power for each unit's RCIC is supplied from the 125 Vdc system for that unit. [5.4-43]

The RCIC system may also be utilized to achieve hot shutdown in the event of a postulated fire since the system has the ability to be operated locally without utilizing the control room. [5.4-44]

A flow-indicating controller is used to control the flow of the RCIC pump. Pump flow is sensed by a flow element installed in the pump discharge line. The flow controller provides an output to a signal converter to either increase or decrease pump flow to achieve the desired flow. The signal converter in turn provides a signal to the turbine governor actuator to open or close the governor valve. [5.4-45]

The turbine controls provide for automatic shutdown of the RCIC turbine upon receipt of any of the following signals:

A. A reactor vessel high water level — indicates that core cooling requirements are satisfied, and to prevent damage to the turbine from water carryover.

B. A pump low suction pressure — to prevent damage to the pump due to a loss of suction.

C. A turbine high exhaust pressure — indicates a turbine or turbine control malfunction.

D. Auto isolation signal.

Upon the receipt of any of the above trip signals, automatic shutdown of the RCIC turbine is accomplished by the closure of the turbine steam supply valve. As a result, the RCIC turbine can be reset remotely and restarted from the control room in the event the RCIC system is required for maintaining reactor vessel water inventory following the receipt of an automatic shutdown signal. The high water level shutdown signal is automatically reset such that if low-low water level is subsequently sensed the RCIC system will automatically restart. A mechanical overspeed trip is installed which trips the turbine trip-throttle valve. The mechanical overspeed trip must be locally reset.

Since the RCIC steam supply line is a primary containment boundary, certain signals automatically isolate this line causing shutdown of the RCIC turbine. Automatic isolation of the RCIC steam supply line is described in the primary containment isolation section (see Section 7.3.2).

The RCIC system is designed to isolate in the event of a break in the steam supply line. The isolation is actuated on a high steam supply line flow or high area temperature. The high flow isolation is designed to respond to a large steam line break, while the high area temperature isolation is designed to respond to smaller steam line breaks. [5.4-46]

The temperature monitoring system has four temperature sensors, two located above the turbine/pump skid and two at the turbine exhaust rupture diaphragm. The two locations minimize the potential for spurious isolations due to minor steam leakage at the turbine gland seals, yet would adequately sense changes in bulk room temperature associated with an actual steam line break. The sensor trip functions follow one-out-of-two-twice logic, and employ separate divisional power supplies.

The trip level allowable value of the area temperature monitoring system is less than or equal to 169°F. This allowable value ensures an adequate system response time while maintaining a low incidence of spurious isolation.

In response to the Three Mile Island Action Plan (NUREG 0737 Item II.K.3.15), the RCIC high flow isolation was equipped with a 3 — 9 second delay feature (analytical limit) so that brief flow surges associated with RCIC system initiation do not cause inadvertent system isolation. The allowable value for this time delay is specified in the Technical Specifications. [5.4-47]

In response to the Three Mile Island Action Plan (NUREG-0737, Item II.K.3.22), RCIC logic was changed to allow Automatic Switchover of the RCIC Suction (close valve 1(2)-1301-22 and open valves 1(2)-1301-25 and 26) on a signal of low Contaminated Condensate Storage Tank (CCST) level or high Torus Level. When this logic was changed, the NRC required that the capability of remote manual containment isolation be retained. This is satisfied by a pull-to-lock feature on the suction valve control switch that overrides the suction transfer and closes the valve.

5.4.6.3 Design Evaluation

Following any reactor shutdown, heat generation continues due to the radioactive decay of fission products. During the first few seconds following a rapid shutdown, such as a scram, the fission product decay heat is augmented by delayed neutrons and the fuel temperature gradient. Immediately after a rapid shutdown from full power operation, the rate of decay heat generation can be approximately 6% of rated power. Since the pressure regulator attempts to maintain a constant pressure, the decay heat rate results in a corresponding steam generation rate, i.e., initially 6% of rated flow. The steam normally flows to the main condenser through the turbine bypass valves or, if the condenser is isolated, to the suppression pool through the relief valves. The fluid removed from the reactor vessel can be entirely made up by the feedwater pumps, or partially made up by excess flow from the control rod drive system supplied by the control rod drive pumps. [5.4-48]

If makeup water were required to supplement these primary sources of water, the RCIC turbine-driven pump unit would either start automatically upon receipt of a reactor vessel low-low water level signal, or would be started by the operator from the control room using remote manual controls.

The RCIC system delivers its design flow within 30 seconds after actuation. The design flow rate of the RCIC system is 400 gal/min, which is approximately equal to the reactor water boil-off rate 15 minutes after shutdown. While the RCIC makeup is closer to the boiloff rate during reactor cooldown following a loss of off-site power, the initial injection of the HPCI cooling water is desirable from the standpoint that the reactor vessel is quickly refilled and that redundancy is ensured. Operator action following a loss of feedwater occurrence would be to establish the flow to equal the boiloff rate. If the RCIC is functioning correctly then the HPCI system would be secured and the cooldown would be followed by manual adjustment of the RCIC flow rate.

The RCIC pump suction is normally valved to the CCST. The CCST maintains a minimum supply of 90,000 gallons in reserve for the RCIC, HPCI, and safe shutdown makeup pump (SSMP) systems. This is sufficient to allow throttled operation of the RCIC system for approximately 7.2 hours after shutdown assuming that none of the steam released from the reactor vessel is returned to the reactor vessel as condensate. Other systems (besides RCIC, HPCI, and SSMP) which use the CCST as a water supply and could jeopardize the availability of the RCIC system minimum supply, are isolated from it by standpipes, with the exception of the RHR/Core Spray Suction from the CCSTs which is isolated by means of two locked closed isolation valves in each flow path.

The RCIC pump discharge piping will remain filled as long as the system is lined up to the CCST. The level in the CCST is maintained at or above 12 feet which ensures that sufficient static head exists to maintain the keep fill in the discharge piping. [5.4-49]

The backup water supply for RCIC is the suppression pool, and the source is automatically transferred upon receipt of high torus level or low CCST level signals. The suction valves are interlocked such that CCST suction valve closure is initiated by limit switches on the suppression pool suction valves; therefore, the CCST suction valve will not start to close until the suppression pool suction valves are open. The turbine-driven pump assembly is located below the elevation of the CCST and below the minimum water level in the suppression pool, which assures a positive static head to the pump. Pump NPSH requirements are met by providing adequate static head and suction line size. [5.4-50]

The RCIC condensate pump is controlled by a float switch in the receiver tank such that the pump is on when level is high and off when level is low. The steam line drain trap bypass valve opens on high level in the drain pot. [5.4-51]

The HPCI system is designed similar to the RCIC system in that it can provide cooling water to the reactor core whenever the feedwater system is lost. In addition, however, HPCI is designed to provide coolant inventory to compensate for a LOCA. Therefore, HPCI serves a complementary function to RCIC. [5.4-52]

Should feedwater flow be lost, vessel level would drop rapidly and a reactor scram would occur. Due to collapse of voids, vessel level would rapidly continue to drop to just below the low-low level setpoint. Setting RCIC to be initiated at a higher level would not prevent the initiation of HPCI because of this rapid drop in vessel level. The probability of needing RCIC is not greater than the probability of needing HPCI. Therefore, HPCI functioning as an ECCS would not be hindered or delayed by the operation of the RCIC system.

Long-term operation of the RCIC system may require space cooling to maintain equipment temperatures within allowable limits. The RCIC system is designed to withstand a complete loss of offsite ac power to its support systems, including coolers, since the room coolers and the diesel generator cooling water pump (which supply the coolers) are supplied from emergency buses fed by the diesel generators. Therefore, continuous power is available for the space coolers following a complete loss of offsite ac power. [5.4-53]

Failure of the high water level trip of the RCIC turbine could lead to overfilling the reactor vessel. However, a finite time is required to reach high water level after receipt of the low-low level initiation signal, on the order of 1 - 2 hours for the RCIC system alone, 5 - 10 minutes for the combination of HPCI and RCIC systems. Operator action can be taken to manually control flow rate and/or shut down the systems prior to flooding the main steam lines. Should water enter the main steam lines, the ability of the RCIC turbine to withstand water slugging is comparable to that of the HPCI turbine. [5.4-54]

It is extremely unlikely that a break in the steam line supplying the RCIC turbine would cause isolation of the HPCI system and result in a loss of HPCI capability. The reason is that the temperature elements for sensing area high temperature (an indication of a steam line break) which cause system isolation are in fact appropriately located within each of the equipment spaces which contain the HPCI and RCIC turbines. There are no other temperature sensors which isolate the HPCI or RCIC turbines. These equipment spaces are completely separated by a 4-foot thick concrete wall. There is no proximity of the two steam supply lines within either space. In fact, neither system's steam supply line passes through the equipment space of the other system. Therefore, a high area temperature within either the RCIC or HPCI equipment space would result only in the isolation of the system which has equipment located within that space, and should have no effect on the other system.

5.4.6.4 Inspection and Testing

5.4.6.4.1 Preoperational Testing

The functional operability of the RCIC system was initially tested at the time of system installation and plant startup. Prior to installation, extensive tests were conducted to demonstrate that the RCIC turbine could perform as designed. These tests were conducted on equipment scheduled for installation on Quad Cities Unit 1, and the results proved the capability of the RCIC turbine assembly to perform as designed. [5.4-55]

The objectives of the test series included:

1. Normal spin test to verify (a) operation and set points of safety devices (alarms, remote trip, oil trip, overspeed trip), (b) operation of the turbine control system (stability, adjustments, oil pressures), and (c) general turbine steady state operation (vibration, steam and oil pressures, and temperatures).

2. Verify governor performance during quick start tests with various inlet pressures.

3. Verify turbine acceleration capabilities and limitations during quick start tests with various inlet pressures.

4. Verify capability of turbine control valve travel limitor (pressure-sensing device) during quick start tests with various inlet pressures.

5. Verify load capacity of the turbine at the low pressure (controlling) design point.

6. Verify capability of the gland condenser system to prevent external steam leakage from the turbine seals during quick start transients and steady-state operation.

Steam inlet conditions were varied between 100 psig and 825 psig (boiler capacity), and load conditions between zero and 350 hp for the series of quick start tests. For all tests, the governor speed set point was 4500 rpm, and the turbine control system was capable of limiting rotor acceleration and holding speed below the overspeed trip setpoint. The maximum governor overshoot, observed with 800 psig, and no load, resulted in turbine speed of 5000 rpm, 3 1/2% below the overspeed trip setting during the test of 5175 rpm, which is fully satisfactory. (Note: the current overspeed trip setting is 5600 rpm.) All control components operated as expected, or better. It should be noted that the quick start test conditions were more severe than those expected during system startup, two major items being: [5.4-56]

1. A constant control signal was input to the governor during the entire startup transient, calling for maximum speed. In actual system startup, the signal diminished during the startup transient, resulting in less governor overshoot.

2. Most quick start tests demonstrating governor capability were conducted with essentially zero load on the turbine, whereas in actual system startup, the turbine will be driving the pump load during the transient, again resulting in less governor overshoot.

All quick start tests demonstrated the ability of the turbine to attain full speed within the required 20—25 seconds. Steady-state tests indicated the turbine load capacity exceeded design requirements.

During all test transients and steady state conditions, the gland condenser system was capable of preventing external steam leakage.

All test objectives were completely satisfied. Test results proved that the turbine equaled or bettered all design requirements.

5.4.6.4.2 Testing During Plant Operation

A design flow functional test of the RCIC system is performed during plant operation by taking suction from the CCST and discharging through the full flow test return line to the CCST. The discharge valve to the feedwater line remains closed during the test, and reactor operation is undisturbed. Opening the pump discharge valve is accomplished by first shutting the upstream discharge valve. Control system design provides automatic return from test to operating mode if system initiation is required during testing. Periodic inspection and maintenance of the turbine-driven pump unit is carried out in accordance with the manufacturer's recommendations.

5.4.6.5 Safe Shutdown Makeup Pump System

5.4.6.5.1 Design Basis

The purpose of the safe shutdown makeup pump (SSMP) system is to provide cooling water to the Unit 1 or Unit 2 reactor core in the event that the reactor becomes isolated from the main condenser simultaneously with a loss of the feedwater system. To achieve this purpose, the SSMP system was designed to supply makeup water to the reactor core at the same capacity as the RCIC system; specifically, 400 gal/min over a reactor pressure range of 1135-165 psia. [5.4-57]

5.4.6.5.2 Design Description

The SSMP system was installed as a common backup to the Unit 1 and Unit 2 RCIC systems to satisfy the requirements of 10 CFR 50, Appendix R, Section III.G, "Fire Protection of Safe Shutdown Capability." The system bypasses fire zones which could theoretically disable the RCIC system; this is discussed in Section 3.0 of the Safe Shutdown Report (Fire Protection Reports, Volume 2). [5.4-58]

The SSMP system is located in a room in the east central area of the turbine building ground floor. The SSMP system consists of a motor-driven pump unit and associated valves and piping capable of delivering makeup water to either reactor vessel. A summary of the design requirements of the pump and motor is shown in Table 5.4-4. The SSMP is shown in FSAR Figure 5.4-10 and P&ID M-70.

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

In the event the reactor vessel becomes isolated, and the feedwater supply becomes unavailable, the automatic pressure relief subsystem (described in Section 6.3) is provided to maintain vessel pressure within specified limits. The water level in the reactor vessel will drop due to continued steam generation by decay heat followed by release of the steam through the relief valves to the suppression pool. The SSMP system can be initiated manually from either the control room or the SSMP room. The motor driven pump will supply demineralized makeup water to either units reactor from the CCST. An alternate source of makeup water is available from the fire header.

The pump discharge is delivered to the reactor vessel through the feedwater line for Unit 1, and through the HPCI system line for Unit 2. The SSMP system injection valves, MO-1(2)-2901-8, are interlocked to allow injection into only one reactor vessel at a time. A flow path must be available to meet the start interlocks for the system. The system will trip if the flow control valve, MO 0-2901-6, closes. This occurs to prevent pump damage due to overheating in low flow conditions.

A room cooler is located in the SSMP room to maintain room temperature during system operation. Service water is supplied to the water-cooled condenser in the unit. An alternate source of cooling water is available from the fire header.

Electric power for the system is supplied from 4-kV bus 14-1 (normal) or 4-kV bus 24-1 (reserve). Power from either of these buses can be fed to 4-kV bus 31, which is located in the SSMP room. Bus 31 supplies the feed breaker for the pump motor and also a feed breaker for 4-kV/480-V transformer 30. This transformer supplies power to MCC 30, also located in the SSMP room, which in turn provides power to the motor operated valves, room cooler, lighting, and local instrumentation. Interlocks preclude closing the normal and reserve power feed breakers in 4-kV bus 31 at the same time, and also require that the feed breaker in 4-kV 14-1 or 24-1 is closed before closing the redundant breaker in 4-kV bus 31. The normal and reserve feed breakers in 4-kV buses 14-1 and 24-1 will shed load on a bus undervoltage, and can be closed into the bus once the undervoltage condition is cleared.

5.4.6.5.3 Design Evaluation

Following any reactor shutdown, heat generation continues due to the radioactive decay of fission products. During the first few seconds following a rapid shutdown, such as a scram, the fission product decay heat is augmented by delayed neutrons and the fuel temperature gradient. Immediately after a rapid shutdown from full power operation, the rate of decay heat generation can be approximately 6% of rated power. Since the pressure regulator attempts to maintain a constant pressure, the decay heat rate results in a corresponding steam generation rate, i.e., initially 6% of rated flow. The steam normally flows to the main condenser through the turbine bypass valves or, if the condenser is isolated, to the suppression pool through the relief valves. The fluid removed from the reactor vessel can be entirely made up by the feedwater pumps, or partially made up by excess flow from the control rod drive system supplied by the control rod drive pumps. If makeup water were required to supplement these primary sources of water, the SSMP system could be started by the operator from the control room. The SSMP system can also be initiated locally in the SSMP room. The flow rate of the SSMP system is approximately equal to the reactor water boil-off rate 15 minutes after shutdown.

