CHAPTER 5 - REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

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5.1 SUMMARY DESCRIPTION

The Reactor Coolant System includes the reactor primary system, relief valves, safety valves and the Isolation Condenser System. The reactor primary system includes the reactor vessel, recirculation piping, valves, pumps and all connected piping up to the first isolation valve. The reactor vessel contains the nuclear fuel, control rods, steam separators, dryers and other reactor internals, and water, which serves both as moderator and coolant. The Reactor Coolant System design data is presented in Table 5.1-1. Connected systems are listed in Table 5.1-2.

The reactor primary system is designed to contain the reactor core and coolant, facilitate reactor operation, and to provide a high integrity barrier against leakage of coolant and radioactive materials throughout the operating life of the plant. The reactor coolant is circulated past the core to cool it. Coolant parameters are continuously monitored to establish system operating conditions and provide input to systems designed to protect it.

The Reactor Recirculation System (Section 5.4) provides coolant flow through the core. Adjustment of the core coolant flow rate changes reactor power output, thus providing a means of matching plant load demand without adjusting control rod position.

Electromatic Relief Valves are provided to perform a dual function. In the overpressure protection mode, the system is designed to remove sufficient energy from the primary system to prevent the safety valves from opening; this is accomplished by the discharge of steam from the Reactor Coolant System to the pressure suppression pool. The system also acts to automatically depressurize the Reactor Coolant System in the event of a small break Loss-of-Coolant Accident (LOCA) and thus allow the low pressure Emergency Core Cooling System (ECCS) to supply sufficient cooling water, distributed in such a way, as to adequately cool the fuel. (Section 6.3)

Nine safety valves provide for pressure relief to the drywell to protect the Reactor Coolant Pressure Boundary, which is defined in Section 5.2, from damage due to overpressure, in the event of failure of all other pressure relief systems. The safety valves are discussed in Section 5.2.

The Isolation Condenser System (ICS) is described in Section 6.3, as part of the ECCS. The ICS is a passive high pressure system which consists of the two independent natural circulation heat exchangers that are automatically initiated by reactor vessel high pressure or low-low water level.

5.1.1 Piping and Instrumentation Diagram

The piping and instrumentation diagram of the Reactor Coolant System is presented in Drawing GE237E798.

5.1.2 Elevation Drawing

The elevation drawing for the Reactor Coolant System is presented in Drawing GE706E206.

TABLE 5.1-1
(Sheet 1 of 3)

REACTOR COOLANT SYSTEM DESIGN DATA

Reactor vessel		
Height (inside)	63 ft 10 in	
Internal diameter	17 ft 9 in	
Design pressure	1250 psig at 575°F	
Normal operating pressure	1020 psig at 548°F	
Maximum normal heatup and cooldown rates	100°F/hr	
Base metal material	SA-302 Grade B*	
Base metal thickness	7 1/8 in min	
Bottom head base metal thickness	8 ¾ in min	
Top head base metal thickness	4 5/16 in min	
Vessel design lifetime	60 yr	
Design maximum lifetime neutron flux (in energies greater than 1 Mev) for 60-yr life at 50 EFPYs	6.97×10^{18}n/cm^2	
Initial NDT temperature (base metal), opposite core	10°F	
other	40°F	
NDT temperature margin (between actual NDT and test temperature)	60°F	
	Expected	worst case
NDT temperature increase at 10^{19}nvt	70°F	265°F
NDT temperature increase at 1×10^{18}nvt	15°F	90°F
Cladding material weld deposited	E-308 electrode	
Cladding thickness	7/32 in nominal	
Nozzle materials	ASME SA-336 modified & SB-166	
Vessel and head flanges	ASME SA-336 modified	
Vessel closure stud material	ASME SA-193-B14 (AISI 4340)	
Vessel design code	ASME B&PV Code Section I & Case Interpretations 1270N & 1273N**	
Head to vessel O-ring material	Ag plated, Inconel 718	

* Modified by the addition of nickel (0.4 - 0.7 percent)

** Refer to Section 5.3.1.1

TABLE 5.1-1
(Sheet 2 of 3)

REACTOR COOLANT SYSTEM DESIGN DATA

Recirculation loops	
Number	5
Material	Stainless Steel
Design codes, piping	ASME B&PV Code Section I; ASA B31.1
Design pressure	1200 psig 570°F
Pipe Size	26 inches
Recirculation pumps	
Number	5
Type	Vertical, Centrifugal
Power rating	1000 HP
Full Load Speed	820 rpm
Flow rate	6400 to 36,000 gpm
Design water temperature	530°F
NPSH	37 ft water (for cold water with reactor cold and depressurized)
Outlet head	120 ft water
Casing material	SA-351 - CF8M
Casing design pressure	1300 psig, 575°F
Design code	ASA B31.1 & ASME Sect. VIII
Recirculation Loop Isolation Valves	
Number	10
Type	Motor Operated, Gate
Body material	SA-351 - CF8M
Design pressure	1200 psig, 575°F
Design code	ASME B&PV Code, Section I and Section VIII, plus G.E. Specification
Steam lines	
Number	2
Material	Seamless Carbon Steel
Design codes	ASME B&PV Code Section I (up to first isolation valve) ASA B31.1 (balance)
Main Steam Isolation Valves	
Number	2 per line
Type	wye pattern globe - air cylinder operated
Main Steam Isolation Valves	
Body Material	A 216 Gr WCB
Design closing time	3 to 10 sec
Design code	ASME B&PV Code, Section I; ASA B31.1

TABLE 5.1-1
(Sheet 3 of 3)

REACTOR COOLANT SYSTEM DESIGN DATA

Electromatic Relief Valves	
Number	5 (three on one steam header, two on the other)
Capacity	602,900 lb/hr each at 1250 psig
Pressure setting	See Technical Specifications
Design code	ASA B31.1 (original valves) ASME Sect. III (replacements)
Safety Valves	
Number	9 (5 on one steam header, 4 on the other)
Capacity	634,000 lb/hr each
Pressure setting	See Technical Specifications
Design code	ASME B&PV, Code, Section I ASA B31.1 (original valves) ASME Sect. III (replacements)
Feedwater Piping	
Design code	ASME Section I (up to first isolation valve) ASA B31.1 (balance)
Isolation Condensers	
Number	2
Design Capacity per isolation condenser (3 percent of 1930 MWt)205 x 10^6Btu/hr at 1000 Psig and 546°F	
Number of isolation	Two (2) normally open valves in inlet line (one ac operated, one dc operated)
Number of isolation	One (1) normally open valves in outlet line (ac operated) One (1) normally closed (dc operated)
Design codes shell	
Shell	ASME B&PV Code Section VIII
Tubes	ASME B&PV Code Section III C1.A
Design pressures	
Shell	15 psig internal 1 psig external, 300°F
Tube	1250 psig, saturated

TABLE 5.1-2
(Sheet 1 of 1)

CONNECTED SYSTEMS

System	Reference FSAR Subsections
Reactor Recirculation System	5.4.1
Shutdown Cooling System	5.4.7
Reactor Cleanup system	5.4.8
Main Steam System	5.4.9, 10.3.1
Feedwater System	5.4.9, 10.4.7
Head Cooling System	5.4.11
Liquid Poison System	9.3.5
Isolation Condenser System	6.3.1
Core Spray System	6.3.1
Control Rod Drive System	3.9.4
Reactor Vessel Instrumentation	7.6.1
Process Sampling System	9.3.2

5.2 INTEGRITY OF REACTOR COOLANT PRESSURE BOUNDARY

5.2.1 Compliance with Codes and Standards

Applicable codes and standards for various Reactor Coolant System components are shown in Table 5.2-1.

5.2.2 Overpressurization Protection

Overpressurization of the reactor primary system due to high reactor power is prevented by the control systems and the Electromatic Relief Valves (EMRVs). Overpressure protection is provided by the code safety valves. Design basis events are analyzed in Chapter 15. Specifically, overpressurization control and protection of the reactor primary system is prevented by:

a. High reactor pressure scram

b. High neutron flux scram

c. Operation of the Isolation Condenser

d. Operation of the Turbine Bypass System

e. Operation of the Electromatic Relief Valves

f. Operation of the code safety valves.

5.2.2.1 Design Basis

The principal design parameters for the reactor system are given in Table 5.2-1.

The nominal operating pressure of 1020 psig was chosen on the basis of economic analyses for boiling water reactors. The reactor vessel 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 design pressures for the piping and other primary system components are based on the reactor vessel design pressure with considerations for: (1) static and dynamic heads due to elevation and pump discharge pressure, and (2) overpressure allowances defined in the ASME Boiler and Pressure Vessel Code, Section I.

The design temperature for various system components varies according to the specific operating condition. The design temperature for the reactor vessel is based on the saturation temperature corresponding to the design pressure. Therefore, no specific system temperatures are designated as safety or operating limits.

In addition to the calculations required by the ASME Code, the vessel specification required additional stress analyses including stresses resulting from internal pressure, external and internal loadings, the effects of steady and fluctuating temperatures and loads. These analyses were conducted for regions involving changes of shape, structural discontinuities and points of concentrated loadings. The allowable stresses were those stated in the ASME Code. (See Section 5.3.)

Design analyses were also performed for cyclic conditions, with respect to material fatigue and aging management (Ref. 12). A listing of the plant transients and their allowable cycles is provided in Table 5.2-2. The two major means for computing and monitoring fatigue usage at fatigue critical locations are: 1) cycle-based fatigue (CBF) and 2) stress-based fatigue (SBF). Table 5.2-3 provides a listing of these critical locations and the methodology used. For all fatigue critical locations, the cumulative usage factor (CUF) is less than 1.0 or within the stress (intensity) allowable for a 60-year plant life.

It is noted that the original cyclic conditions were documented in Reference 1 of FSAR Section 5.3.4 (FDSAR Amendment 16). The number of heat-up and cool-down cycles were then reanalyzed later for reactor vessel studs and reactor vessel seal skirt due to their high fatigue usage factors and documented in Reference 10 to allow for higher number of cycles than expected in the original analysis. These components had the potential to exceed the allowable fatigue usage factor if the number of thermal cycles (e.g., heat-up / cool-down) exceeded the design assumptions. All other components had relatively low usage factors and were not expected to exceed the fatigue usage factor of 0.8 for the design life of 40 years. The design cycles were then compared with Bureau of Ships fatigue curves, which showed wide margins of safety in the design of the Reactor Coolant Pressure Boundary (RCPB).

As part of the License Renewal Application, Oyster Creek has revised the fatigue evaluation methodology, and the cumulative usage factor (CUF) acceptance criteria to be consistent with the ASME Section III criteria as allowed by ASME Section XI Non-Mandatory Appendix L. Accordingly, the allowable fatigue CUF was changed from 0.8 to 1.0.

To further assure the integrity of the RCPS, a comprehensive Inservice Inspection and Testing Program has been instituted.

5.2.2.2 Design Evaluation

Overpressure control and protection of the reactor vessel and main steam piping is provided by the 40 percent of rated steam flow Turbine Bypass System capacity, a 40 percent capacity set of the Electromatic Relief Valves (EMRVs), a set of spring loaded safety valves and the Isolation Condenser System. The bypass system is generally sufficient to relieve pressure transients in normal operating situations including full capacity turbine load rejection. The EMRVs will operate in case of failure of the bypass valves, turbine trip with the bypass valves available, closure of the Main Steam Isolation Valves (MSIVs), and failure of a feedwater controller. The spring loaded safety valves will open only in the event of MSIV closure, turbine trip without bypass valves available and failure of the Electromatic Relief Valves.

5.2.2.3 Piping and Instrumentation Diagrams

The overall boundaries of the Reactor Coolant System are shown on Figure 5.2-1. The Reactor Coolant System P&ID is presented as Drawing GE237E798.

5.2.2.4 Safety and Relief Valves

There are 14 safety and relief valves on the 24 inch main steam headers inside the drywell. Nine are spring loaded safety valves and discharge directly into the drywell. The remaining five are the Electromatic Relief Valves, controlled automatically by reactor pressure switches or manually from the Control Room, and discharge to the suppression pool. These valves are the main components of the Automatic Depressurization System (ADS) which is discussed in Section 6.3. The spring loaded safety valves will not open on any pressure transient except those resulting from isolation with scram failure, failure of the Electromatic Relief Valves, or failure of the turbine bypass valves.

5.2.2.4.1 Relief Valves

Upon rapid reactor isolation at power, the Isolation Condenser System is not designed to remove energy rapidly enough to prevent pressure relief through the safety valves. Thus, the primary system relief valves are provided to remove sufficient energy from the primary system to prevent the safety valves from opening. In the event of a small line break these valves will be actuated to reduce reactor pressure so that the Core Spray System can inject cooling water into the reactor vessel.

5.2.2.4.1.1 Relief Valve Sizing

For overpressure protection during power operation of the reactor, the relief valves are designed with sufficient capacity to preclude actuation of safety valves during normal operational transients under the following conditions at the plant:

- The reactor is operating at licensed core thermal power level.

- All system and core parameters are at values within normal operating range that produce the highest anticipated pressure.

  All components, instrumentation, and controls function normally.

- The relief valves open in 1.3 seconds and each valve has a capacity specified in table 6.3-2.

The Turbine Trip (with bypass) is analyzed with the above conditions to demonstrate the adequacy of the relief valves to preclude safety valve operation. The transient is described in Chapter 15, section 15.2.3.

5.2.2.4.2 Safety Valves

Nine safety valves provide reactor vessel overpressure protection for plant operations at the licensed core thermal power level of 1930 Mw. The safety valves are designed to limit the reactor vessel pressure to 110% of the design pressure during an inadvertent MSIV closure with reactor scram on a high neutron flux.

5.2.2.4.2.1 Safety Valve Sizing

The safety valves were analyzed for an inadvertent MSIV closure and a reactor scram on a high neutron flux. This analysis is performed each fuel cycle and the results reported in the supplemental reload licensing report. For analysis performed in previous cycles, the analysis used the Cycle 12 reload model with the RETRAN-02 Mod4 code. In order to assure conservative results for the peak pressures, the rainout velocity in the upper downcomer volume was increased from 3 feet per second to 1000 fps. Further, the analysis did not take credit for a recirculation pump trip, operation of the isolation condensers, or operation of the EMRV's during the transient (Reference 2).

For this transient, inadvertent MSIV closure, a reactor scram is designed for a MSIV closure of 10%. For the analysis, reactor scram is not assumed on MSIV closure or high pressure that would occur early in the transient. The reactor scram is assumed on a conservative setpoint (120%) for high neutron flux. For these assumptions, the analysis demonstrated that nine safety valves were acceptable (Reference 3), and limited the Cycle 12 peak reactor vessel and recirculation piping pressures to 1361 psig and 1367 psig respectively. These peak pressures are within the ASME code acceptance criterion of 1375 psig. The results of the safety valve sizing transient are presented in Figures 5.2-4 and 5.2-5.

5.2.2.4.2.2 Safety Valve and Relief Valve Accident Monitoring Instrumentation

The purpose of the safety/relief valve accident monitoring instrumentation is to alert the operator to a stuck open safety or relief valve which could result in a reactor coolant inventory reduction event. The primary detectors are acoustic monitors with thermocouples providing backup capability.

During 13R, the acoustic monitors removed for the safety valve reduction modification were relocated such that seven of the nine remaining safety valves now have two operational acoustic monitors. The relocated monitors are considered as backups or operational spares.

As the safety valves present distinctly different concerns than those related to the relief valves, the Technical Specifications separately discuss the actions to be taken upon inoperability. The actuation of a safety valve will be immediately detectable by observed increase in drywell pressure. Further confirmation can be gained by observing reactor pressure and water level. Operator action in response to these symptoms would be taken regardless of the acoustic monitoring system status. Acoustic monitors act only to confirm the closed/not closed status of the safety valve. In actuality, the operator actions in response to the lifting of a safety valve will not change whether or not the safety valve reclosed. Therefore, the actions taken for inoperable acoustic monitors on safety valves are significantly less stringent than those taken for the monitors associated with the relief valves.

Should an acoustic monitor on a safety valve become inoperable, setpoints on adjacent monitors would be reduced to assure alarm actuation should the safety valve lift. It is not important to the operator which valves lift, but only that one has lifted. Analyses, using very conservative blowdown forces and attenuation factors, show that reducing the alarm setpoint on adjacent monitors to less than 1.4g will assure alarm actuation should the adjacent safety valve lift.