The SSMP suction is normally valved to the CCST. The CCST maintains a minimum supply of 90,000 gallons in reserve for the SSMP, RCIC, and HPCI systems. This is sufficient to allow controlled operation of the SSMP for approximately 7.2 hours after shutdown without assuming that any of the steam released from the reactor vessel is returned as condensate. Other systems (besides SSMP, RCIC, and HPCI) which use the CCST as a water supply and could jeopardize the availability of the SSMP minimum supply are isolated from it by standpipes, with the exception of the RHR/Core Spray Suction from the CCSTs which is isolated via two locked closed isolation valves in each flow path.

The backup supply of water for the SSMP is the fire header. This water source may be manually aligned in response to low CCST level signals. [5.4-59]

All components necessary for initiating operation of the SSMP are completely independent of all offsite ac power, plant service air, and external cooling water systems.

5.4.6.5.4 Inspection and Testing

The functional operability of the SSMP system was tested at the time of system installation.

A design flow functional test of the SSMP is performed during plant operation by taking suction from the CCST and discharging through the full flow test return line to the CCST. The injection valves to the reactors remain closed during the test, and reactor operation is undisturbed.

5.4.7 Residual Heat Removal System - Shutdown Cooling and Other Functions

Reactor shutdown cooling is accomplished by use of the RHR system, operating in the shutdown cooling mode. The RHR system is described in detail in Chapter 6. The equipment and operations associated with the shutdown cooling mode are discussed in the following paragraphs.

The residual heat removal (RHR) system has three modes of operation to satisfy all design objectives and bases. The original modes were 1) low pressure coolant injection (LPCI), 2) containment cooling, and 3) reactor shutdown and head cooling. The head cooling piping has been removed and head cooling is no longer part of this mode of operation. The LPCI and containment cooling modes are primarily safety functions and are described in detail in Section 6.3 and 6.2, respectively. Shutdown cooling is a normal operating mode and is discussed in this section. [5.4-60]

The major equipment of the RHR system consists of two heat exchangers, four main system pumps, and four RHR service water pumps. The equipment is connected by associated valves and piping, and controls and instrumentation are provided for proper system operation. The RHR system is shown in FSAR Figure 5.4-11 and P&IDs M-39 and M-37 for Unit 1, and in M-81 and M-79 for Unit 2. The RHR service water system is discussed in Section 9.2.1. [5.4-61]

5.4.7.1 Design Bases for Shutdown Cooling

The design bases of the shutdown cooling mode of the RHR system are as follows: [5.4-62]

The shutdown cooling mode is designed to be utilized (after a normal depressurization using the main condensers) when reactor pressure reaches 100 psig. The system is designed to remove reactor residual and decay heat at a rate such that the vessel temperature will be reduced to 125°F 20 hours after rod insertion. Further, the system is designed to maintain this temperature (or lower) for refueling and servicing operations. These capabilities are based upon a service water coolant temperature of 85°F at the heat exchanger inlet with a maximum of 125°F at the outlet. [5.4-63]

Residual heat removal system equipment is designed in accordance with Class I seismic criteria (see Chapter 3) to resist sufficiently the response motion within the reactor building from the design basis earthquake. The pumps are designed and constructed in accordance with the Hydraulic Institute Standards. The shell side of the heat exchangers is designed in accordance with ASME Section III, Class C vessels, and the tube side is designed in accordance with ASME Section VIII. The provisions through Winter 1966 Addenda, paragraph N2113, apply. The residual heat removal system meets the code requirements of the State of Illinois, TEMA, and USAS specification.

5.4.7.2 System Design

5.4.7.2.1 General System Description

A summary of the design requirements of the RHR pumps and heat exchangers is presented in Tables 5.4-5 and 5.4-6. The pump characteristics are shown in Figure 5.4-13. [5.4-64]

One loop, consisting of a heat exchanger, two RHR pumps in parallel, and associated piping, is located in the northeast corner of the reactor building. Another similar loop is located in the southeast corner of the reactor building to minimize the possibility of a single physical event causing the loss of the entire system. Both loops are located as close to the suction header as practical to minimize the vulnerability of the piping. The two loops of the RHR system are cross-connected by a single header, making it possible to supply either loop from the pumps in the other loop.

The RHR pump seals and motor are cooled by the water being pumped. Cooling water is therefore available whenever these pumps are in operation. Two small heat exchangers are provided for each pump: one for the pump seals and one for the cooling coil located in the motor upper thrust bearing lube oil reservoir. The process fluid that is being pumped is circulated through the primary side of the heat exchangers while flow through the secondary side is taken from the discharge of the RHR service water pumps discussed in Section 9.2.1.

Redundant system flow paths are provided by two independent lines, each sized for 100% flow, that are physically separated and protected.

Generally, RHR system piping is carbon steel. The piping from the isolation valves to the reactor system, however, is stainless steel because it normally contains reactor coolant. Pressure relief valves are employed in the carbon steel section of piping to provide overpressure protection. All system components are designed in accordance with applicable codes for reactor auxiliary systems.

5.4.7.2.2 Shutdown Cooling Mode

The shutdown cooling mode of the RHR system is intended for routine operation and is not a safety requirement. Although the same heat exchangers and pumps used for safety modes are also used for operational purposes, this usage is not sufficient to result in degradation of the equipment due to wear. On the contrary, such use "exercises" the equipment and verifies its operability. [5.4-65]

The shutdown cooling mode is provided to perform the reactor cooling function after reactor pressure and temperature have been reduced to the point where cooling via the main condenser is no longer efficient. This mode is not an automatic feature of the RHR system, but is manually actuated during a normal plant shutdown. [5.4-66]

The shutdown cooling mode utilizes two of the four RHR pumps, one of the two RHR heat exchangers, two of the four RHR service water pumps, and the necessary valves and piping to connect the components and connect the suction and discharge lines to the vessel through the recirculation system piping. Interties are also available on the suction side to connect to the suppression pool, fuel pool cooling system, and the condensate storage tanks. [5.4-67]

In the shutdown cooling mode, the RHR system is designed to draw saturated liquid from the reactor and pass it to the RHR heat exchangers. Adequate NPSH is available over the entire range of reactor pressures that can occur in the shutdown cooling mode.

During the shutdown cooling mode of operation the process fluid being pumped would approach 338°F, corresponding to a saturated steam pressure of 100 psig in the vessel. Since the RHR process fluid being pumped provides the cooling to the RHR pump seals and motor, this temperature would be too high to ensure adequate cooling. However, in the shutdown cooling mode, the RHR service water pumps are in operation and RHR service water flow is established through the secondary side of the pump seal and motor cooler heat exchangers. This allows the temperature of the cooling water being supplied to the pump and motor to be maintained within allowable limits.

5.4.7.2.3 Other Functions of the Residual Heat Removal System

Beyond its three basic modes of operation, the RHR system can perform specific manually actuated functions, including: [5.4-68]

A. Supplementing the fuel pool cooling system;

B. Draining the condenser to the suppression chamber by taking water from a condensate pump;

C. Delivering and returning reactor water to the fuel pool cooling system demineralizer for cleanup;

D. Transferring water from the reactor vessel and cavity to the main condenser or to the suction of the condensate pumps; and

E. Transferring water from the suppression chamber to the radwaste system or via the radwaste system to the main condenser.

Use of the RHR system for supplemental fuel pool cooling requires several manual operations including the installation of spool pieces joining the RHR system to the fuel pool cooling system. This configuration would render one of the two loops (two pumps and one heat exchanger) unavailable for use in either of the safety modes (LPCI or containment cooling). [5.4-69]

Systems that cool the fuel pools can also be used as an alternate method of decay heat removal from the reactor cavity during refueling outages when the reactor cavity is flooded above a level of 23 feet. When the gates between the reactor cavity and the fuel pool and between the two fuel pools are removed, a natural circulation develops between the reactor cavity and spent fuel pools due to the temperature and density differences between the three bodies of water. To qualify this alternate method of decay heat removal, an analysis is performed prior to the refueling outage to evaluate the heat load in both the reactor vessel and spent fuel pool that will be unique to each refueling outage.

The heat load is calculated using ANSI/ANS 5.1-1979 with two sigma adders. From the heat load, the required number of fuel pool cooling (FPC) system trains and RHR loops aligned to fuel pool assist (FPA) are determined. It may be necessary to route a portion of the cooling flow directly to the refueling cavity instead of the fuel pool. Conservative values for the RHR service water temperature and reactor building closed cooling water (RBCCW) are determined based on the time of year during which the refueling outage occurs. This analysis demonstrates that the temperature of the water in the reactor cavity will not exceed Technical Specification limits if specified FPC and/or RHR-FPA system flow rates and cooling water temperatures are maintained. Requirements for fuel pool cooling as described in UFSAR Section 9.1.3.1 must also be satisfied. Furthermore, analysis is performed to show that no local boiling will occur on the surface of the fuel rods. Administrative controls are procedurally implemented and the water temperature in the reactor cavity and the fuel pools is monitored to ensure compliance with the analysis assumptions and results such as time, flow, and temperature limits.

When the HPCI system is operated the suppression chamber water level rises. Normal level can be restored by opening RHR system valves and transferring water to the radwaste system, or to the condenser hotwell via the radwaste system. [5.4-70]

To support the variety of other functions which the RHR system is capable of, the RHR system pumps may draw suction from several sources and deliver its discharge to various places. Appropriate electrical interlocks are provided between the shutdown cooling suction valve and the suppression pool suction valve, the torus spray/test return valve, and the inter-loop crosstie valve to prevent inadvertently draining the reactor vessel or the fuel pool into the suppression chamber. [5.4-71]

The following valve interlocks are provided to reduce the potential for inadvertent draining of the reactor vessel by establishing a high flow drain path. The operability of these interlocks is not required to perform any safety related function (i.e., the interlock need not be operable to consider the RHR loop operable) but the interlock circuitry is composed of safety related Class 1E components that should provide a high degree of reliability and availability to prevent the inadvertent draining of the vessel. Administrative controls or station procedures may be used to bypass these interlocks provided the possibility of inadvertent draining of the vessel has been evaluated. It is anticipated that the bypasses would be in place for only short periods of time and for very specific plant evolutions or interlock equipment failure. [5.4-72]

<u>MO 1001-7A, B, C, D</u>

The suppression pool suction valves are interlocked with their respective shutdown cooling valve (MO 1001-43A, B, C, or D) such that the 7 valve can not be opened if the corresponding 43 valve for the same pump has been opened.

<u>MO 1001-43A(C) and 43B(D)</u>

The shutdown cooling valves are interlocked with their respective suppression pool suction valve (MO 1001-7A, B, C, or D) such that the 43 valve can not be opened if the 7 valve for the same pump has been opened.

The shutdown cooling valves 43A and 43B for the A-loop are interlocked with the crosstie valve (MO 1001-19A) and the torus spray/test return valve (MO 1001-34A). Neither the 43A nor the 43B valve may be opened if either the 19A or the 34A valve has been opened. [These interlocks are new and were added in response to INPO SOER 87-2.]

The shutdown cooling valves 43C and 43D for the B-loop are interlocked with the crosstie valve (MO 1001-19B) and the torus spray/test return valve (MO 1001-34B). Neither the 43C nor the 43D valve may be opened if either the 19B or the 34B valve has been opened. [These interlocks are new and were added in response to INPO SOER 87-2.]

<u>MO 1001-34A, B</u>

The torus spray/test return isolation valves are interlocked with their respective shutdown cooling suction valves. For the A-loop, the 34A valve may be opened provided neither the 43A or the 43B valve is open. For the B-loop, the 34B valve may be opened provided neither the 43C nor the 43D valve has been opened. [These interlocks are new and were added in response to INPO SOER 87-2.]

5.4.7.3 Performance Evaluation

5.4.7.3.1 Equipment Capability

The specifications for the RHR system pumps are shown in Table 5.4-5 and the pump performance curve is shown in Figure 5.4-13.

5.4.7.3.2 Shutdown Cooling Mode Performance

The shutdown cooling mode functions to cool the vessel by taking suction from the recirculation system suction line and discharging to the recirculation system discharge line. Three RHR system suction line isolation valves must be open to provide the suction source. Two of the valves, the containment isolation valves, can be opened only after the vessel pressure has been reduced to 130 psig or less (Technical Specifications allowable value). It is also necessary to open the valves leading to the recirculation discharge line. These valves can be opened whenever vessel pressure is less than or equal to 342 psig (Technical Specification allowable value) which is the upper pressure limit for the RHR system when operated in the LPCI mode. Manual start of the RHR system pumps for the shutdown cooling mode can be accomplished only after the suction valves are opened.[5.4-73]

Shutdown cooling capacity is based on a two-pump flow rate of 10,700 gal/min with the heat exchanger in operation. Normal operation allows injection of the full 10,700 gal/min directly into the vessel through the jet pumps.

Heat removal calculations are based upon a 7000 gal/min flow of RHR service water through the heat exchanger (3500 gal/min per pump). Although the heat removal calculations are based upon a flow rate of 7000 gal/min, the operator will actually use flow much less due to cooler river temperatures and ambient temperature losses. During normal shutdown cooling operation, the operator typically uses only one RHR service water pump at a reduced flow. Cooldown rate is monitored by the operator. Flows and RHR heat exchanger bypass valve positions are adjusted to control the cooldown rate. The RHR service water pumps can be started manually at any time and would normally be started prior to starting a RHR system pump. The service water pressure in the tube side of the heat exchanger is 20 psi higher than the primary coolant pressure in the shell side. This feature ensures that no potentially contaminated water flows from the process side of the heat exchanger to the service water side which ultimately discharges to the river. An alarm is sounded in the control room in the event that this differential pressure drops to 15 psi or less.

During shutdown cooling operation, the sensors that initiate ECCS are not bypassed. (See Section 6.3.2.2.4 for information on LPCI operation during shutdown cooling.)

5.4.7.4 Testing and Inspection

5.4.7.4.1 Preoperational Testing

Prior to plant startup, a preoperational test of the RHR system was conducted. This test assured the proper functioning and operation of all instrumentation, pumps, heat exchangers, and valves and verified that the system met its design performance requirements. In addition, system reference characteristics, such as pressure differentials and flow rates, were established at that time to be used as base points for testing performed during plant operation. [5.4-74]

5.4.7.5 Residual Heat Removal or Reactor Water Cleanup Pipe Break Detection

High temperatures in the spaces occupied by the reactor shutdown cooling mode of RHR system piping and the reactor water cleanup system piping outside the primary containment are sensed by temperature switches that activate alarms on the RHR and RWCU systems, plus initiates automatic isolation of the RWCU System, indicating possible pipe breaks. [5.4-75]

Automatic isolation of the RWCU system on high temperature plus the reactor vessel low water level isolation function is adequate to prevent the release of significant amounts of radioactive material in the event that the system suffers a breach. For RHR reactor vessel low water level, isolation function provides adequate prevention if a breach occurs.

5.4.8 Reactor Water Cleanup System

5.4.8.1 Design Bases

The purpose of the reactor water cleanup (RWCU) system is to: [5.4-76]

A. Remove insoluble, waterborne activation products from the reactor coolant;

B. Prevent soluble inorganic impurities (i.e., chlorides) from concentrating in the reactor coolant and exceeding specified water quality limits;

C. Reduce beta and gamma radiation sources in the reactor coolant resulting from the presence of corrosion and fission products; and

D. Remove water from the reactor coolant system at reduced activity levels during startup and shutdown.

5.4.8.2 System Description

The RWCU system is shown in UFSAR Figure 5.4-12 and P&IDs M-47 for Unit 1 and M-88 for Unit 2. [5.4-77]

The RWCU system design provides for the continuous treatment of approximately 100,000 lb/hr of reactor water to remove various impurities and thus maintain coolant quality in accordance with reactor water quality specifications. Flow can be increased to approximately 200,000 lbm/hr to increase decay heat removal or improve reactor water chemistry. To increase flow, a second recirculation pump must be started and both non-regenerative heat exchanger trains must be in service.

The important operations in the RWCU process are as follows:

A. Removing soluble and insoluble impurities by using filter-demineralizer equipment (waterborne impurities are removed by filtration, absorption and ion exchange by the filter-demineralizer cake);

B. Maintaining system inventory by returning the same quantity of water that was extracted or providing a reject flow path to maintain reactor water level while water is being added to inventory; and

C. Reducing and increasing water temperature with regenerative and nonregenerative heat exchangers, at the appropriate points in the process, to protect the ion-exchange resins and minimize system heat losses.