5.2.3 Reactor Coolant Pressure Boundary Materials

5.2.3.1 Material Specification

The material specifications for various Reactor Coolant Pressure Boundary components are shown on Table 5.2-1.

5.2.3.2 Compatibility with Reactor Coolant

Materials in the Reactor Coolant System are primarily Type 316 and Type 304 stainless steels. The fuel cladding is zircaloy. The reactor water chemistry limits are established to prevent damage to these materials. Limits are placed on chloride concentration, pH and conductivity. The most important limit is that placed on chloride concentration to inhibit stress corrosion cracking of the stainless steels.

Figure 5.2-3 illustrates the results of tests on stressed 304 stainless steel specimens. The results of these tests were considered in the original design of the facility. Subsequent investigations have shown different results which are worth noting (see Reference 1). In the original test results (Figure 5.2-3), failures occurred at concentrations above the curve; no failures occurred at concentrations below the curve. According to the data, allowable chloride concentrations could be set at several orders of magnitude above the established limits, at the oxygen concentrations experienced during operation. Zircaloy does not exhibit similar stress corrosion failures. Other means of preventing stress corrosion cracking are constantly being evaluated.

A second chemistry control limit of importance is pH (limit range 5.6-8.6). This limit range is based on minimizing corrosive damage to system materials. The lower limit is the most important in that acidic conditions could produce the most corrosion. Since various reactors do operate near the upper pH limit without deleterious effects, the upper pH limit is not based on damage considerations; however, this upper limit was established as being a substantial departure from normal operation (approximately neutral pH) and so indicates a need to determine the reasons for such a change.

The lower pH limit was chosen to limit long term corrosion to a minimum and yet provide some latitude for recovery without requiring a sudden shutdown.

In this regard, it should be noted that a number of techniques are available to correct offstandard pH by:

a. removal of impurities by the Cleanup System

b. stop input of impurities causing offstandard condition

c. shutdown and cool reactor to ambient temperature. The major benefit here is to reduce the temperature dependent corrosion rates and to provide time for the Cleanup System to moderate the pH. The Cleanup System is described in Section 5.4.

Since all these steps take time, the pH at which shutdown action is initiated was selected as that for which minimal corrosion would occur.

Conductivity of the reactor water is more important a parameter to monitor in view of the fact that there are no chemical additions to the reactor coolant. Thus, conductivity is the only basic variable which is continuously monitored. When the conductivity is in its proper normal range, pH and chloride and other impurities affecting conductivity must also be within their normal range. When and if conductivity becomes abnormal, the pH and chloride measurements are made to determine whether or not they are also out of their normal operating values. This would not necessarily be the case as described below.

Conductivity could be high due to the presence of a neutral salt, e.g., Na_2S0_4, which would not have an effect on pH or chloride. In such a case, high conductivity alone is not a cause for shutdown. In the case of BWRs however, where no additives are used and where neutral pH is maintained, conductivity provides a very good measure of the quality of the reactor water. Significant changes in conductivity provide the operator with a warning mechanism which reflects the need to investigate and remedy the condition causing the change before limiting conditions are exceeded (with respect to variables affecting the boundaries of the reactor coolant).

5.2.3.3 Hydrogen Water Chemistry System

A permanent Hydrogen Water Chemistry (HWC) System is provided at OCNGS to mitigate the potential for IGSCC in the reactor coolant piping by reducing the dissolved oxygen level in the reactor coolant with the addition of hydrogen gas into the feedwater, however, this has reduced the dissolved oxygen level in the feedwater/condensate systems to below its normal range. A rotameter is utilized to bleed air into the suction of the "A" condensate pump to raise the dissolved oxygen to within limits. The hydrogen injection subsystem of the HWC System is located in the Reactor Feedwater Pump (RFP) Room and consists of a hydrogen gas supply header which is fed from the hydrogen storage facility, injection lines for each RFW Pump at the pump suction lines, and main control station and individual flow balancing valve/rotameter for each RFW Pump with all associated instrumentation and controls. To allow insitu calibration of the HWC flow elements, a vent line to the outdoors is provided. The vent line routes hydrogen to atmosphere outside of the Turbine Bldg. only when the flow element is calibrated, and at all other times the line will remain voided.

Hydrogen header and inlet pressure is monitored by locally mounted pressure switches and pressure gauges. High pressure turbine exhaust pressure is monitored by a locally mounted pressure transmitter. An in-line flow element and locally mounted flow transmitter are provided to monitor flow through each of two parallel flow-control trains. Control panel mounted flow indicator and flow recorder are provided to monitor hydrogen injection flow rate. In-line local rotameter flow indicators are provided downstream of each hydrogen branch line isolation valve to monitor and control hydrogen flow to each RFW pump. Controls are provided to allow the operator to control the hydrogen injection flow rate and system inlet and RFW pump injection isolation valves. Normally the flow controller automatically controls the hydrogen injection rate.

Hydrogen Water Chemistry can reduce the dissolved oxygen level in the feedwater condensate system to below its normal range. A rotameter is provided to bleed air into the suction of the "A" condensate pump to raise the dissolved oxygen to within limits.

Alarms are provided at the local control panel for high/low hydrogen inlet pressure, low hydrogen header pressure, high hydrogen area concentration and high hydrogen flow. These alarms are also transmitted to the Main Control Room as a hydrogen injection trouble alarm.

5.2.3.4 Noble Metal Chemical Addition

During the 1R19 Refueling Outage the NMCA injection process added a solution of platinum (Pt) and rhodium (Rh) noble metal compounds to the reactor and adjacent piping. After the plant returned to power, platinum and rhodium chemically bonded to reactor internal surfaces, which serve as catalyst sites for recombination of hydrogen and oxidants. The noble metal surfaces are chemically benign in the BWR environment and will have little effect on the bulk water concentration of hydrogen and oxygen during normal operation.

In a BWR reactor water environment elevated dissolved oxidant levels result in a very aggressive environment for Stress Corrosion Cracking (SCC) crack growth rates. SCC crack growth rate is a function of the dissolved oxidant levels in the coolant. With the Hydrogen Water Chemistry (HWC) process excess of hydrogen is added to the reactor water. HWC suppresses the dissolved oxygen by causing the recombination of hydrogen and oxidants back into water. As dissolved oxidant levels are decreased the SCC crack growth rate decreases.

NMCA treatment of internal structural materials increases the effectiveness of HWC by acting as a surface catalyst and providing recombination sites for hydrogen and oxidants on the coated surfaces. Therefore, HWC along with NMCA requires less hydrogen to produce the same ECP. This combination improves radiation dose rates in the Main Steam System and other plant secondary systems.

Laboratory samples of specimens treated in autoclaves showed the presence of noble metal atoms to a depth of 300 to 500 Å in oxidized stainless steel material. The noble metal is chemically bound to the oxide film of the wetted component surfaces. Surface scans of autoclave specimens have shown that the noble metal atoms present on the surface do not completely cover the surface but are distributed randomly across the surface. Consequently, the surface is not plated and the Pt/Rh layer is discontinuous.

On an atomic scale, the deposited noble metals may be discontinuous. Based on GE laboratory data, if gaps larger than 0.1-1.0 mm in the noble metal coverage exist, they will not be protected locally. If cracks develop in these regions; however, the lower ECP of the adjacent noble metal coated regions will arrest the cracks after a microscopic amount of crack growth into the noble metal protected region.

Cracks present in reactor vessel internals components prior to NMCA addition will be SCC mitigated in the presence of excess stoichiometric hydrogen. Data from over 60,000 hr. of tests on 17 different crack growth specimens that were first IGSCC pre-cracked and then NMCA treated in the unloaded condition, indicate that deposited noble metal arrests an existing IGSCC crack. Further studies have revealed that noble metal deposited in the mouth of the crack or crevice is sufficient to mitigate IGSCC and is effective even after the noble metal is removed from the bulk surfaces.

The effectiveness of the NMCA is monitored the Noble Metals Monitoring SYSTEM (NMMS) which contains an ECP probe and A Durability Monitor. The ECP probe monitors electrochemical corrosion potential of reactor coolant water and the Durability provides a means of simulating and trending noble metals the deposition.

5.2.4 Inservice Inspections and Testing of Reactor Coolant Pressure Boundary

By letter dated April 16, 1992 (C321-92-2119), GPUN informed NRC of approved code cases, and specific ASME XI relief requests, which are part of the Inservice Inspection Program Update for the third ten-year interval. This update meets the requirements of the 1986 edition of ASME XI with no addenda. The Inservice Inspection Program is currently under review by the NRC staff.

In addition to the Inservice Inspection Program, piping weldments made of austenitic stainless steel are subject to augmented inspections for Intergranular Stress Corrosion Cracking (IGSCC). These augmented inspections are performed on stainless steel piping greater than four inches in nominal diameter and which contains reactor coolant above 200°F during power operation. Details of the OCNGS program are found in Reference 3, Section 3.9.7

5.2.5 Detection of Leakage Through Reactor Coolant Pressure Boundary

5.2.5.1 Description

The reactor primary coolant Leakage Detection Systems are significant means of preventing primary system boundary failure by identifying leaks before failures occur (leak-before-break).

There are two distinct means of detecting leakage: monitoring of water influx to drains/sumps, and monitoring of changes in Containment atmosphere. Six indicators of leaks inside the drywell are monitored: equipment drain pump flow, floor drain pump flow, drywell temperature rise, drywell pressure rise, drywell humidity rise and floor sump fill time. In addition, containment atmosphere is monitored for changes in noble gas and particulate concentrations which could be indicative of leakage.

The OCNGS Technical Specifications contain specific requirements for monitoring of Reactor Coolant System leakage.

5.2.5.1.1 Equipment Drain Tank and Floor Drain Sump

The primary means of detecting leaks inside the drywell are the Drywell Equipment Drain Tank (DWEDT) pump flow and the floor drain sump fill time. All unidentified leakage is routed into the floor drain sump while only identified or normal leakage such as pump seal leakage and valve gland leakage, drains to the equipment drain tank. Drains and sumps are discussed in detail in Section 9.3.

The design basis for the sump pumps located in the Reactor Building floor is to accommodate minor leakage. The size of the sumps and pumps are for expected operation based on past experience. These pumps are not sized for an amount of water resulting from a catastrophic break from a large water source system. The drywell floor drain sump and the DWEDT have a capacity of approximately 375 gallons each. The drywell floor drain sump is equipped with two sump pumps, each of which is rated at 50 gpm. The DWEDT is also equipped with two pumps rated at 50 gpm each.

The floor drain sump receives the condensation from the drywell air coolers and therefore will be the primary sump to measure both steam and reactor coolant leaks.

Leakage which remains liquid drains by gravity to the floor drain sump, and leakage which flashes to vapor is condensed by the air cooler and then also drains to the floor drain sump. There are many potential sources of leakage into the drywell atmosphere, such as flanges and safety valves, which do not affect plant safety; however, until visual inspection of such leakage allows reclassification, all leakage into the floor drain sump must be considered unidentified leakage. Any increase in the flow into the DWEDT is positively due to increased seal and gland leakages only, which by definition is identified leakage. (Recirculation pump seal failure is detected in this way.) The only source of leakage inside the drywell, besides the primary system, is the Reactor Building Closed Cooling Water (RBCCW) System. The failure modes and effects analysis of the RBCCW System is presented in Section 9.2.

The floor drain sump is equipped with alarms which actuate if the sump fill rate exceeds a predetermined value (see Section 9.3) as well as with integrating flow meters which allow day by day monitoring of the total pumpout. The sump and drain tanks are equipped with two pumps for total pumpout. The sump and drain tanks are equipped with two pumps for redundancy. The alarms are actuated by level switches and the sump control includes a timer which measures the time between successive pump operations and sets off an alarm for abnormal operation. The pump activates on high sump level and pumps out until the sump water is reduced to the low trip point, then pump out stops. Condensate cannot leave the sump unless the pump is running so that leakage simply collects and raises the level until the high level trip is reached again. The time required to raise the level as a function of leak rate into the sump is calculated from the flow-volume relationship for the sump.

If this time period is shorter than that corresponding to some preselected average pump flow rate (as set by the timer), thus indicating excessive flow into the sump, an alarm trips. The exact setting of the trip on the floor drain sump will vary as the identified innocuous leakage into the floor drain sump varies based on previous operating experience. The reactor will be shutdown before leakage rates exceed Technical Specification limits.

Additional information on drainage systems and sumps can be found in Subsection 9.3.3.

5.2.5.1.2 Drywell Temperature, Humidity and Pressure

As an additional qualitative backup to the floor drain sump, the drywell atmospheric conditions are monitored. Because the drywell is a closed, relatively compact vessel, the drywell temperature, pressure and humidity respond promptly to leaks from the Reactor Coolant System, thus providing the operator with additional information on leaks. However, this method is not quantitative since fluctuations in atmospheric conditions are normally expected, and specific measurements are not possible. Nevertheless, the increase in these parameters does serve to alert the operator of a potentially abnormal condition. Calculations show that a detectable increase in dewpoint temperature should occur for steam leaks equivalent to about 2 gpm of condensate, and for liquid breaks of 5 gpm. The dewpoint is continuously recorded and will be periodically checked. However, dewpoint is less sensitive to liquid leaks because about 60 percent of the leakage may not flash and, therefore, does not affect the atmospheric conditions.

An increase in drywell temperature (5-10°F) as well as pressure (0.5-0.7 psi) will occur in the event of a 5 gpm steam leak, or a 10 gpm reactor coolant liquid leak. Drywell temperature and pressure are both monitored in the Control Room and the temperature is also recorded. Slightly larger leaks will result in an alarm due to high drywell pressure. At a steam leak of about 12 gpm, the reactor will scram due to high drywell pressure in less than about 30 minutes from the onset of the leak.

5.2.5.1.3 Containment Atmosphere Particulate and Gaseous Radioactivity Monitor System

The Containment Atmosphere Particulate and Gaseous Radioactivity Monitor System provides a diverse means of reactor coolant system leak detection by detecting the release of radioactivity from a leak and subsequent flashing to steam. The system is designed to detect both particulate and noble gas radiation. Due to the lack of direct relationship between absolute activity level, and leak rate, the information provided by CAPGRMS is qualitative and not quantitative but may alert operators to changing radiological conditions in the drywell.

The CAPGRMS continuously samples containment atmosphere for a fixed period and measures the activity levels for both particulates and noble gases. The measured countrates are compared to expected values stored in the CAPGRMS microprocessor. The CAPGRMS activates a Control Room annunciator when either the particulate or noble gas channel activity increases at a rate which exceeds the expected increase from a previous sample period, or exceeds a preestablished setpoint for either channel. The CAPGRMS is equipped with Control Room annunciation of system malfunction.

The CAPGRMS is qualified to operate before and after the Safe Shutdown Earthquake and provides no postaccident function.

5.2.5.2 Detection of Large Pipe Breaks

An analysis of the feedwater, steam flow and reactor water level instrumentation would indicate a large break to the operator in the Control Room. This would be in the order of 5 percent of feedwater flow at rated power. If there is excessive flow through the main steam venturi flow limiter (Subsection 5.4.4), a primary system isolation is initiated. This condition is alarmed in the Control Room.

Another indication of a large main steam leak, in addition to the flow limiter, is an alarm which compares steam flow at the flow limiter, the turbine first stage pressure, and steam flow to the reheaters. This alarms a mismatch of about 7 percent (Discussed in Section 7.7.).

Leakage through the safety or relief valves can be determined by reading thermocouples whose indications are outside of the drywell. The thermocouple sensors are located (one each) at a point downstream of each of the 14 valves. (See Subsection 5.2.2.4.)

An increase in differential pressure between the sensing tap on the core spray pipe near the reactor vessel and the other tap indicating pressure above the core plate would detect a break in the core spray pipe between the vessel and shroud or else a break in the drywell between the reactor vessel and the testable check valve.

5.2.6 Reactor Coolant System Vents

The reactor coolant system vents are designed to allow the venting of large quantities of noncondensible gases that can be generated within the reactor following core damage. Vent paths are provided from the reactor cooling system and reactor head to ensure that non-condensible gases cannot accumulate in the core to the point where core cooling would be interrupted and further core damage would occur. The two locations for the venting are: (a) the Isolation Condensers' steam line high points and (b) the top of the reactor vessel.