The RWCU system is operated during reactor startup, reactor power operation, and reactor shutdown. In conjunction with the condensate filter-demineralizer equipment, the RWCU system maintains specified reactor water quality by limiting the input of waterborne impurities to the reactor and by removing such impurities from the reactor water. (During refueling, the RWCU system, in conjunction with the fuel pool filter-demineralizers, also maintains fuel pool water clarity and reduced activity levels.)

The RWCU system is operated by continuously diverting a portion of the reactor coolant flow from the suction line of one of the recirculation system pumps. The system design flow rate is based on the operating water volume in the reactor, and is established on the basis of limiting total iodine concentration to the order of 1 μCi/cc in the reactor water assuming a cleanup half-life of approximately 4 hours. The turnover time is approximately 6 hours.

The water taken from the reactor at operating pressure is cooled by passage through the tube sides of regenerative and nonregenerative heat exchangers. One or both of the RWCU pumps may be used to overcome piping and equipment pressure drops and provide the driving head for the system. The RWCU pump discharge flows through two parallel filter-demineralizer units for the removal of impurities (a temperature less than 120°F is normally maintained to optimize performance of the ion exchange resins). A bypass line with a remote motor operated valve provides a means to bypass the filter-demineralizer units when necessary.

Flow from the filter-demineralizers is normally routed to the shell side of the regenerative heat exchanger, and returns to the reactor through the feedwater system. During startup, flow from the filter/demineralizers can be directed to the main condenser hotwell or the waste collector tank for the removal of "swell" and control rod drive water.

The purpose of the regenerative heat exchanger is to recover sensible heat in the reactor water and to reduce the cycle heat loss. The purpose of the nonregenerative heat exchanger is to further cool the reactor water by transferring heat to the reactor building closed cooling water system.

During normal operation, the temperature of reactor water leaving the nonregenerative heat exchanger is 80° to 120°F. During blowdown or startup modes temperature may increase to approximately 130°F. This is due to reduced effectiveness of the regenerative heat exchanger resulting from part of the shell side return flow being bypassed to the main condenser hotwell or the radwaste system.

Two 250 gpm capacity, parallel-operated, filter-demineralizer units are provided. They are of the powdered resin precoat type, using finely ground, nonregenerable, mixed (cation and anion) ion exchange resins. [5.4-78]

The end of a filter-demineralizer service cycle is identified either by high pressure drop across the unit, or by exhaustion of the ion exchange resins. Normally, resin exhaustion, typically on silica, will limit the precoat life. When a unit's service cycle has ended, the unit is isolated by closure of the inlet and outlet valves while the parallel unit remains in service. The off-line unit is backwashed using service air and condensate transfer to remove all of the resin material and accumulated insoluble material. The backwashing process is designed to efficiently remove these materials with a minimum water volume discharged to the phase separator tanks.

Backwash water flows by gravity to a phase separator tank located below each filter-demineralizer. Vent lines from the unit are also routed to the phase separator tank. The water and resin slurry is normally held in the phase separator tank to allow decay of short-lived radioactive material, and accumulation of sufficient material for disposal. The slurry is then pumped to the radwaste system for disposal.

After backwashing, the unit is precoated. Precoat recirculation is continued for approximately 30 minutes. After precoating, the small resin feed tank is nearly empty of resin, and no resin is evident in the return water pumped to the larger precoat tank. At this time, a holding pump is started which maintains the resin in place and then the precoat pump is stopped and associated valving closed. The unit is then placed in service by opening the inlet and outlet valves, establishing system flow through the filter-demineralizer, and stopping the holding pump by placing it in "auto start" mode.

There are 87 one inch diameter filter elements and vent tubes in the reactor water cleanup (RWCU) filter/demineralizers constructed of a porous metal. The elements have a maximum differential pressure rating of at least 80 psid and a filtration rating of two microns. The filter/demineralizer has a post strainer that is designed with a maximum differential pressure rating of at least 250 psid. The filter vessels and post strainers have differential pressure alarms and differential pressure trips that will automatically place the demin in hold. This instrumentation is to enhance system reliability by notifying the operator of off normal conditions that may indicate depleted resin or resin bleed through. [5.4-79]

It should also be recognized that the elements are undergoing a crushing force, as the flow of liquid is from the outside towards the center of the tube. This type of flow takes advantage of the inherent geometrical strength of the cylindrical element.

The basic cleanup system filter vessel design at Quad Cities is of the type proven in other services with the added requirements that the vessels meet specific ASME code requirements. The RWCU Filter Demineralizer Vessels & Pumps were designed, fabricated, tested, and inspected in accordance with ASME Section III, Class C. The vessels are designed for 1400 psi at 150°F, fabricated of Type 304 stainless steel with full penetration welds and 100% radiograph on all circumferential and longitudinal welds. The vessels were hydrotested to 2160 psi at ambient temperature and carry the "U" and "N" stamps as required by Class C of the ASME code.

The RWCU system is protected with relief valves on the shell and tube sides of both the regenerative and nonregenerative heat exchangers, and on the filter-demineralizers. The low-pressure precoat system auxiliary equipment is also protected with relief valves. To protect the ion exchange resins from high temperature, an alarmed temperature indicator is provided on the outlet of the nonregenerative heat exchanger. High temperature (130°F) will annunciate in the main control room. The isolation valves will close and the RWCU system pumps will trip at 140°F. The isolation valves are motor operated and located on each side of the primary containment wall. In addition to high fluid temperature, the isolation valves are closed automatically upon actuation of the standby liquid control system. Valve isolation signals are also provided from the containment isolation system (Group 3 isolation) upon reactor low water level, high area temperatures in the RWCU piping areas, and the MSIV Steam Tunnel High Temperature. [5.4-80]

In the event of loss of flow or unusually low flow through the filter/demineralizer, the holding pump will automatically start after a time delay of approximately five seconds. There are both local and main control room alarms to notify the operator of the low flow condition lasting five seconds. After flow is restored, the holding pump will automatically stop. The filter/demineralizers are equipped with local flow controllers that can be operated in an automatic or manual flow control mode.

Continuous sampling stations are located in the influent header and in each effluent line of the two filter-demineralizer units. The influent sample station is also a source for obtaining reactor water grab samples.

The cleanup system instrumentation for flow, pressure, temperature, and conductivity are recorded or indicated in the main control room with appropriate alarms provided. Instrumentation and controls for backwashing and precoating the filter-demineralizers are located at a local panel in the reactor building. Valves behind shielding are individually controlled from a local panel and are furnished with on-off air operators or with extension stems that penetrate the shielding for manual operation.

Blind flanges have been provided for chemical cleaning and decontamination of the RWCU system for maintenance.

Original RWCU system piping that was susceptible to IGSCC was replaced from a point inside the drywell, through penetration X-14, out to the containment isolation valve. An isolation valve and flued head transition piece were also replaced. All materials were IGSCC-resistant, and welds were made with IGSCC-resistant techniques. Modification M04-1(2)-91-027A&B was implemented to replace the IGSCC susceptible piping, valves and regenerative heat exchangers on the non-safety related portions of the system. The piping installed extends from the outboard containment isolation valve MOV 1(2)-1201-5 up to and including the regenerative heat exchangers, and the piping between the regenerative and non-regenerative heat exchangers. On the return side, piping was replaced from the regenerative heat exchangers up to and including the 1(2)-1201-82 valve. Also, included in the replacement was the pump suction piping from the non-

regenerative heat exchangers to the RWCU pumps and the pump discharge piping from the pumps up to and including the 1(2)-1299-11 valve. Piping was fabricated in accordance with B31.1, 1989 Edition. [5.4-81]

All materials were IGSCC resistant and welds were made with IGSCC resistant techniques.

The new RWCU system will be a modified single train consisting of one set of three regenerative heat exchanger shells in series with a parallel path configuration of two existing non-regenerative heat exchanger trains. This heat exchanger layout is capable of passing up to 2% of feedwater flow to meet the station operational requirements. In changing from the dual train to a single train regenerative heat exchanger arrangement, the total heat removal capacity and the total system flowrate will remain the same as the original dual heat exchanger train configuration.

A regenerative heat exchanger bypass line was added that bypasses flow around the regenerative heat exchanger to increase the capacity of the RWCU recirculation loop for decay heat removal when the unit is in cold shutdown. [5.4-82]

The replacement regenerative heat exchangers were designed and fabricated to ASME Section VIII, 1992 Edition, incorporating the supplemental design rules of ASME Section III, 1965 Edition and the requirements of General Electric Specifications 21A5789, Rev. 0, 21A9220, Rev. 7, and 21A9220AB, Rev. 2. The vessels were hydrotested to 2370 psi and carry the "U" stamp as required by ASME Section VIII. [5.4-83]

5.4.8.2.1 Operating Modes

The following is a brief description of RWCU system operation under the various plant conditions. [5.4-84]

A. During normal power operation, water flows from the reactor through the regenerative heat exchangers, nonregenerative heat exchangers, RWCU pumps, filter-demineralizer, then back through the regenerative heat exchangers and to the reactor vessel through the feedwater line. System flow is limited by pump impeller design (runout) during single pump operation, and by pump NPSH requirements during dual pump operation.

B. At the beginning of reactor startup, reactor water level is above the low water level scram to accommodate water swell. During startup, the RWCU pumps are in operation, the regenerative heat exchanger is under partial load, the nonregenerative heat exchanger is under maximum startup load, the filter-demineralizers are in operation, and excess inventory due to control rod drive and swell are routed to the main condenser (primary) or to radwaste. During startup, at about 212°F, minimum NPSH occurs. The pump NPSH design requirements are based on the beginning of startup mode of operation, when reactor heatup is in progress and no steam pressure exists in the reactor vessel.

C. For the reactor coolant system, the blowdown flow rate is limited by the maximum allowable outlet temperature of the reactor building closed cooling water on the shell side of the nonregenerative heat exchanger. The maximum

temperature differential across the nonregenerative heat exchanger occurs during this mode of operation. (The number of temperature cycles [maximum temperature differential across the nonregenerative heat exchanger assumed to occur whenever the nonregenerative heat exchanger inlet temperature exceeds 300°F] in 40 years is limited to a maximum of 550, per paragraph N-415.1 of 1968 ASME Section III.) During blowdown, the RWCU pumps are in operation or bypassed, the regenerative heat exchanger is under no load, the nonregenerative heat exchanger is under full load, the filter-demineralizer is online or bypassed, and the entire flow is discharged to radwaste or to the main condenser.

D. During reactor hot standby mode, the RWCU pumps are in operation, the regenerative heat exchanger is under full load, the nonregenerative heat exchanger is under partial load, the filter-demineralizers are in operation, and the flow is back to the reactor vessel with no bypass flow.

E. Normally, the single train of regenerative heat exchangers and both trains of nonregenerative heat exchangers are in service. If one of the trains of nonregenerative heat exchangers are removed from service during system operation, system flow is limited by heat exchanger thermal design and cooling water design limits. Plant procedures limit the number of pumps that can be run to the number of sets of nonregenerative heat exchangers that are in service. [5.4-85]

F. During shutdown or refueling conditions, the RWCU system is used for water level control by adjusting the reject (to condenser) flow rate and to assist in decay heat removal (via the non-regenerative heat exchangers). Additional decay heat removal capacity can be achieved by opening the bypass around the regenerative heat exchangers to run the RWCU system in a decay heat removal mode. The thermal limitations on the feedwater piping and nozzles require that the temperature difference between the RWCU outlet and the feedwater inlet to the reactor be less than 220°F. RWCU is used in the decay heat mode (bypass open) only in modes 4, 5, and * (no mode), so as to not exceed this thermal limit. RWCU may also be used for decay heat removal during the hydrostatic or inservice leak test. RWCU is used to maintain the temperature in the reactor vessel by removing excess heat added by the recirculation pump(s). Procedural limitations during this test ensure that the reactor vessel metal temperatures exceed the pressure vs temperature (P/T) curves in Technical Specifications without RWCU outlet temperature exceeding the thermal limitations on the feedwater piping and nozzles. [5.4-86]

5.4.8.2.2 Radiological Considerations

The total amount of radioactive material contained in the reactor coolant is limited in order to minimize the potential offsite doses resulting from reactor coolant released from the unit. At high activity rates, the coolant activity is proportional to the off-gas release rate, since both result from leaking fuel. For example, the coolant activity corresponding to a chimney release rate of 0.7 μCi/s is approximately 1.7×10^{3} μCi/cc which could result in a maximum offsite lifetime thyroid dose under normal operating conditions of 3.0×10^{-3} rem. This is 10^{5} times lower than the 10 CFR 100 guideline dose of 300 rem. A coolant activity of 1.7×10^{3} μCi/cc would result in a maximum offsite lifetime thyroid dose under accident conditions of only 3.0 rem (considering a main steam line break accident) which is still a factor of 100 below the 10 CFR 100 guideline dose. Therefore, by limiting the

reactor coolant activity to 1.7 x 10^3 μCi/cc, a factor of 100 conservatism is obtained. The limiting condition for operation of each unit given in the Technical Specifications is less than or equal to 0.2 μCi/gram dose equivalent I-131. This limit provides considerably more conservatism. [5.4-87]

5.4.8.3 Inspection and Testing

The RWCU system is normally in continuous use, so full system testing and inspection are not necessary. The isolation valves are tested periodically to verify operability and leak tightness. A sample point on the cooling water side of the nonregenerative heat exchanger permits analysis for tube leaks. [5.4-88]

5.4.9 Main Steam Line and Feedwater Piping

5.4.9.1 Description

Steam piping is shown on FSAR Figure 10.3-1 and P&IDs M-13 for Unit 1, and M-60 for Unit 2, and is described in Section 10.3. The feedwater piping is described in Subsection 10.4.7 and shown on P&IDs M-15 for Unit 1, and M-62 for Unit 2.

The feedwater piping consists of two lines of 18 inch nominal diameter from the high pressure feedwater heaters to the reactor.

The materials used in the steam and feedwater piping comply with the design codes and supplementary requirements described in Section 3.2. The general requirements of the feedwater system are described in Sections 7.7 and 10.4.

5.4.9.2 Performance Evaluation

Differential pressure on reactor internals under the assumed accident condition of a ruptured steam line is limited by a flow restrictor in each of four main steam lines.

5.4.9.3 Inspection and Testing

Inspection and testing are carried out as described in Sections 10.3 and 10.4. Inservice inspection was considered in the design of the main steam and feedwater piping, and adequate working space and access for inspection of selected components are provided.

5.4.10 Pressurizer

This section is not applicable to Quad Cities Station.

5.4.11 Pressurizer Relief Tanks

This section is not applicable to Quad Cities Station.

5.4.12 Valves

Refer to Tables 5.1-1, 6.2-1, 6.2-6, and 6.2-7 and Section 5.4.1.4.3.1.

5.4.12.1 Design Bases

The criteria are as described in Section 3.9 for ASME Class 1, 2, and 3 valves. Compliance with ASME Codes is discussed in Sections 3.2 and 3.9.

5.4.12.2 Description

For the recirculation system, valve sizes were selected to match piping sizes, which were determined based upon flow rate and velocity requirements. The recirculation loop flow rates through the suction and discharge valves were determined based upon heat balance and jet pump hydraulic requirements. [5.4-89]

The maximum differential pressure ratings for fully open valves are based upon rated flow requirements under normal operating conditions. The maximum differential pressure ratings for closing valves are based upon LPCI requirements under LOCA conditions for the discharge and equalizer valves, and upon maintenance considerations for the suction valves. The maximum differential pressure when opening the discharge valves is based on opening against the shutoff head of the recirculation pump at 30% of rated speed. The suction valves are only required to open under static head conditions with no flow in the recirculation loop. [5.4-90]

All Limitorque operators on valves which function to mitigate the consequences of a LOCA, MSLB or HELB, as required by 10 CFR 50.49, are environmentally qualified. [5.4-91]

Three of the four valves in the equalizer line between the recirculation loops are closed at all times during reactor operation. The fourth valve is open to accommodate thermal expansion. [5.4-92]

None of the motors on recirculation system valve operators can be started in the valve-opening direction if the valve is already fully open. This feature precludes driving the valve stem into the back seat and causing damage to the seating seal.