5.2.6.1 Isolation Condensers High Point Vents

In the present configuration, the Isolation Condensers can be vented to the main steam header downstream of the Main Steam Isolation Valves. The high points of the main steam lines are vented through electromatic relief valves. The Isolation Condenser High Point Vents are isolated during a LOCA. However, these vents can be remotely opened by operator action post LOCA. The NRC has determined this manual action acceptable (Reference 5) for compliance with 10CFR50.44(c)(3)(iii). For a detailed description of the Isolation Condenser System, refer to FSAR Section 6.3.

5.2.6.2 Reactor Head Vent

The reactor head vent (see Figures 5.2-1 and 5.2-2) does not perform a specific safety function other than to maintain the reactor pressure boundary. Noncondensables are vented from the top of the reactor vessel to the 24 inch diameter main steam lines during reactor operation. The vent path to the Drywell Equipment Drain Tank is closed during reactor operation using valves V-25-21 and V-25-22. These valves form part of the reactor pressure boundary, but are not required to operate to safely shut down the reactor. The valves are located near the top of the reactor vessel, under the Drywell cap. The control panel for these valves is located outside the Drywell and will have the key locks on the operating switches. Also, the fuse for the valve power supply will be removed to prevent the vent valve operation during reactor operation.

The valves are classified as seismic Category I, and are capable of maintaining their isolation function inside the inerted Drywell (275°F and 3.5×10^7 rads integrated dose during normal operation). The valves, however, are not required to be operable under normal plant operating conditions (they are not to be energized except when the reactor is at atmospheric pressure). All other electrical components are also suitable for use inside the Drywell.

5.2.7 References

(1) Gordon, B.M., "Materials Performance," 19, No. 4, pp 29-38, 1980.

(2) Reference deleted.

(3) Reference deleted.

(4) Reference deleted.

(5) Safety Evaluation by the Office of NRR Relating to High Point Vents for the Isolation Condenser, dated April 24, 1986.

(6) GPUN Safety Evaluation, SE 402915-001, "Safety Valve Reduction".

(7) Safety Evaluation by the Office of NRR Related to Amendment No. 150, dated March 6, 1991.

(8) GPUN Topical Report, TR-101, Considerations Associated with Changing EMRV Setpoints.

(9) Safety Evaluation by Office of NRR Related to Amendment 177, dated February 21, 1995.

(10) GPUN Safety Evaluation, SE-000221-004, “Reactor Vessel Thermal Cycles”

(11) 50.59 Evaluation, OC-2006-E-001, Revise Method for the Determination of Fatigue Cumulative Usage Factor for ECR 06-00046.

(12) 50.59 Screening OC-2007-S-056, “Issue New Design Analyses for License Renewal Project,” ECR 05-00365.

TABLE 5.2-1
(Sheet 1 of 3)

REACTOR COOLANT SYSTEM DATA

Reactor vessel		
Height (inside)	63 ft 10 in	
Internal diameter	17 ft 9 in	
Design pressure	1250 psig at 575°F	
Normal operating pressure	1020 psig at 548°F	
Maximum normal heatup and cooldown rates	100°F/hr	
Base metal material	SA-302 Grade B*	
Base metal thickness	7 1/8 in min	
Bottom head base metal thickness	8 3/4 in min	
Top head base metal thickness	4 5/16 in min	
Vessel design lifetime	60 yr	
Design maximum lifetime neutron flux (in energies greater than 1 MeV) for 60-yr life at 50 EFPYs	6.97 x 10^{18} n/cm^2	
Initial NDT temperature (base metal), opposite core	10°F	
other	40°F	
NDT temperature margin (between actual NDT and test temperature)	60°F	
	expected	worst case
NDT temperature increase at 10^{19} nvt	70°F	265°F
NDT temperature increase at 1 x 1018 nvt	15°F	90°F
Cladding material weld deposited	E-308 electrode	
Cladding thickness	7/32 in. nominal	
Nozzle materials	ASME SA-336 modified & SB-166	
Vessel and head flanges	ASME SA-336 modified	
Vessel closure stud material	ASME SA-193-B14 (AISI 4340)	
Vessel design codes	ASME B&PV Code Section I & Case Interpretations 1270 N & 1273 N**	

* Modified by the addition of nickel (0.4 - 0.7 percent)
** Refer to Section 5.3.1.1

TABLE 5.2-1
(Sheet 2 of 3)

REACTOR COOLANT SYSTEM DATA

Reactor Vessel	
Head to vessel O-ring material	Ag plated, Inconel 718
Recirculation loops	
Number	5
Material	Stainless Steel
Design codes	ASME B&PV Code Section I; ASA B31.1
Design pressure	1200 psig, 570°F
Pipe Size	26 inches
Recirculation pumps	
Number	5
Type	Vertical, Centrifugal
Power rating	1000 HP
Full load speed	820 rpm
Flow rate	6400 to 36,000 gpm
Water temperature	530°F
Design NPSH	37 ft water (Cold Water with Reactor Cold and Depressurized)
Outlet head	120 ft water
Casing material	SA-351 - CF8M
Casing design pressure	1300 psig, 575°F
Design codes	ASA B31.1 & ASME Section VIII
Recirculation loop isolation valves	
Number	10
Type	Motor Operated, Gate
Body Material	SA-351 - CF8M
Design pressure	1200 psig, 575°F
Design code	ASME B&PV Code, Section I and Section VIII, plus G.E. Specification.
Steam lines	
Number	2
Material	Seamless Carbon Steel
Design codes	ASME B&PV Code Section I (up to first isolation valve) ASA B31.1 (balance)

TABLE 5.2-1
(Sheet 3 of 3)

REACTOR COOLANT SYSTEM DATA

Main Steam line isolation valves	
Number	2 per line
Type	wye pattern globe - air cylinder operated
Body material	A216 Gr WCB
Closing time	3 to 10 sec.
Design code	ASME B&PV Code Section I ASA B31.1
Electromatic Relief Valve	
Number	5 (three on one steam header two on the other)
Capacity	602,900 lb/hr each at 1250 psig
Pressure setting	See Technical Specifications
Design code	ASA B31.1 (original valves) ASME Sect. III (replacements)
Safety Valves	
Number	9 (5 on one steam header, 4 on the other)
Capacity	634,000 lb/hr each
Pressure setting	See Technical Specifications
Design code	ASME B&PV Code, Section I ASA B31.1 (original valves) ASME III (replacements)
Feedwater Piping	ASME Section I
Design Code	(up to second isolation valve) (first valve outside of the drywell) ASA B31.1 (balance)
Isolation Condensers	
Number	2
Capacity per isolation condenser (3 percent of 1930 MWt)	205 x 10^6 Btu/hr at 1000 psig and 546°F
Number of isolation valves in inlet line	Two (2) normally open (one ac operated, one dc operated)
Number of isolation valves in outlet line	One (1) normally open (ac operated) One (1) normally closed (dc operated)
Design codes	
Shell	ASME B&PV Code Section VIII
Tubes	ASME B&PV Code Section III C1.A
Design pressures	
Shell	15 psig internal 1 psig external, 300°F
Tube	1250 psig, saturated

TABLE 5.2-2
(Sheet 1 of 1)

TRANSIENTS AND ALLOWABLE CYCLES

Transient No.	Description	Counting Type	Allowable Number of Cycles
1 / 2	Vessel Head Removal & Reinstallation (Boltup/Unbolt)	Manual	80
3	Design Pressure Test (Leak Test)	Automatic	130
4	Heatup = Normal Startup (100°F/hr)	Automatic	240
5	Turbine Roll and Increase to Rated Power	Automatic	240
6	Cooldown = Normal Shutdown (100°F/hr)	Automatic	240
8	Hot Standby (Feedwater Cycling)	Automatic	400
9	300°F/hr Emergency Cooldown	Automatic	5
10	Safety Relief Valve (SRV) Blowdown	Automatic	1
11	SCRAM	Automatic	200
16	Turbine Trip	Automatic	40
17	Loss of Feedwater Heaters	Automatic	80
18	Interruption of Feedwater Flow	Automatic	80
19	Overpressure to 1,250 psig	Manual	1
20	Overpressure to 1,375 psig	Manual	1
21	Hydrostatic Pressure Test (Code Hydro Test to 1563 psig)	Manual	3
22	Core Spray Injections	Automatic	10
23	Electromagnetic Relief Valve (EMRV) Actuation	Automatic	"A" – 450 "B" – 450 "C" – 450 "D" – 450 "E" – 450
27	Shutdown Cooling Operation *	Automatic	Not Specified
28	Isolation Condenser Actuation*	Automatic	Not Specified
29	Unisolation of an Isolated Loop*	Automatic	6500

* Note: These events were not in the original design bases documents. It has since been decided to count these events automatically via the FatiguePro cycle counting logic.

TABLE 5.2-3
(Sheet 1 of 1)

FATIGUE MONITORING LOCATIONS AND METHODS

NO.	COMPONENT	LOCATION	METHOD
1	RPV	Closure Region (Bolts)	CBF
2		Bottom Head (Vessel-Head Junction)	CBF
3		Support Skirt (Transition Taper Top)	CBF
4		Recirculation Inlet Nozzle	CBF
5		Recirculation Outlet Nozzle	CBF
6		RPV Core Spray Nozzle, Safe End (304SS Point 20)	CBF
7		RPV Core Spray Nozzle, Nozzle Forging (302B Point 35)	CBF
8		Basin Seal Skirt to Vessel Flange Junction	CBF
9	Torus	Downcomer/Vent Header Intersection	CBF
10		Vent Line/Drywell Intersection	CBF
11		Torus Shell	CBF
12		Torus Shell at SRV-Supporting Ring Girder	CBF
13		SRV-Supporting Ring Girders	CBF
14		SRV Piping Penetration on Vent Pipe	CBF
15		Vent Header Ring Collar	CBF
16		Nozzle: Drywell to Torus Vacuum Relief (Torus End)	CBF
17		Attached Piping: Vacuum Relief-Type 2	CBF
18		Penetrations: Isolation Condenser	CBF
19	Isolation Condenser	Bounding Location	CBF
20	Feedwater	Nozzles	SBF

Notes:
1) CBF - Cycle Based Fatigue
2) SBF - Stress Based Fatigue

5.3 REACTOR VESSEL

5.3.1 Reactor Vessel Materials

5.3.1.1 Introduction

The Oyster Creek reactor vessel was originally designed to ASME Boiler & Pressure Vessel (B&PV) Code Section I, 1959 Edition through Winter 1963 Addenda with Code Cases 1270N and 1273N and GE Specification 21A1105. The reactor vessel was fabricated, inspected and tested in accordance with the ASME Boiler and Pressure Vessel Code, Section I - Power Boilers, 1962 Edition and Addenda plus the Nuclear Code cases applicable on December 11, 1963. The vessel purchase specification, dated January 22, 1964 further directed the use of Section VIII - Unfired Pressure Vessels where Section I did not cover specific details.

In addition to the ASME Code sizing calculations, a detailed stress analyses were performed which covered both steady state and transient conditions with respect to material fatigue.

ASME B&PV Code Section III - Nuclear Vessels Code was not originally used, as this Section was not available to the general public until after the contract date for purchase of the reactor vessel. The vessel was ordered in December, 1963 and Section III was not released until 1964.

For reactor pressure vessels designed and built prior to the adoption of the ASME Boiler and Pressure Vessel Code Section III the General Electric Company developed a method for performing a fatigue analysis which would provide assurance that vessels installed in General Electric designed nuclear power plants would safely withstand all anticipated operating and transient conditions, both normal and emergency. This original method was based upon the method of analysis developed for Naval reactors and upon industry's experience using it. This method was defined by and was a specific requirement of the design specification for the Oyster Creek reactor vessel, in addition to the requirements for compliance with Section I, Section VIII, and applicable Nuclear Code Cases. Section III now includes the requirements of the code cases, and the method of analysis with only minor changes. However, the allowable stresses under Section III are greater in the areas of concern than were permitted by Sections I and VIII and the fatigue curves used in the design specification. Therefore, the use of Section I and Section VIII plus the Nuclear Code Cases plus the General Electric Specification defined analysis results in a completed vessel for the Oyster Creek plant which has safety margins that are generally equivalent to those which would result from using Section III.

As part of the License Renewal project, Oyster Creek has revised the fatigue evaluation methodology, and the cumulative usage factor (CUF) acceptance criteria to be consistent with the ASME Section III criteria as allowed by ASME Section XI Non-Mandatory Appendix L (Ref. 9 & 10). Accordingly, the allowable fatigue CUF was changed from 0.8 to 1.0. The Code reconciliation was performed to ASME Section III, 1995 Edition with Addenda through 1996. For further details on fatigue, transients and cycles considered, see Section 5.2.2.1.

There were no deviations to the formal codes throughout the design, fabrication, inspection and testing of the reactor vessel. Details of the analyses, tests, inspections and materials are presented in Reference 1.

Thermal stresses occur in a vessel when two segments or areas are at different temperatures. The thermal stresses and strains on the reactor vessel or reactor system which can result from system operation are limited in order to prevent fatigue or distortion of the vessel. The thermal strains are controlled by limitations on allowed heatup and cooldown rates, and limitations on cold inlet water temperatures (and thermal sleeves). The expected thermal strains on the vessel are analyzed and included in the fatigue analysis. It was concluded from the fatigue analysis that the vessel could withstand thermal strains beyond those anticipated during normal operation. In fact, conditions well beyond the design limits would be required to actually produce vessel failure. (Refer to Section 5.2 for anticipated operational cycles).

A discussion of conformance with pertinent Regulatory Guides is presented in Section 1.8.

5.3.1.2 Fracture Toughness

To ensure that ferritic components of pressure retaining components of the Reactor Coolant Pressure Boundary exhibit adequate fracture toughness under service hydrostatic tests and during heatup and cooldown, temperature and pressure limitations are established in the Technical Specifications (refer to Subsection 5.3.2). Beltline materials for the vessel were ordered with a reference nil ductility transition temperature (RT_{NDT}) of +10F or less.

5.3.1.3 Material Surveillance

The General Electric Company developed and provided for an irradiation surveillance program for the Oyster Creek reactor vessel. A series of mechanical test specimens from the base metal of the reactor vessel and from weld heat affected zone metal, and weld metal taken from a weld joint made from the reactor steel and simulating a welded joint in the reactor vessel were selected for the program at Oyster Creek. Specimens were placed in the reactor vessel close to the vessel wall to be exposed to conditions similar to that of the vessel wall. Both tensile specimens and Charpy specimens have been provided for all the above materials and locations. Various wires and materials are used to measure the integrated flux to which the specimens have been exposed. Selected groups of the specimens in the vessel are removed at recommended intervals over the life of the reactor and are tested to determine changes in mechanical properties, and effects on operational parameters (see Subsection 5.3.2).

The following reactor pressure vessel steel surveillance specimens from the actual vessel were placed in the Oyster Creek reactor vessel: Twelve impact specimens and one specimen each of iron, nickel and copper flux wire were placed in each impact capsule.

Two tensile specimens were placed in each tensile capsule. Groups of these capsules are held in baskets. Three of these baskets are hung on the wall of the vessel at the core midplane, spaced 120 degrees apart. The number of specimens is detailed in Table 5.3-1.

Material for base metal specimens has been taken from a plate used in vessel beltline regions or from a plate of the same heat of material. The same plate used for base metal specimens is used for production of heat affected zone specimens, and the weld specimens are produced by the identical weld practice and procedures used in the vessel fabrication. Thus, the surveillance specimens do represent the materials and processing of the vessel beltline region.

The steps taken during the production of BWR pressure vessel surveillance specimens assure reasonable representation of the vessel material. Any variations in irradiation behavior between the surveillance materials and additional heats of vessel materials is expected to be minimal.

In 2003, the NRC approved Oyster Creek's participation in the BWR Vessel and Internals Project (BWRVIP) Integrated Surveillance Program (ISP) as described in BWRVIP-78 and

BWRVIP-86-A, “BWR Vessel and Internals Project, Updated BWR Integrated Surveillance Program (ISP),” Final Report, October 2002. The ISP meets the requirements of 10CFR50 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 10CFR50 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 is based on BWRVIP-86-A. Based on this schedule, Oyster Creek is not scheduled to withdraw an additional material specimen.

5.3.2 Pressure - Temperature Limits

The requirements of 10CFR50, Appendix G, "Fracture Toughness Requirements," establish that pressure temperature limits be established for Reactor Coolant System heatup and cooldown operations, inservice leak and hydrostatic tests, and reactor core operation. These limits are required to ensure that the stresses in the reactor vessel remain within acceptable limits, and to provide adequate margin against brittle fracture. They are intended to provide adequate margins of safety during any condition of normal operation, including anticipated operational transients.