A safety-related motor operated valve program has been established out of Exelon/Quad Cities response to the NRC Generic Letter (GL) 89-10, “Safety-Related Motor-Operated Valve Testing and Surveillance.” As part of the commission’s closeout of GL 89-10, Generic Letter (GL) 96-05, “Periodic Verification of Design-Basis Capability of Safety Related Motor Operated Valves” was issued. All safety-related motor-operated valves are being tested and evaluated as per the criteria set forth in GL-96-05. These tests and evaluations are being continually performed to demonstrate that sufficient capacity exists to operate the valves within the design pressures and voltages established by the design basis review. [5.4-93]

Motor operated valves 1(2)-1001-29A/B, 1(2)-1301-48, 1(2)-1301-49, 1(2)-1402-24A/B, 1(2)-1402-25A/B, 1(2)-2301-8 and 1(2)-2301-9 were evaluated in accordance with Generic Letter 95-07 “Pressure Locking and Thermal Binding of Safety-Related Power Operated Gate Valves”, and determined to be potentially susceptible to pressure locking. These valves have either been modified (i.e., drilling a hole in the disc) to eliminate the susceptibility to pressure locking; or have had administrative controls established for surveillance testing purposes, and to ensure that the pressure locking modification is performed at the next scheduled internal maintenance on the valve.

5.4.13 Safety/Relief Valves

Refer to Table 5.1-1 and Section 5.2.

5.4.13.1 Design Description

Pressure relief valves are designed and constructed in accordance with the same code classes as those of the line valves in the system. Section 3.2 lists the applicable code classes for valves. The design criteria, design loading, and design procedure are as described in Section 3.9.

5.4.13.2 Performance Evaluation

The use of pressure-relieving devices assures that operating pressures and pressure transients do not exceed 110% of the design pressure of the system. The number of relieving devices on a system or portion of a system was determined on an individual component basis.

5.4.13.3 Inspection and Testing

Quad Cities Station has a preventive maintenance program for relief valves. Per Technical Specifications, relief valve settings are checked to assure that the set pressures are as specified. In addition, each Target Rock safety/relief valve is overhauled or replaced with an overhauled valve at least every second refueling outage. Each Electromatic type relief valve is replaced at least every second refueling outage with a rebuilt valve (i.e., at least 50% of the valves are replaced every refueling outage). The pilot solenoid for each valve is replaced with a rebuilt unit every refueling outage. [5.4-94]

5.4.14 Component Supports

Information on component supports is contained in Section 3.9.

Table 5.4-1

JET PUMP CHARACTERISTICS

General	
Number of jet pumps	20
Throat ID	8.1 in.
Diffuser ID	20 in.
Nozzle internal diameter	3.31 in.
Hydraulic Parameters	
Diffuser exit velocity	13.2 ft/s
Driving flow (per pump)	4720 gal/min, 1.79 x 10^6 lb/hr
Suction flow (per pump)	8160 gal/min, 3.106 x 10^6 lb/hr
M ratio	1.729
MN efficiency (includes 180° bend)	30.8%
Jet pump head (discharge to suction)	67.6 ft

Table 5.4-2

HYDROGEN WATER CHEMISTRY SYSTEM TRIPS

Area hydrogen concentration high

Reactor scram

** Low hydrogen flow

Hydrogen storage area trouble (excess flow valve 1/2-2799-4 CLOSED)

** Low power level (main steam flow)

Operator request (manual)

** - Bypassed during plant start and early power ascension.

Table 5.4-3

REACTOR CORE ISOLATION COOLING SYSTEM
EQUIPMENT SPECIFICATIONS

Pump	
Number	1
Discharge pressure	525 — 2800 ft
Flow rate	400 gal/min
NPSH	20 ft
Turbine	
Steam pressure inlet	150 — 1120 psia
Steam pressure exhaust	25 psia
Power	80 — 500 HP
Steam flow rate	6,000 — 16,500 lb/hr

Table 5.4-4

SAFE SHUTDOWN MAKEUP PUMP SYSTEM
EQUIPMENT SPECIFICATIONS

Pump	
Number	1
Discharge pressure	2885 ft
Flow rate	400 gal/min
NPSH	15 ft
Motor	
Voltage	4000 V
Phase	3
Cycles	60
RPM	3550
Power	600 HP

Table 5.4-5

RESIDUAL HEAT REMOVAL EQUIPMENT DESIGN PARAMETERS

Pumps, Main System

Number	4 (Note 1)
Type	Single stage-vertical-centrifugal
Seals	Mechanical
Drive	Electric Motor
Power source	Normal auxiliary or standby diesel
Speed	3600 rpm
Pump casing	Cast steel
Impeller	Stainless steel
Shaft	Stainless steel
Code	ASME Section III, Class C

Note:

1 The parameters used in the integrated ECCS performance LOCA analysis required by 10 CFR 50 Appendix K are discussed in Section 6.3.3 (SAFER/GESTR).

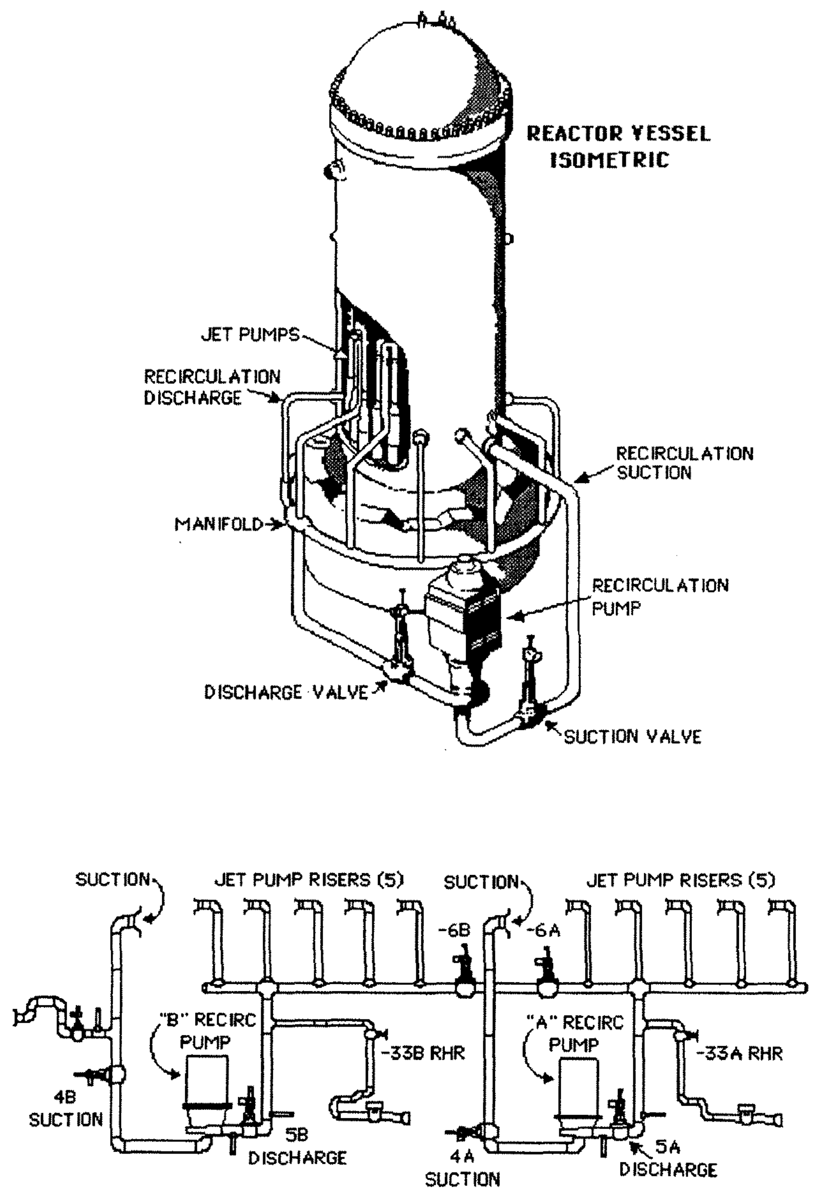
Table 5.4-6

RESIDUAL HEAT REMOVAL HEAT EXCHANGER DESIGN PARAMETERS

Heat Exchangers

Quantity	2
Heat Load	105 x 10^6 Btu/hr each
Primary side flow (containment water)	10,700 gal/min
Secondary side flow (river water)	7,000 gal/min*
Design temperatures	
River water	95°F
Containment water	165°F
Design pressure	
Primary (shell)	450 psi
Secondary (tube)	350 psi
Design codes	
Code (shell)	ASME Section III, Class C
Code (tube)	ASME Section VIII

*Note: Heat exchanger design parameter is 7,000 gal/min; however, only 3,500 gal/min (one RHR service water pump) is needed during accident conditions. Shutdown cooling mode was sized for 7,000 gal/min; however, typical operation in this mode uses only one pump at reduced flow.

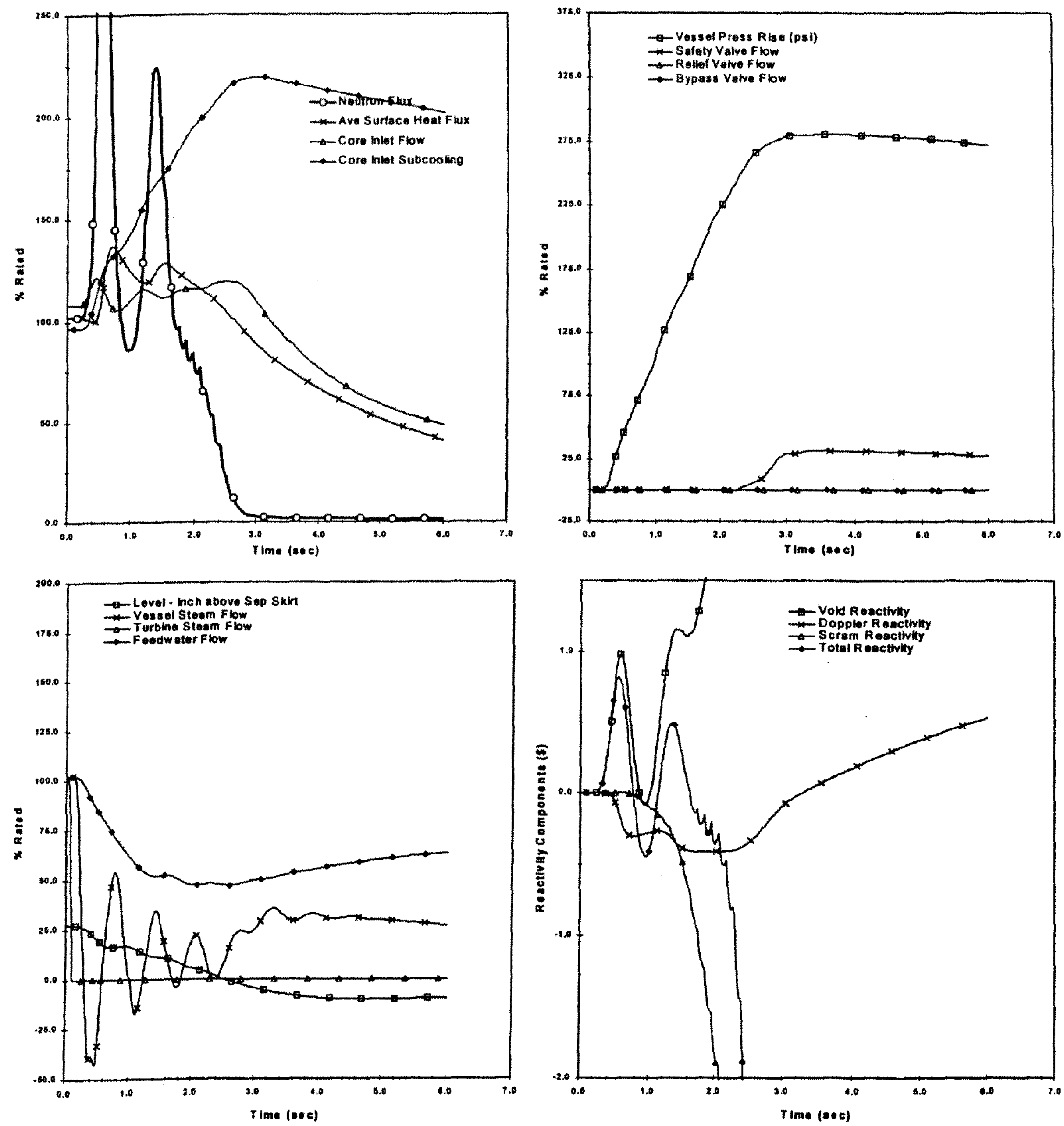


QUAD CITIES STATION UNIT 1

DIAGRAM OF NUCLEAR BOILER & REACTOR RECIRCULATION PIPING

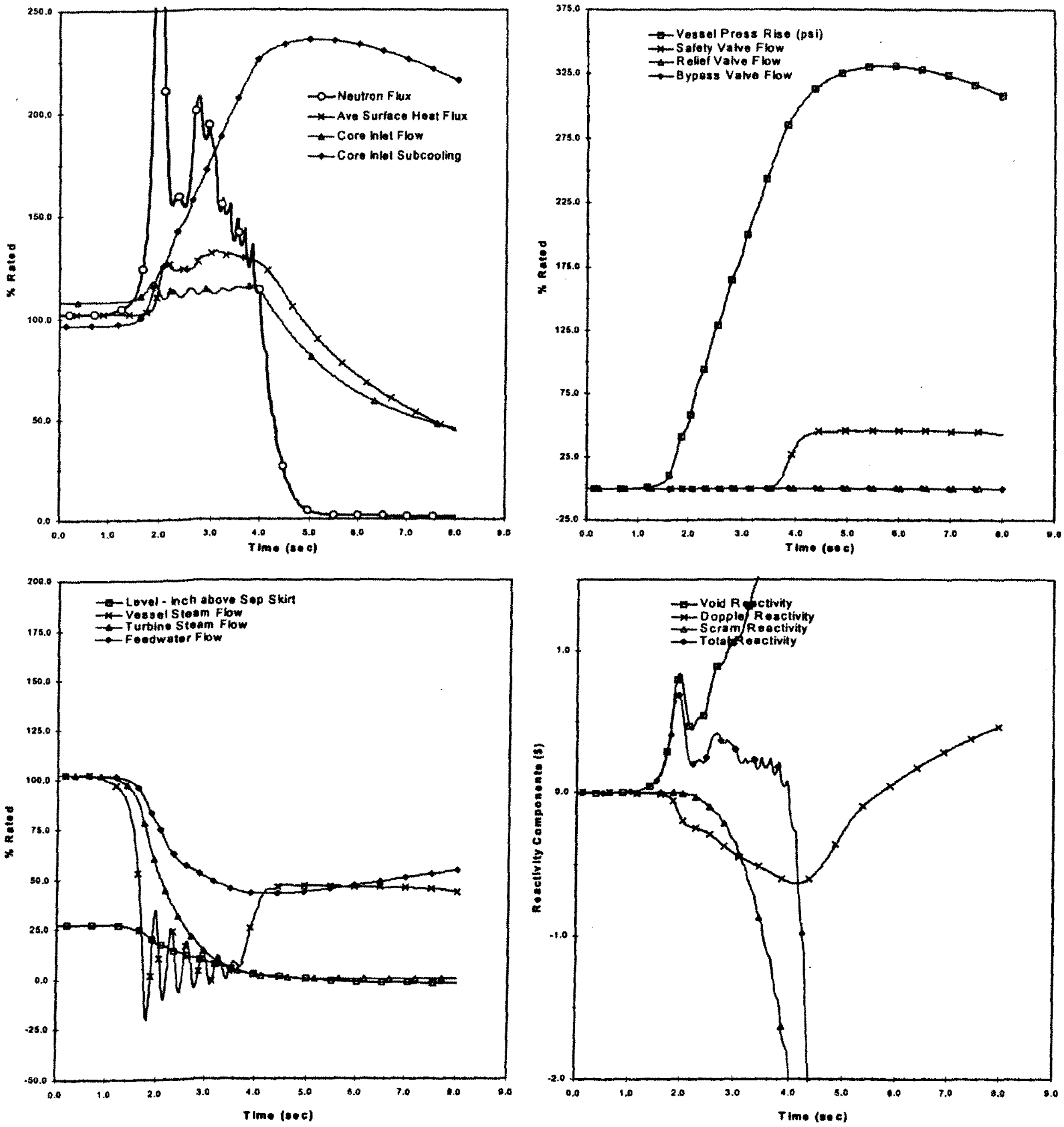
FIGURE 5.1-1

REV. 3, DECEMBER 1995



See the discussion in Section 5.2.2.2.2 for information regarding use of details from this analysis which may not be applicable to the current fuel cycle.