The pressure temperature limits depend upon the metallurgical properties of the reactor vessel materials. The properties of materials in the vessel beltline region vary over the lifetime of the vessel because of the effects of neutron irradiation. One principal effect of the neutron irradiation is that it causes RT_{NDT} to increase with time. The pressure-temperature operating limits must be modified periodically to account for this radiation induced increase in RT_{NDT} by increasing the temperature required for a given pressure. The operating limits for a particular operating period are based on the material properties at the end of the operating period. By periodically revising the pressure temperature limits to account for radiation damage, the stresses and stress intensities in the reactor vessel can be held within acceptable limits. At the beginning of life, material other than that in the beltline region may be the limiting material because it is subjected to high stresses and stress intensities. However, since material outside the beltline region is not subjected to high level irradiation, its RT_{NDT} will not change as the beltline region will and at some period of life, the beltline materials will become limiting.

The magnitude of the shift in RT_{NDT} is proportional to the neutron fluence to which the materials are exposed. The shift in RT_{NDT} can be predicted from the results of tests on material surveillance specimens or from the guidance contained in Regulatory Guide 1.99, Rev 2.

Revised operational limits are periodically submitted to the NRC and incorporated into the Technical Specifications upon approval. Operating limits are in accordance with 10CFR50, Appendix G. Conformance with Appendix G in establishing safety operating limitations ensures adequate safety margins during operation, testing and maintenance and constitutes an acceptable basis for satisfying the requirements to NRC General Design Criterion 31 10CFR50 Appendix A.

5.3.3 Reactor Vessel Integrity

5.3.3.1 Design

The reactor vessel is a vertical cylindrical pressure vessel. The base plate material is high strength alloy carbon steel SA-302, Grade B modified by the addition of nickel. The vessel interior is clad with Type 304 stainless steel applied by weld overlay.

Fine grained steels and advanced fabrication techniques were chosen to minimize radiation effects. The water in the annulus between the core shroud and the vessel reduces radiation levels at the vessel wall to a very low integrated exposure for the lifetime of the plant.

The head closure is designed for easy removal and reassembly. The head is bolted to the vessel with high strength studs. A double "O" ring type seal is provided, and the area between the seals can be monitored for leakage.

The control rod drive housings and the in-core instrumentation thimbles are welded to the bottom head of the reactor vessel.

Steam outlets are from the vessel body, thus eliminating the need to break flanged joints in the steam lines when removing the vessel heads for refueling. Relief valves and safety valves are mounted on nozzles in the main steam lines. The five Electromatic Relief Valves discharge directly to the torus while the safety valves discharge to the drywell.

The reactor vessel is supported by a steel skirt. The top of the skirt is welded to the bottom of the vessel. The base of the skirt is continuously supported by a ring girder fastened to a concrete foundation, which carries the load through the drywell to the Reactor Building foundation slab.

Stabilizer brackets, located below the vessel flange, are connected to tension bars with flexible couplings. The bars are then connected through the drywell to the concrete structure outside the drywell to limit horizontal vibration and to resist seismic and jet reaction forces. The bars are designed to permit radial and axial expansion.

Additional details of the reactor vessel design, materials, fabrication methods, inspection and shipment and installation are contained in References 1 through 8. Inservice surveillance is addressed in Subsection 5.2.4.

The review of reactor vessel integrity per the Systematic Evaluation Program for the Oyster Creek Nuclear Generating Station is presented in Section 1.10.

5.3.3.2 Reactor Vessel Repair Program

In the course of construction, during the field hydrostatic test of the Oyster Creek Plant reactor pressure vessel conducted on September 29, 1967, it was noted that a small leak emanated from an area near a control rod drive housing.

The primary system, including the pressure vessel, was then subjected to chemical cleaning after which personnel were sent into the pressure vessel to investigate the source of the leak.

A dye penetrant examination, conducted for the purpose of finding the source of the leak, revealed that the leak was the result of welding flaws and incomplete fusion in the field weld made to join the control rod drive housing to the stub tube of the pressure vessel. Probe grinding on this weld indicated that the lack of fusion penetrated the weld and that it was responsible for the leak observed during hydrostatic test.

Extension of dye penetrant examination to other areas of the stub tubes and pressure vessel revealed that a number of fine cracks and flaws existed on the outer surface of the stub tubes.

An immediate program was initiated to investigate these indications, determine the cause of this condition, and perform repair procedures that would fully establish the required reactor vessel integrity. Details of this program can be obtained from References 2 through 8.

5.3.4 References

(1) Oyster Creek Nuclear Power Plant, FDSAR, Amendment 16, Reactor Pressure Vessel Design Report, September 1967.

(2) Oyster Creek Nuclear Power Plant, FDSAR, Amendment 29, Status Report on Reactor Vessel Repair Program, December 1967.

(3) Oyster Creek Nuclear Power Plant, FDSAR Amendment 35, Final Report on Reactor Vessel Repair Program, March 1968.

(4) Oyster Creek Nuclear Power Plant, FDSAR Amendment 36, Reactor Vessel Repair Program, March 1968.

(5) Oyster Creek Nuclear Power Plant, FDSAR Amendment 37, Reactor Vessel Repair Program Additional Information, April 1968.

(6) Oyster Creek Nuclear Power Plant, FDSAR Amendment 40, Report on Reactor Vessel Repair Program, August 1968.

(7) Oyster Creek Nuclear Power Plant, FDSAR Amendment 43, Supplemental Report Reactor Vessel Repair Program, October 1968.

(8) Oyster Creek Nuclear Power Plant, FDSAR Amendment 47, Post Hydro Examination, October 1968.

(9) Letter from U.S. NRC to Mr. Christopher M. Crane, “Oyster Creek Nuclear Generating Station (OCNGS) – Issuance of Amendment Re: Use of Integrated Surveillance Program for Reactor Vessel Specimen Surveillance (TAC NO MB7005),” dated April 27, 2004.

(10) 50.59 Evaluation, OC-2006-E-001, Revise Method for the Determination of Fatigue Cumulative Usage Factor for ECR 06-00046.

TABLE 5.3-1
(Sheet 1 of 1)

OYSTER CREEK VESSEL SPECIMENS IN REACTOR[b]

No. of Basket No.	No. of Impact Spec.				No. of Tensile Spec.				Flux Wires		
	Base	Weld	HAZ	Std[a]	Base	Weld	HAZ	Std[a]	Fe	Ni	Cu
1	12	12	12	--	3	2	3	--	6	3	6
2	9	9	9	9	2	2	2	2	3	3	3
3	8	8	8	--	2	2	2	--	2	2	2

(a) Humboldt base metal.

(b) Capsule Number 1 was removed in 1971 and not replaced. Capsule Number 2 was removed from the reactor during the Cycle 10 refueling outage and analyzed to establish the 15 Effective Full Power Years Pressure-temperature curves (refer to Technical Specifications); a replacement capsule containing specimens from Capsule Number 2 will be installed.

(c) The RVMSP Capsule was installed at the location for basket No. 2 during the Cycle 14 refueling outage with the following inventory:

5	Melt Wire Thermal Monitors
3	Sets of Flux Wires (minimum Fe and Cu)
18	Miniature Fracture Toughness Specimens (Base)
12	Irradiated Charpy Impact Specimens (Base)
12	Irradiated Charpy Impact Specimens (Weld)
12	Irradiated Charpy Impact Specimens (HAZ)
15	Irradiated Charpy Impact Half-Specimens (Base)
15	Irradiated Charpy Impact Half-Specimens (Weld)
3	Miniature Irradiated Tensile Specimens (Base)
3	Miniature Irradiated Tensile Specimens (Weld)
9	Standard Unirradiated Tensile Specimens (Base)
15	Unirradiated Charpy Impact Specimens (Base)
15	Unirradiated Charpy Impact Specimens (Weld)

5.4 COMPONENT AND SUBSYSTEM DESIGN

5.4.1 Reactor Recirculation System

5.4.1.1 Design Bases

The Reactor Recirculation System has been designed to perform the following functions:

a. To provide forced circulation of reactor water through the core to overcome the power density limitation of the fuel.

b. To provide a variable moderator (coolant) flow through the core to control reactor power without manipulation of the control rods.