QUAD CITIES STATION UNITS 1 & 2
TURBINE TRIP, NO BYPASS –TRANSIENT ANALYSIS
FIGURE 5.2-1
REVISION 7, JANUARY 2003

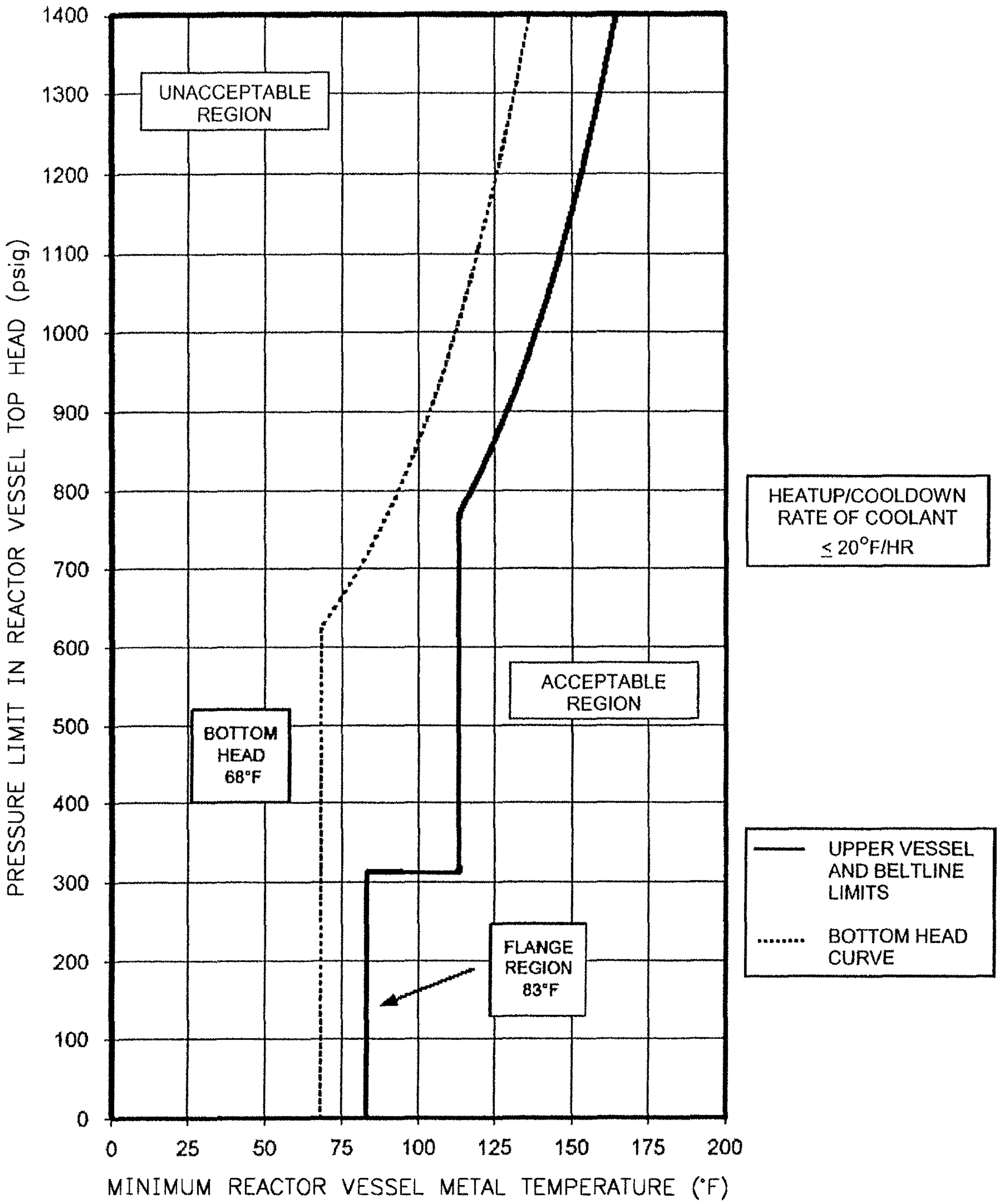


See the discussion in Section 5.2.2.2.3 for information regarding use of details from this analysis which may not be applicable to the current fuel cycle.

QUAD CITIES STATION UNITS 1 & 2
MSIV CLOSURE, FLUX SCRAM – TRANSIENT ANALYSIS
FIGURE 5.2-3
REVISION 7, JANUARY 2003

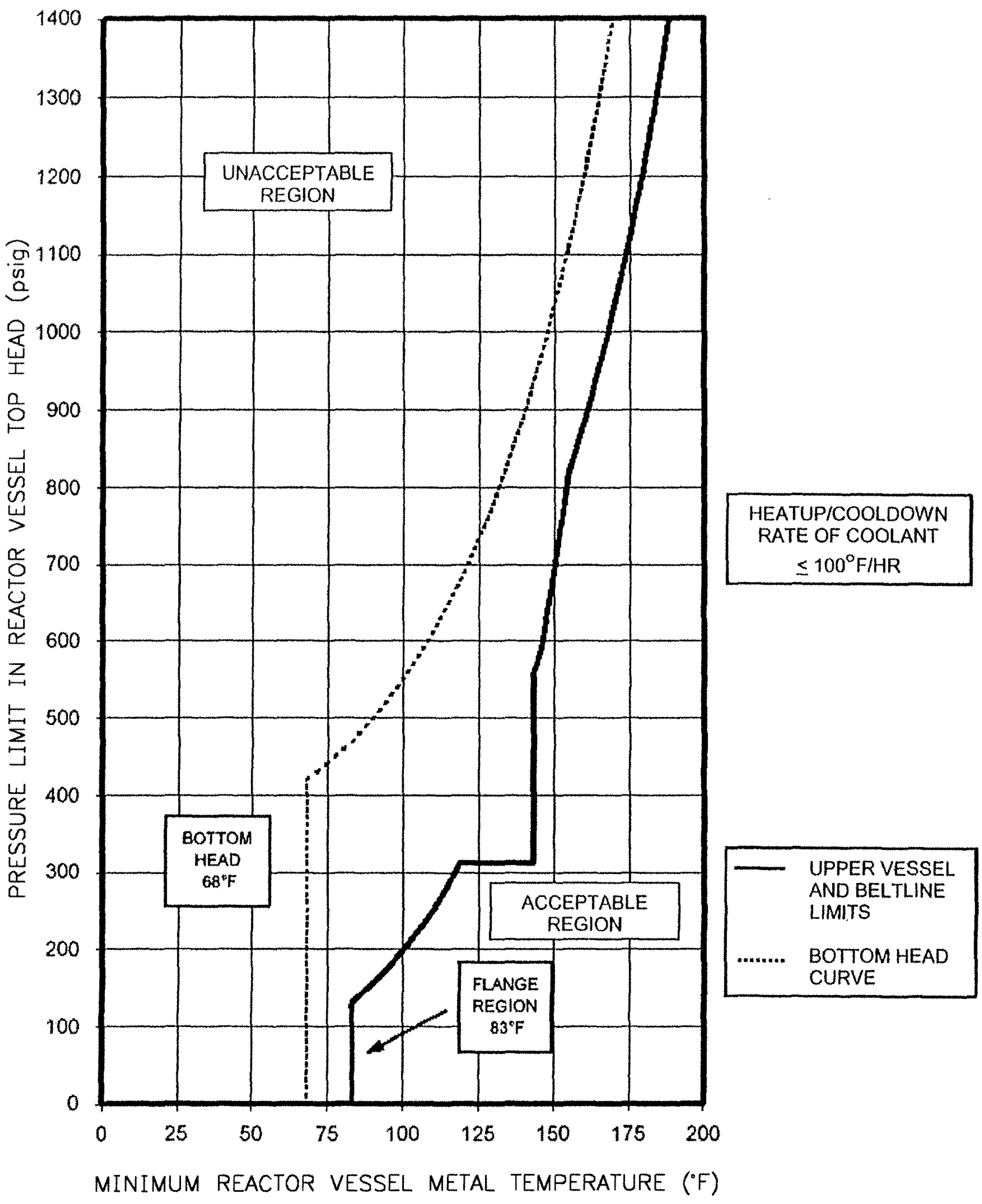
CURVE A

PRESSURE-TEMPERATURE LIMITS FOR PRESSURE TESTING - VALID TO 54 EFPY



QUAD CITIES STATION UNITS 1 & 2
PRESSURE-TEMPERATURE LIMITS FOR PRESSURE TESTING - VALID TO 54 EFPY
FIGURE 5.3-4a
REVISION 9, OCTOBER 2007

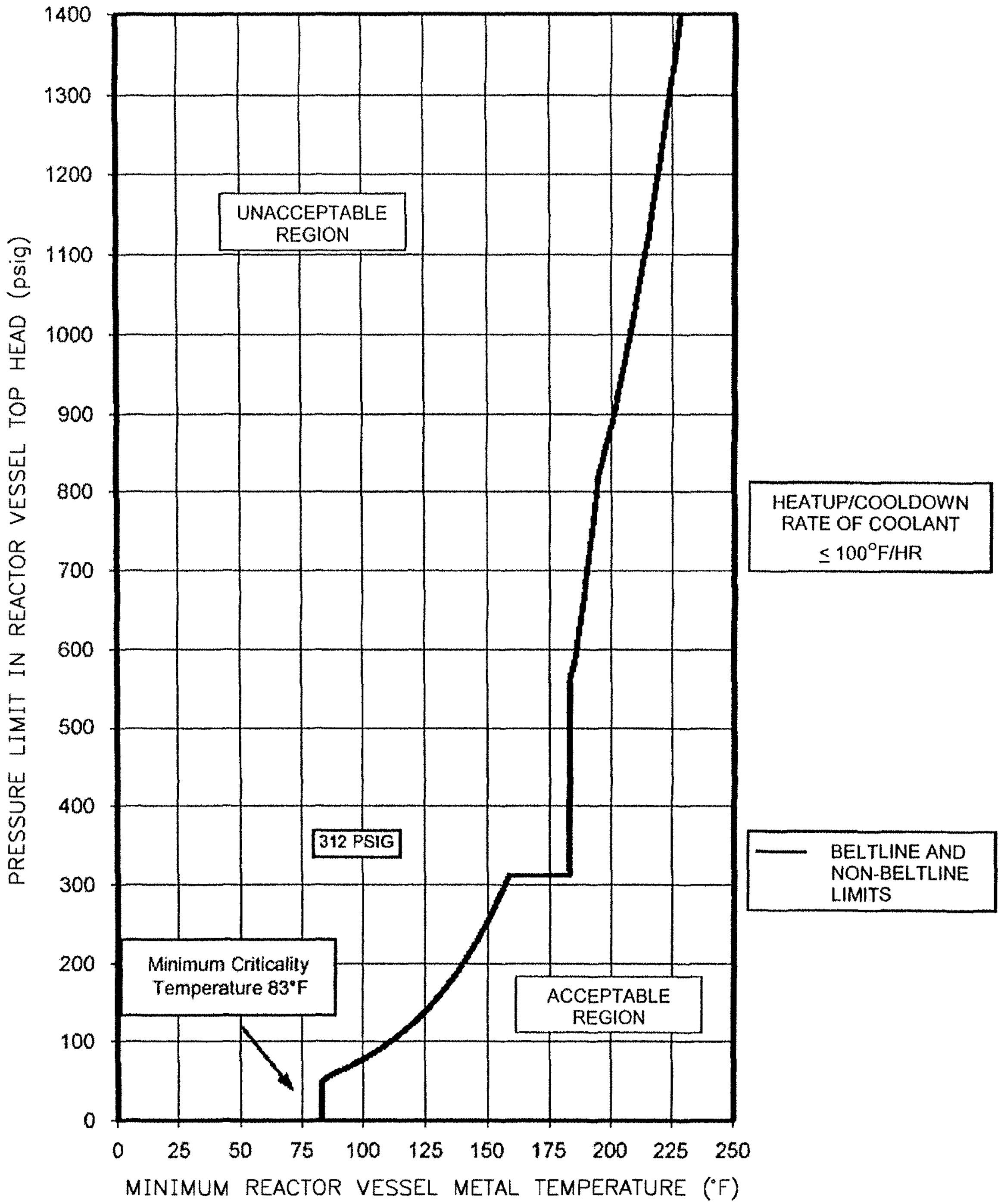
CURVE B
PRESSURE-TEMPERATURE LIMITS FOR NON-NUCLEAR HEATUP/COOLDOWN - VALID TO 54 EFPY



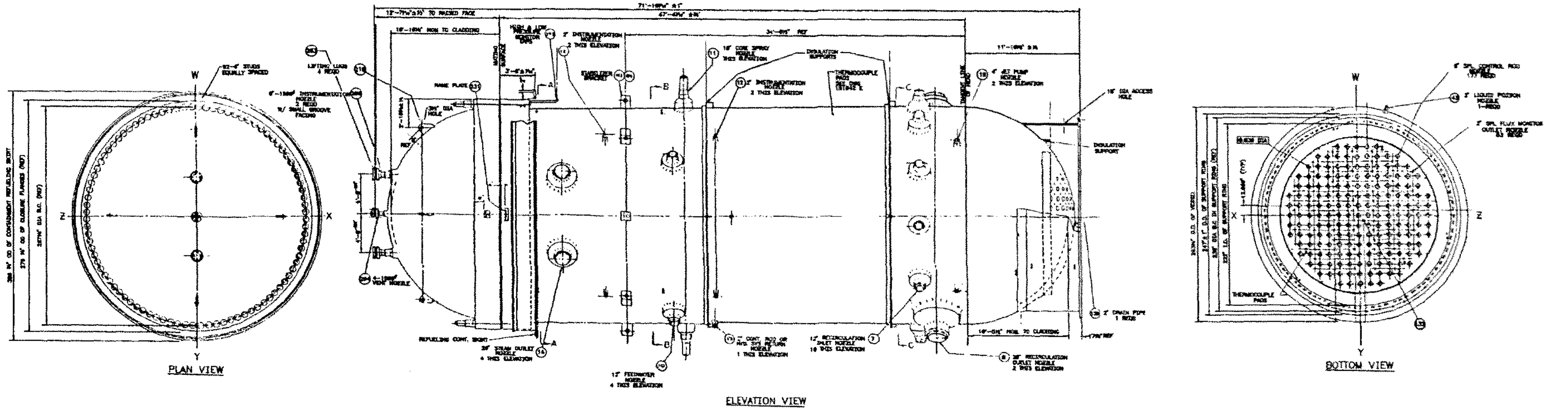
QUAD CITIES STATION UNITS 1 & 2
PRESSURE-TEMPERATURE LIMITS FOR NON-NUCLEAR HEATUP/COOLDOWN - VALID TO 54 EFPY
FIGURE 5.3-4b
REVISION 9, OCTOBER 2007

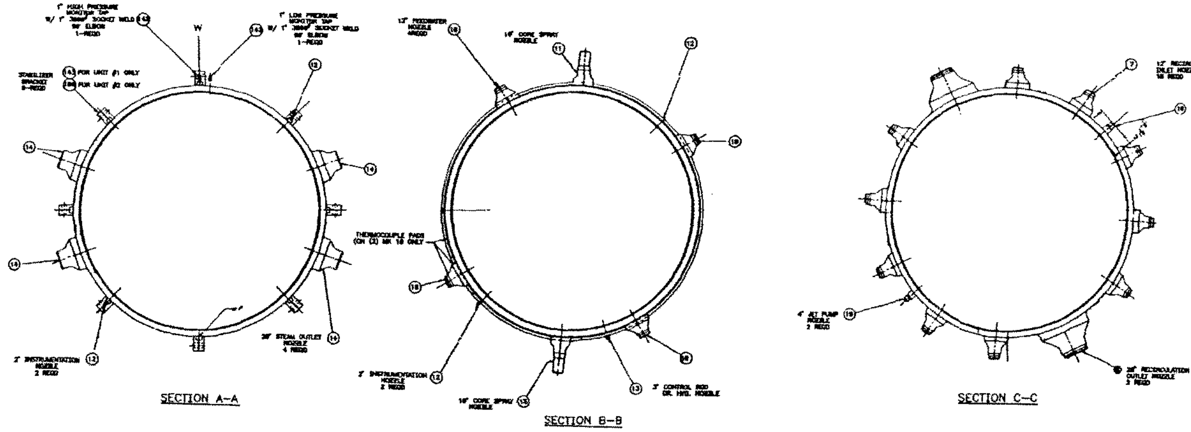
CURVE C

PRESSURE-TEMPERATURE LIMITS FOR CRITICAL CORE OPERATIONS - VALID TO 54 EFPY



QUAD CITIES STATION UNITS 1 & 2
PRESSURE-TEMPERATURE LIMITS FOR CRITICAL CORE OPERATIONS - VALID TO 54 EFPY
FIGURE 5.3-4c
REVISION 9, OCTOBER 2007





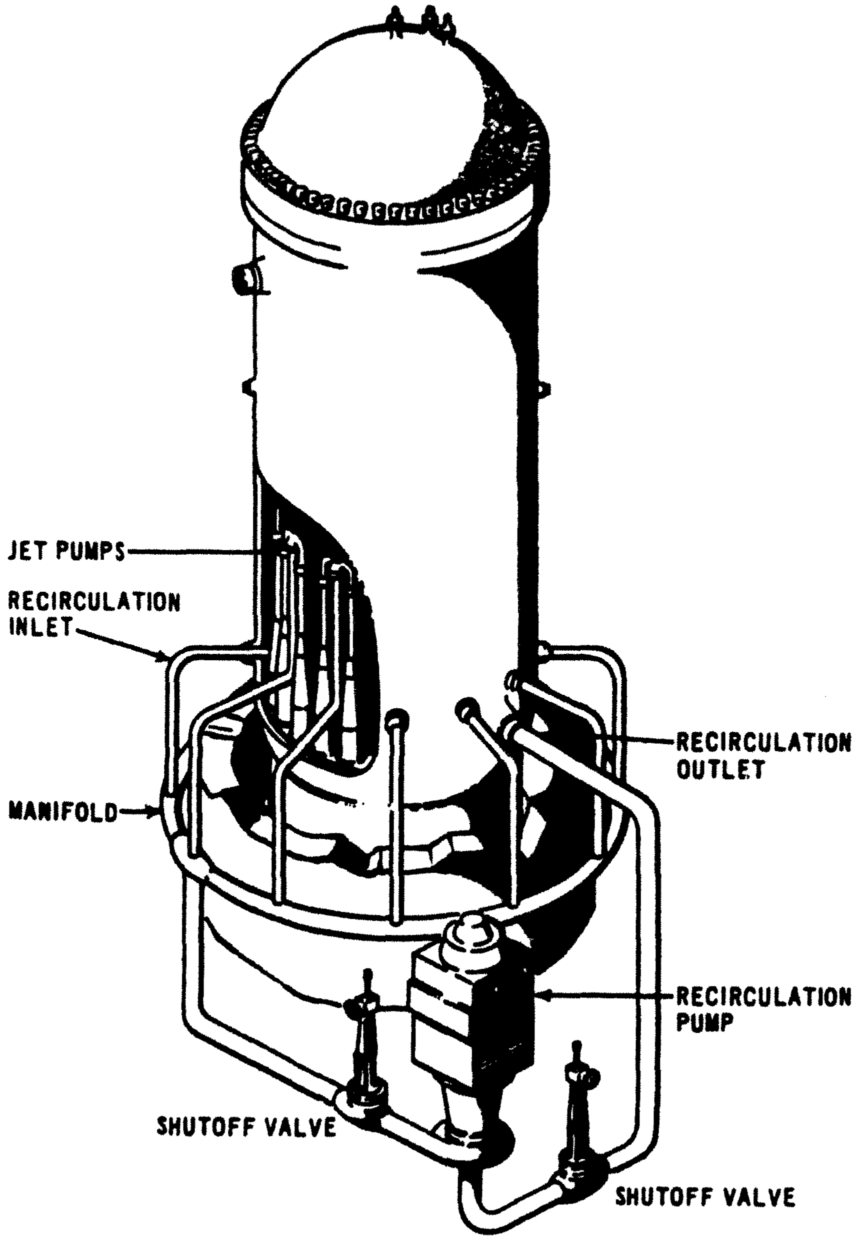
TYPICAL WELD PREPS FOR NOZZLE ENDS					
NOZZLE	MK NO	NOM SIZE		I.D.	O.D.
RECIRC OUTLET	8	28"		25.938"	[illegible]
RECIRC INLET	7	12"		11.232"	[illegible]
STEAM OUTLET NOZZLE	14	20"		18.326"	[illegible]
FEEDWATER	18	12"		10.959"	[illegible]
CORE SPRAY	11	10"		9.671"	[illegible]
CONT. ROD DR. HYD RETURN	13	3"		2.934"	[illegible]
LIQUID POISON	43	2"		1.953"	[illegible]
INSTRUMENTATION	12	2"		1.953"	[illegible]
JET PUMP	19	4"		3.688"	[illegible]

QUAD CITIES STATION UNITS 1 & 2

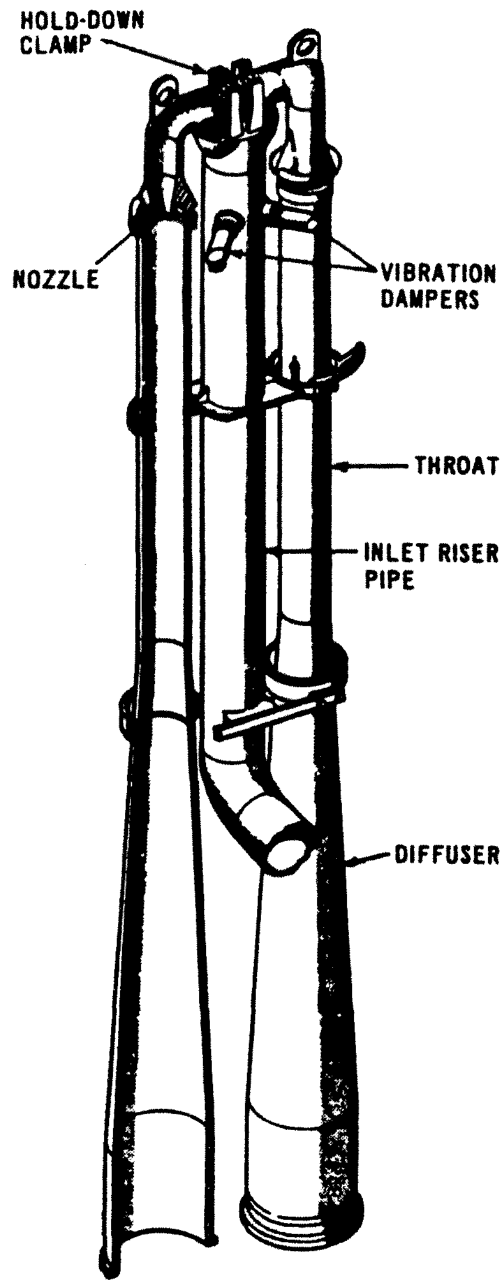
REACTOR VESSEL

FIGURE 5.3–5

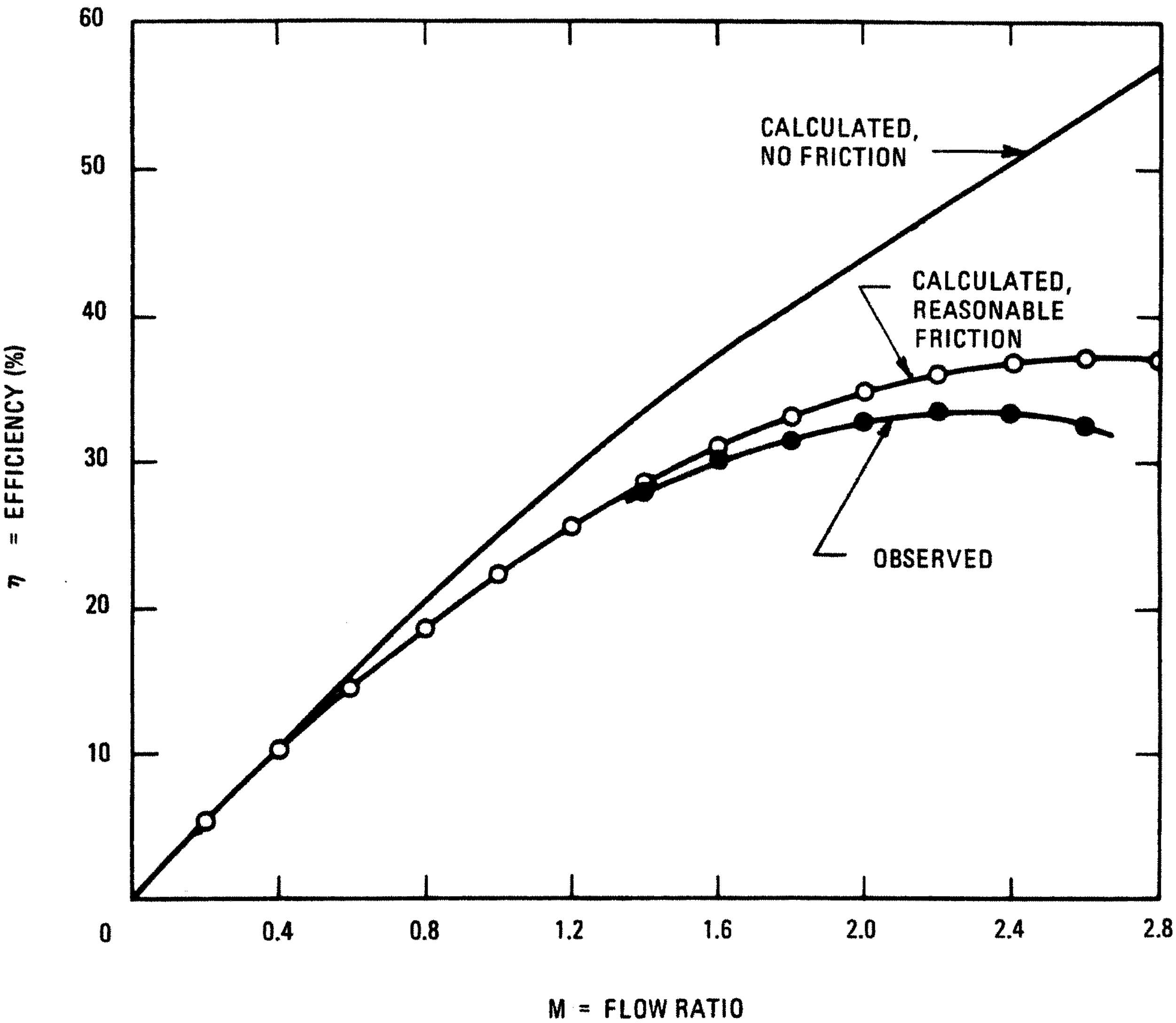
REVISION 6, OCTOBER 2001



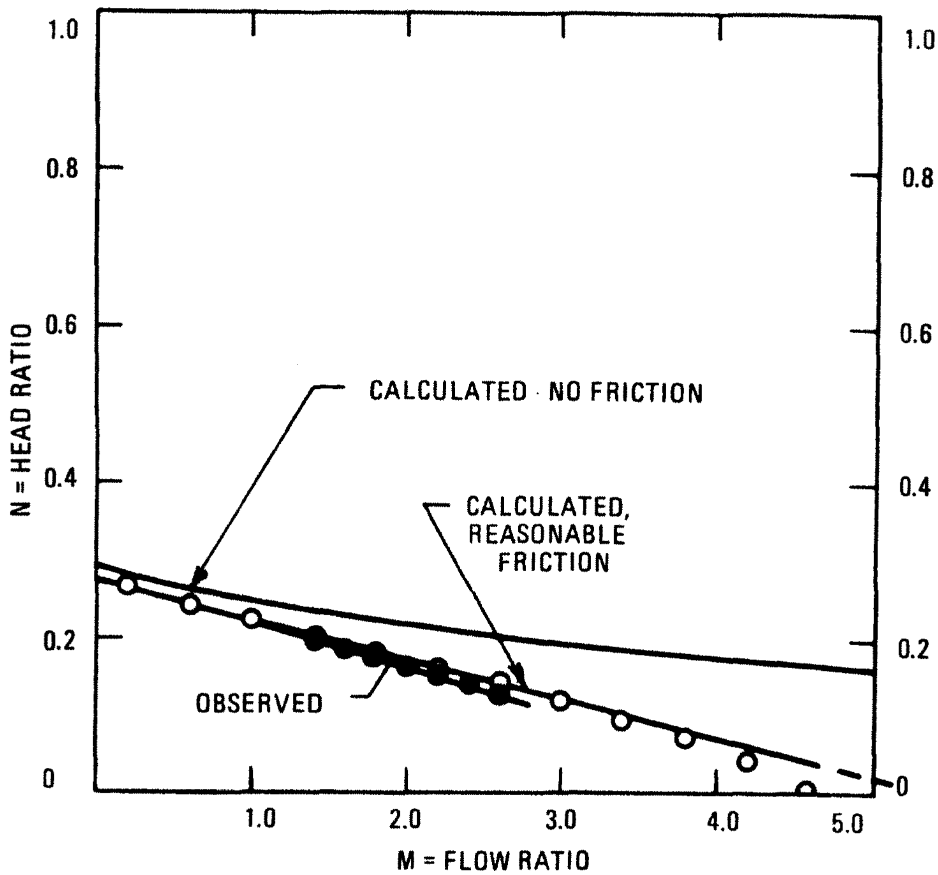
QUAD CITIES STATION UNITS 1 & 2	
REACTOR VESSEL ISOMETRIC	
FIGURE 5.4-1	



QUAD CITIES STATION UNITS 1 & 2
JET PUMP ISOMETRIC
FIGURE 5.4-2



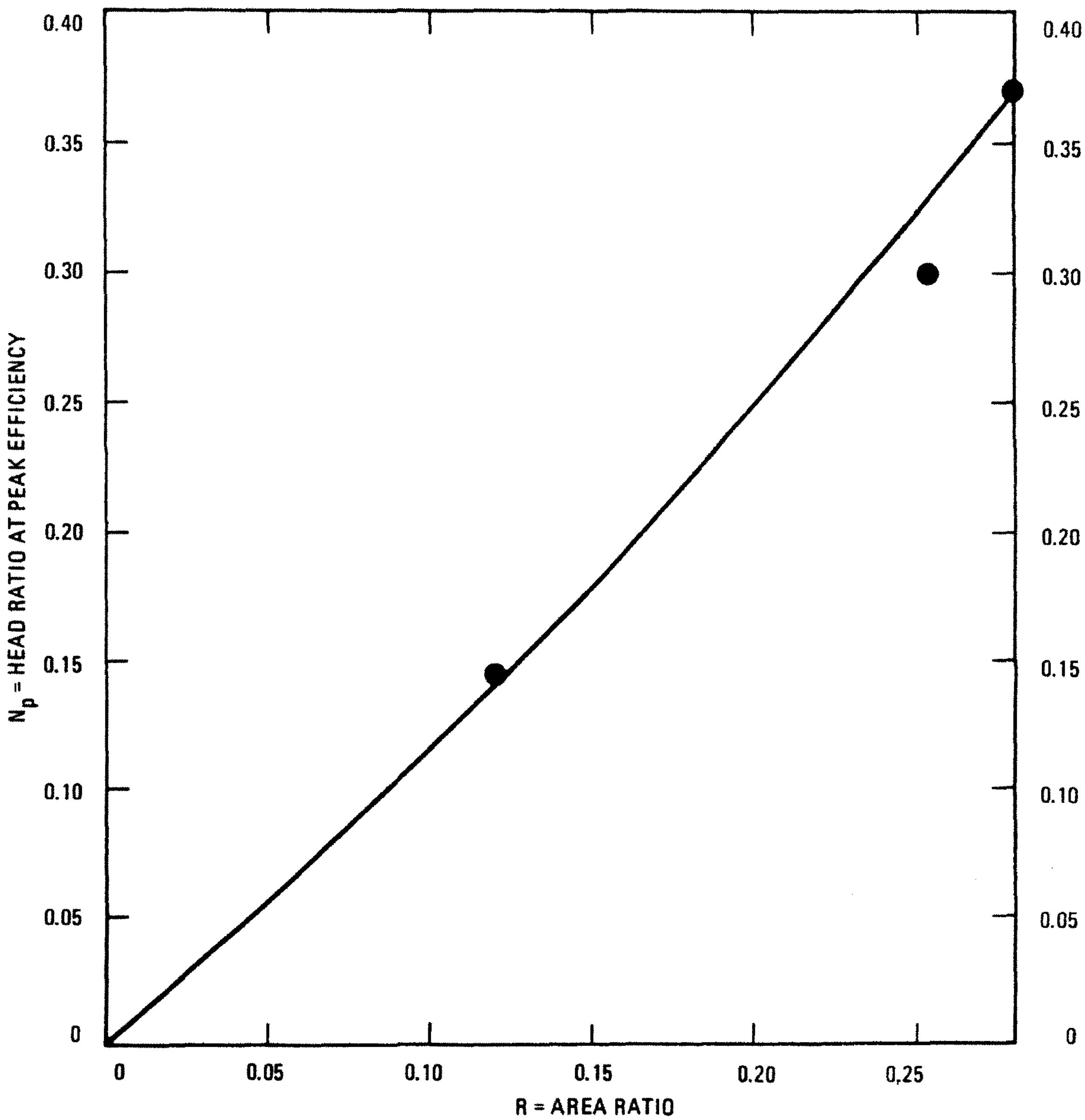
QUAD CITIES STATION UNITS 1 & 2
JET PUMP EFFICIENCY VS. M-RATIO
FIGURE 5.4-3



QUAD CITIES STATION
UNITS 1 & 2

JET PUMP CHARACTERISTIC CURVE

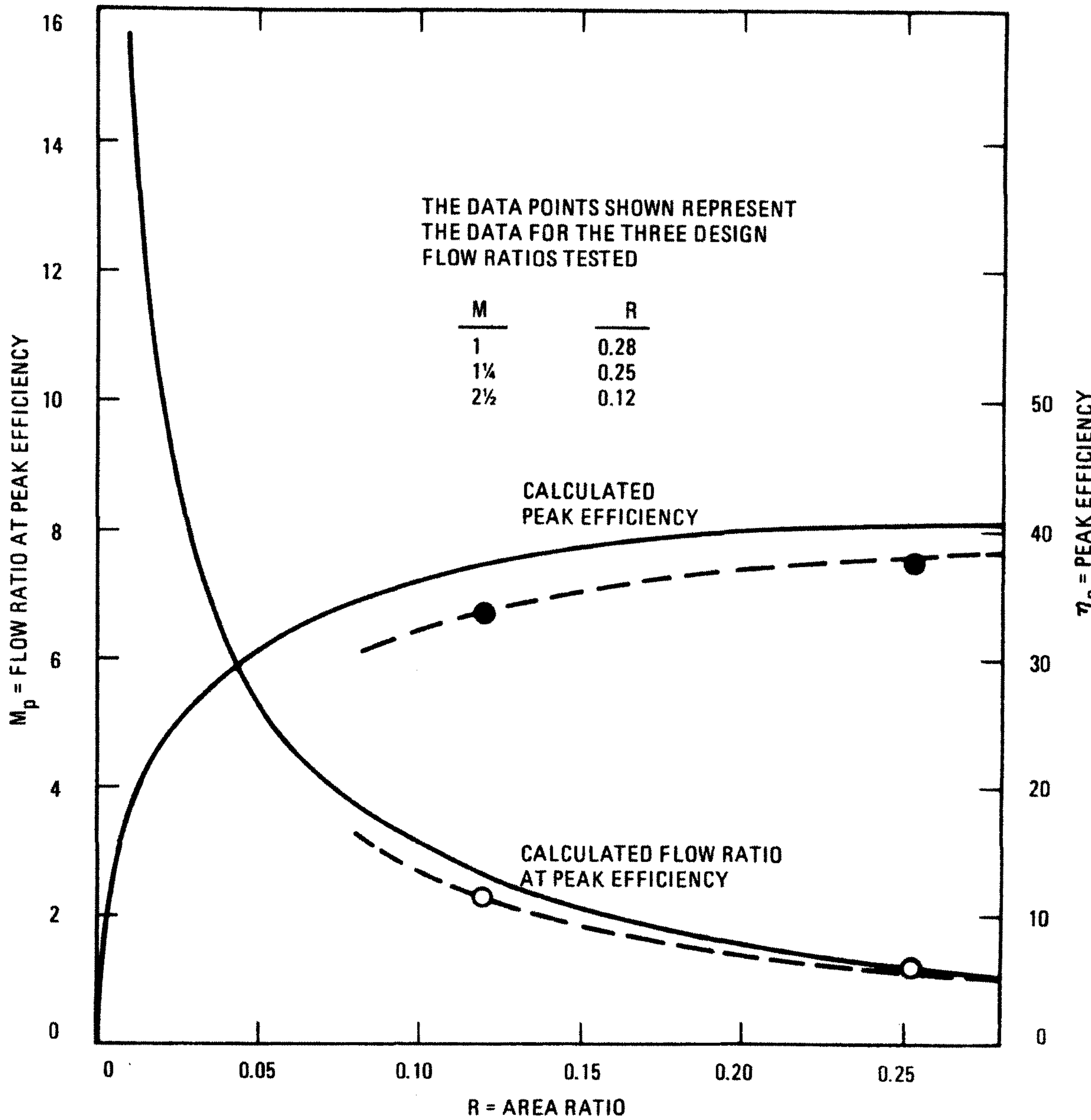
FIGURE 5.4-4



QUAD CITIES STATION
UNITS 1 & 2

JET PUMP HEAD RATIO VS. AREA RATIO

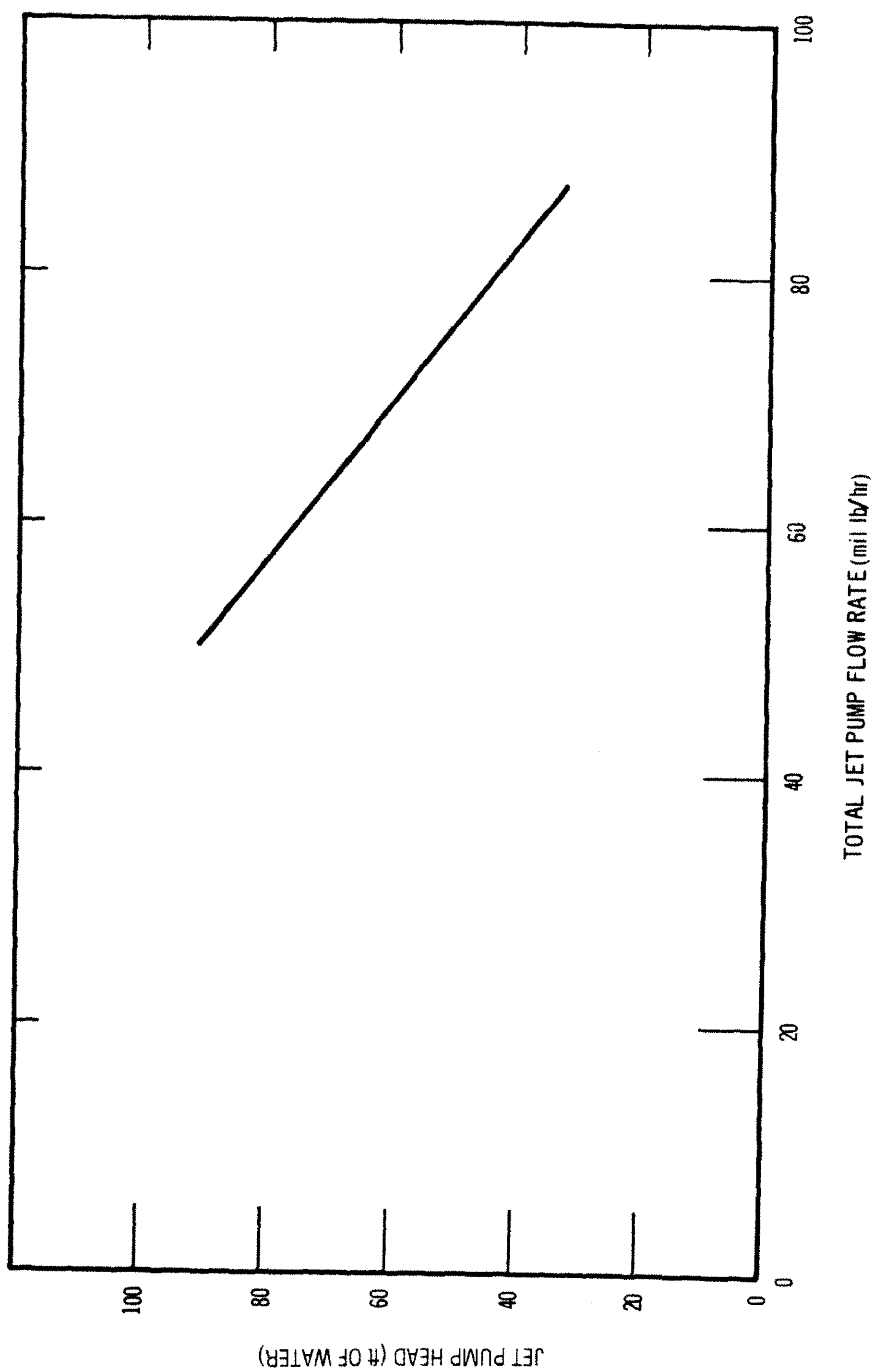
FIGURE 5.4-5



QUAD CITIES STATION
UNITS 1 & 2

JET PUMP FLOW RATIO VS. AREA RATIO

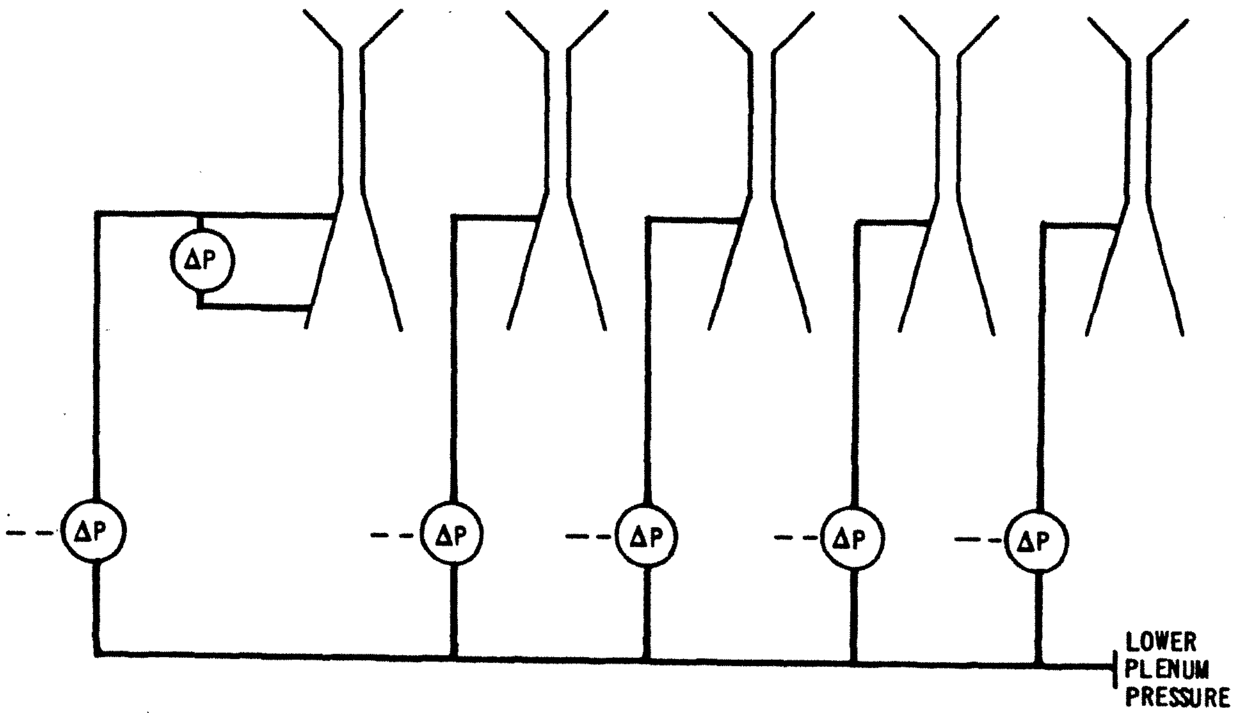
FIGURE 5.4-6



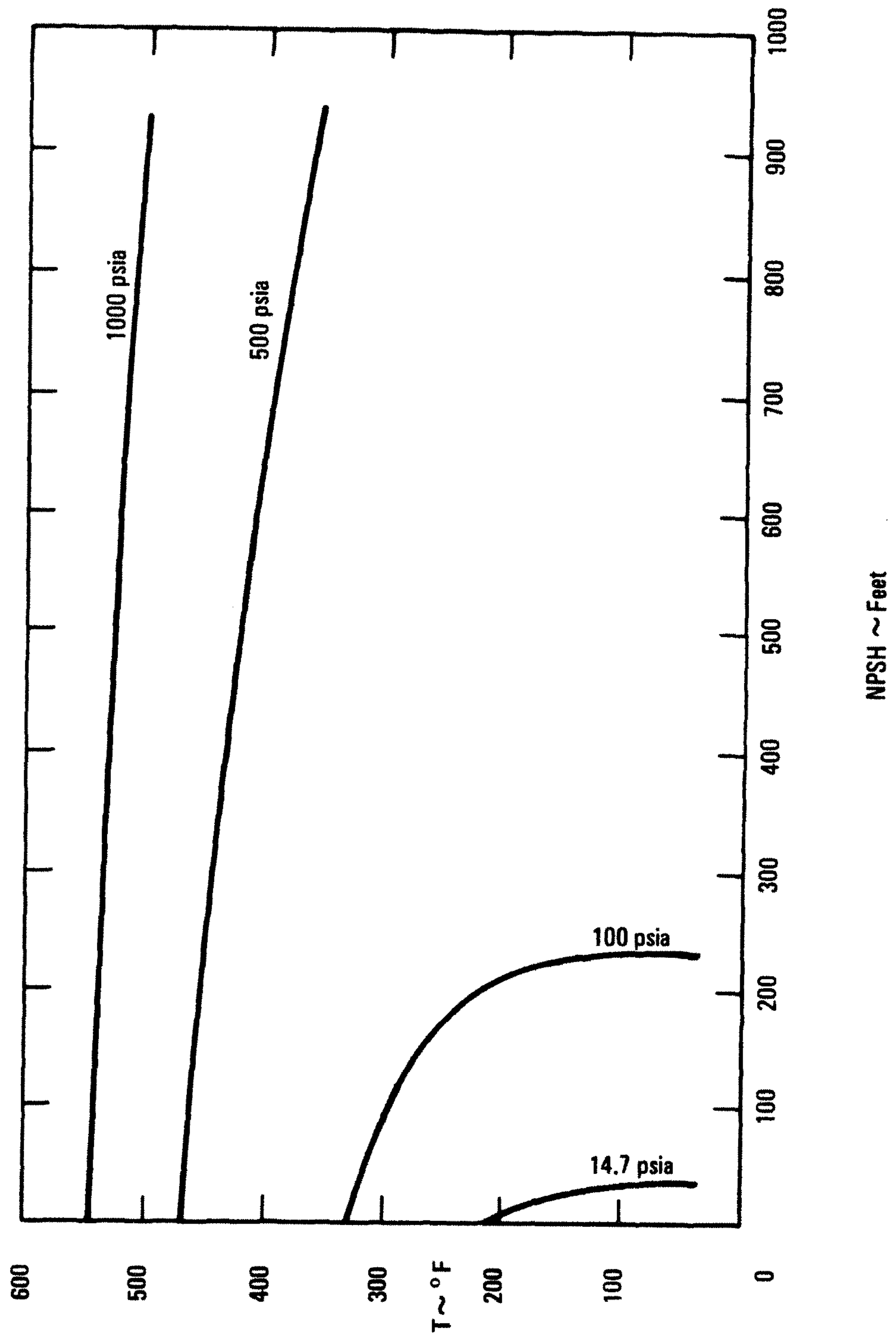
QUAD CITIES STATION
UNITS 1 & 2

TYPICAL JET PUMP HEAD CAPACITY CHARACTERISTIC

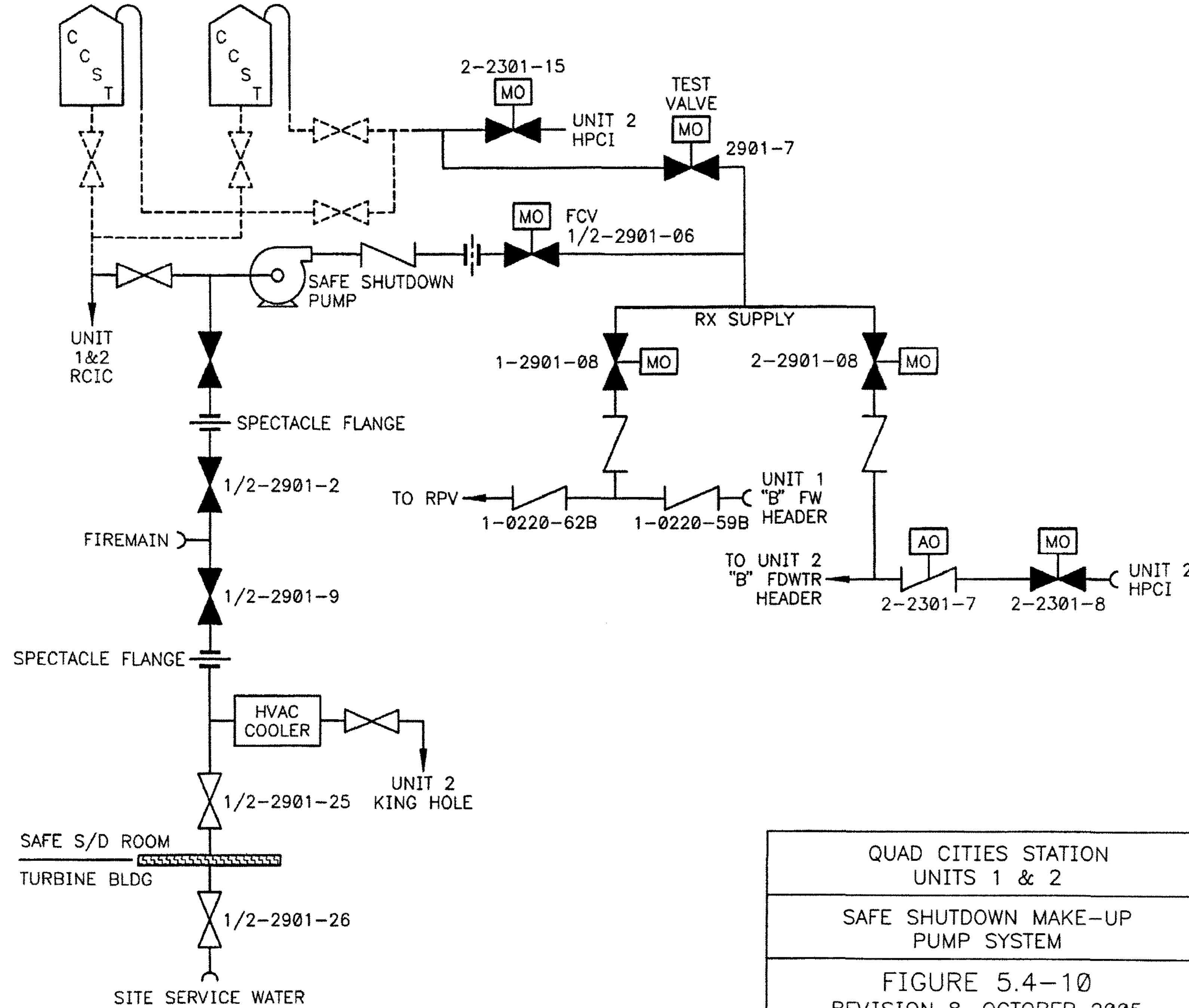
FIGURE 5.4-7



QUAD CITIES STATION UNITS 1 & 2
CORE FLOW MEASUREMENT SYSTEM SCHEMATIC
FIGURE 5.4-8 REVISION 4, APRIL 1997



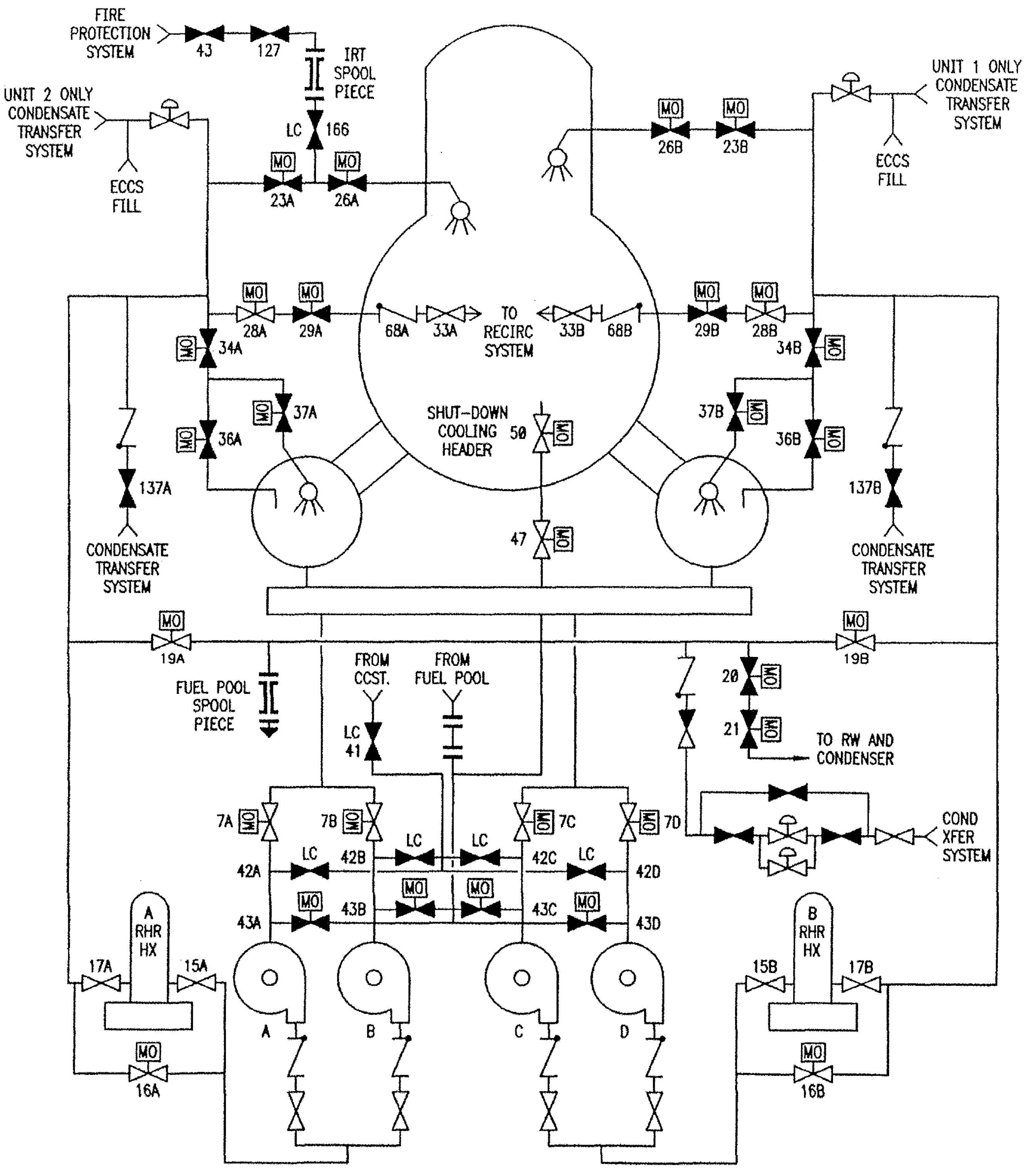
QUAD CITIES STATION UNITS 1 & 2
AVAILABLE NPSH UNDER VARIOUS OPERATING CONDITIONS
FIGURE 5.4-9



QUAD CITIES STATION
UNITS 1 & 2

SAFE SHUTDOWN MAKE-UP
PUMP SYSTEM

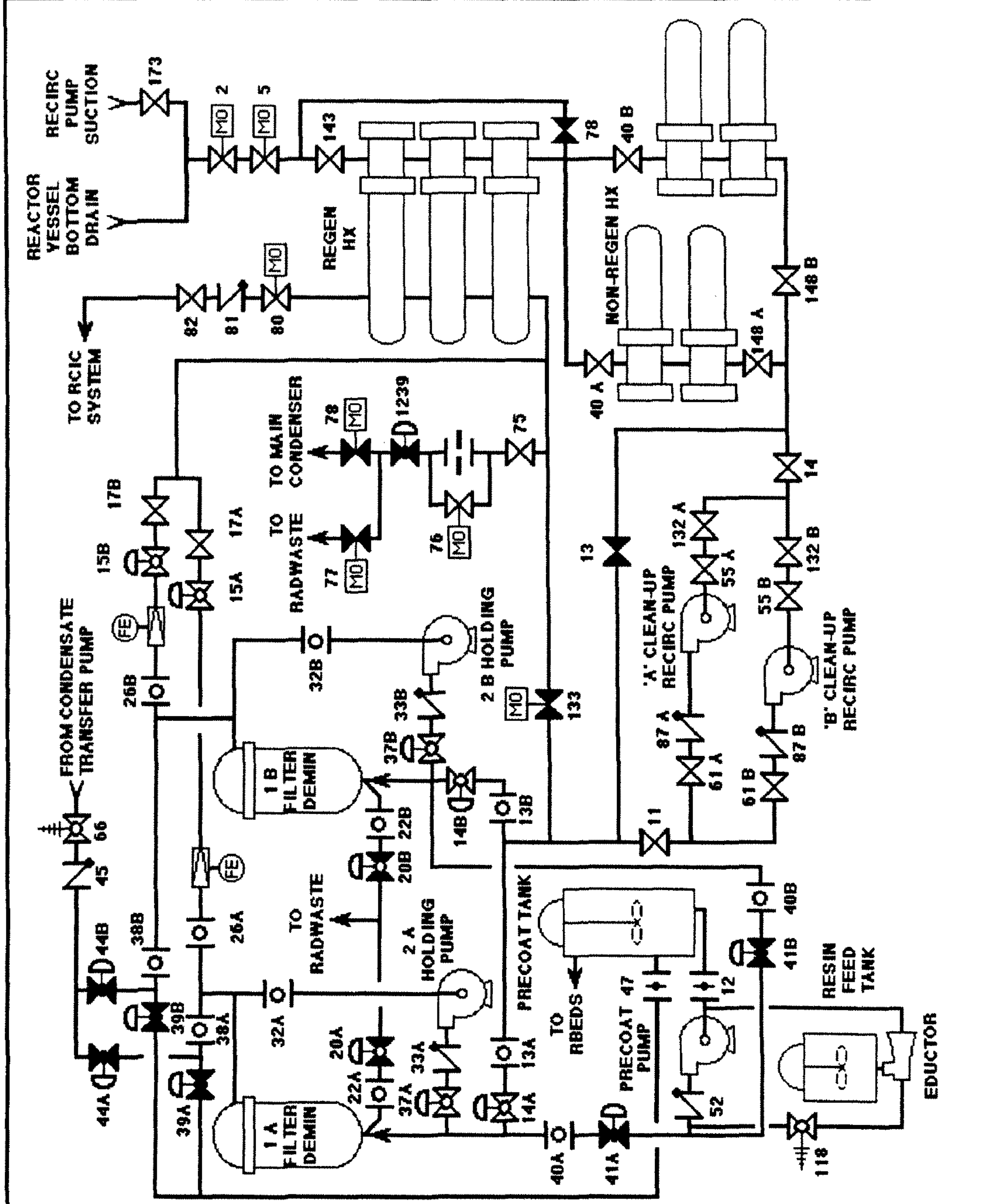
FIGURE 5.4-10
REVISION 8, OCTOBER 2005



QUAD CITIES STATION
UNITS 1 & 2

DIAGRAM OF RESIDUAL
HEAT REMOVAL (RHR) PIPING

FIGURE 5.4-11
REVISION 10, OCTOBER 2009

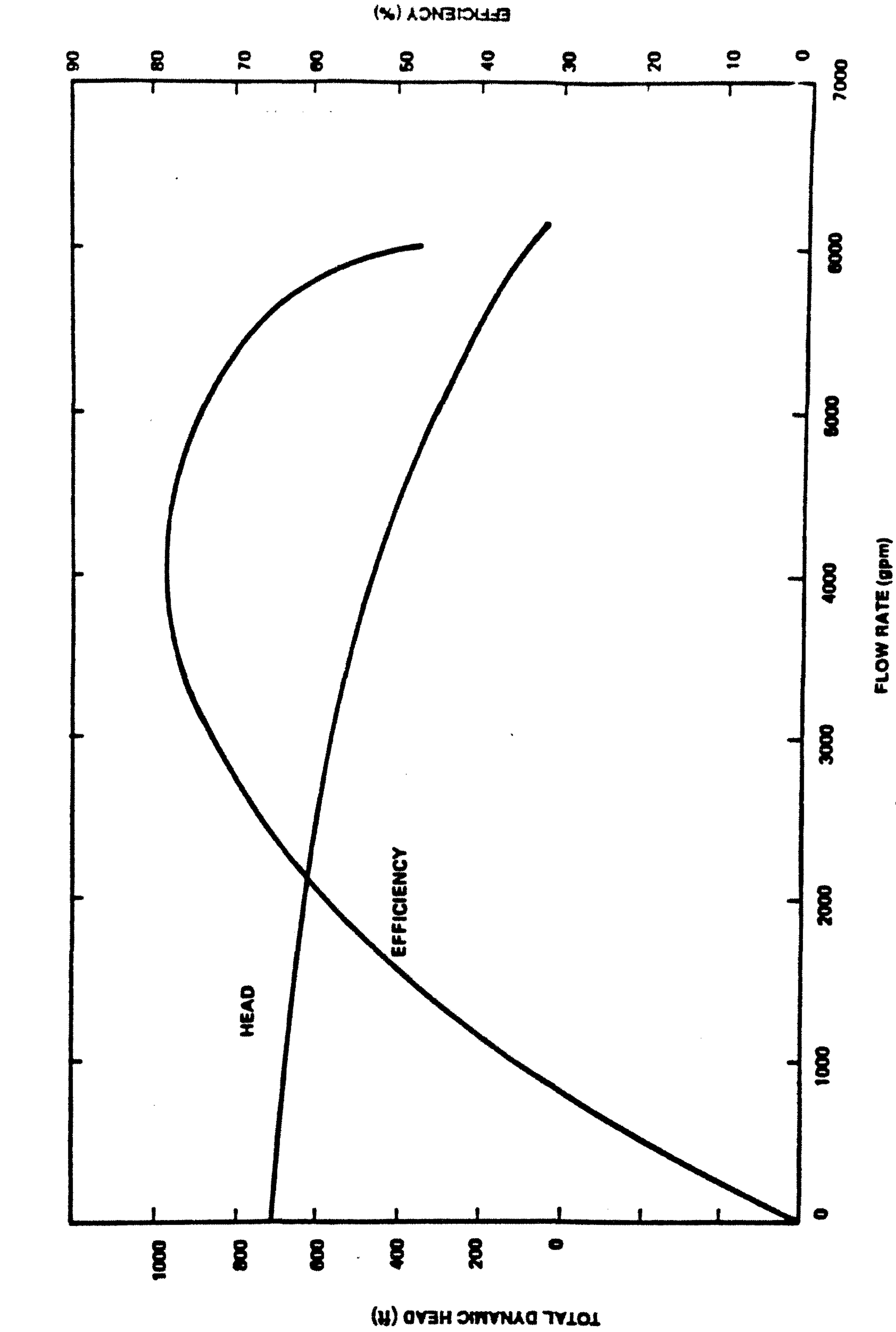


QUAD CITIES STATION UNIT 2

DIAGRAM OF REACTOR WATER CLEAN-UP PIPING

FIGURE 5.4-12

REV. 3, DECEMBER 1995



QUAD CITIES STATION
UNITS 1 & 2

LOW PRESSURE COOLANT INJECTION/CONTAINMENT COOLING SYSTEM PUMP CHARACTERISTICS

FIGURE 5.4-13

REV. 3, DECEMBER 1995