The Reactor Recirculation System has been sized to provide a total flow capacity equal to the required flow at rated load. The original design total flow capacity at rated load was 61 Mlbm/hr. The Reactor Recirculation System has an actual capacity that exceeds the original design value and been analyzed for operation up to 67.5 Mlbm/hr (see Section 15.10, Reference 25). The design pressure of the recirculation pumps is 1300 psig, with a design temperature of 575°F. The system piping and valves have been designed for a pressure of 1200 psig and a temperature of 570°F.

5.4.1.2 System Description

The system consists of the reactor vessel and five piping loops, as shown in Drawing GE237E798. Each loop comprises one motor driven pump, a motor generator (M-G) set, suction and discharge valves, a bypass valve around each discharge valve, pipe support hangers, piping, and associated system controls and instrumentation.

Recirculated coolant enters the lower head of the reactor through vessel nozzles, passes through the diffuser and orifices at the bottom of the core and flows upward through the core where bulk boiling produces steam. The steam-water mixture enters the moisture separators and then the steam dryers. The water separated from the steam flows downward across the top of the plenum, where it mixes with the incoming feedwater, and enters the downcomer annulus between the shroud and the vessel wall. The coolant flows through the downcomer region, through the outlet nozzles, and into the recirculation pumps suction piping. The coolant is then returned to the vessel via the pumps and discharge piping.

The continuous circulation ensures that hot spots are not created by steam bubbles, which would result in steam blanketing around fuel rods and in reduction of the heat removal capability of the coolant. To control reactor power level, the system makes use of the boiling water reactor large negative power coefficient. A power level increase is achieved by increasing the recirculation flow, which reduces core voids and thereby increases reactivity in the core. As power increases, the boiling rate increases and the void fraction increases, adding negative reactivity and stopping the power increase. The core then stabilizes at a high power level. Decreased flow results in the opposite effect.

Reactor operation is permitted with up to two inoperable recirculation loops within the following constraints:

a. No more than one loop is isolated (suction, discharge and discharge bypass valves are closed) at any time during power operation.

b. The isolated loop is not restarted until the plant is in the cold shutdown condition.

c. Power is limited to 90% of rated when there are two inoperable recirculation loops.

5.4.1.2.1 Recirculation Pumps

The Recirculation Pumps (Drawing GE107C5339) are vertical, single stage, centrifugal pumps driven by 1000 horsepower motors. With five pumps in operation, each pump will deliver up to approximately 36,000 gpm at 1017 psig and 520°F.

The pump motors are 2400 volts, open drip-proof, three phase, squirrel cage induction type, especially designed for operation on a variable frequency power supply with an operating range between 11.5 and 57.5 cps.

Oil lubricated double acting thrust bearings are provided for the motors. Cooling is provided by the Reactor Building Closed Cooling Water (RBCCW) System, at a rate of approximately 4 gpm, to an internal oil cooler.

The pump seal assembly for each pump consists of two sets of cartridge type mechanical seals and a breakdown bushing. Under normal operating conditions, each seal provides approximately 500 psi pressure drop and forms two cavities from which the pressures are measured. Restrictive orifices control leakage to approximately 0.5 gpm. Seal cooling is supplied to the shell side of the seal heat exchanger by the RBCCW System at approximately 25 gpm. An auxiliary impeller, mounted on the main shaft, circulates coolant through the tube side of the heat exchanger to the seal cartridge.

5.4.1.2.2 Motor Generator Sets

Five variable frequency motor generator (M-G) sets are provided to control the speed of the five Recirculation Pumps. The M-G sets are located at ground level in a separate room adjacent to the southwest corner of the Reactor Building. The electrical output of each generator is directly tied to its associated Recirculation Pump motor in the drywell. Each M-G set consists of a horizontal induction motor driving a synchronous generator through an adjustable fluid coupling.

Generator drive is provided by one 1250 hp, 4160 volt, air cooled motor for each of the M-G sets. The major components of the fluid coupler are an impeller, a runner and a scoop tube. The coupler transmits power from the drive motor to the variable speed generator by a vortex of oil.

The impeller is coupled directly to the input shaft and rotates at a constant speed of 1180 rpm. The runner is directly coupled to the output shaft and to the generator. The runner rotates at a speed determined by the quantity of oil in the vortex and load conditions. The output shaft can operate at any speed between 230 rpm and 1160 rpm without causing overheating or oil foaming.

The scoop tube controls the quantity of oil in the vortex and, in turn, the amount of power transmitted from the drive motor to the generator. The oil acts as the coolant as well as the power transmission medium. An oil cooler is provided to ensure safe operating temperatures, and receives cooling water from the Turbine Building Closed Cooling Water System. Positioning of the scoop tube can be accomplished from the Control Room by means of an electropneumatic positioner, or manually at the M-G Set. The positioning mechanism will lockup the scoop tube on loss of air supply or upon failure of the instrument circuit.

5.4.1.2.3 Valves

The recirculation loops contain valves for isolating any pump from the Reactor Coolant System. The 26 inch motor operated suction gate valves have a full port opening of 26 inches. These are designed to open against a 50 psi differential pressure which is equivalent to the static head of water in the reactor vessel.

The discharge valves are 26 inch motor operated gates with a reduced port opening of 24 inches. These are designed to open against a pressure of 100 psid. The discharge bypass valves are provided to serve as a low flow bypass during pump startup and as a pressure equalizing valve across the suction and discharge valves. The bypass valves are 2 inch motor operated and designed to open against full reactor pressure.

The stroking rate of the recirculation loop valves is approximately 12 inches per minute, resulting in approximate closure times of: 2 minutes and 20 seconds for the suction valves, 2 minutes for the discharge valves, and 10 seconds for the discharge bypass valves.

The suction and discharge valves of the recirculation loops have a direct impact on the communication of reactor coolant between the reactor downcomer region and the reactor core region. If the suction and discharge valves of all five recirculation loops are closed, a water level reduction within the reactor core region will not result in a corresponding water level reduction within the reactor downcomer region. The instruments that detect low-low reactor water level are located within the reactor downcomer region. The closed valves will isolate the flowpath between the reactor downcomer region and the reactor core region. For this reason, the suction and discharge valves of at least one recirculation loop shall remain in the full-open position when 1) there is irradiated fuel in the reactor pressure vessel and 2) the reactor coolant temperature is greater than 212ºF. There are two exceptions to the full-open valve/recirculation loop configuration.

The first exception is when the reactor water level is greater than 185 inches above TAF and the reactor coolant temperature is less than 212ºF. With the reactor water level greater than 185 inches above TAF, the reactor coolant will communicate between the reactor downcomer region and the reactor core region. Any decrease in reactor water level will be detected by the downcomer instruments and will allow for appropriate operator action.

The second exception is when the steam operator and steam dryer are removed from the reactor pressure vessel and the reactor coolant temperature is less than 212ºF. With the removal of the stem separator and steam dryer, the reactor coolant will communicate between the reactor downcomer region and the reactor core region to below the Core Spray System actuation setpoint (low-low reactor water level) of 86” above TAF. Also, Plant Technical Specification No. 2.1.D ensures that reactor water level will be maintained 4’-8” above TAF.

The discharge and suction valves have a direct impact on the communication between the reactor vessel downcomer and core levels. With these valves open, changes in core water level result in corresponding changes in the vessel's downcomer water level. To insure ECCS actuation on Lo-Lo level, at least one suction and discharge valve in a recirculation loop must be open. The exception to this is in the refueling mode with level above 185 inch TAF and temperature less than 212°F and no work is being done which could cause water level to be reduced to less than 4' 8" above the top of active fuel.

5.4.1.2.4 Supports

The recirculation piping, pumps and drive motors are suspended from constant support hangers to reduce thermal expansion stresses. Travel of the hangers is greater than calculated thermal growth.

5.4.1.2.5 Piping and Nozzles

The recirculation loop piping is all of Stainless Steel welded construction and has been designed, manufactured and constructed to meet, as a minimum, the requirements of the ASME B&PV code, Section I, and ASA B31.1 - Code for Pressure Piping.

There are five 26 inch loops, with a 2 inch bypass line around each pump discharge valve. The vessel outlet nozzle is 33 inches reducing to 26 inches and the vessel inlet nozzle is 26 inches in diameter.

The return lines from the Isolation Condenser System (Section 6.3) are connected to the suction side of recirculation loops A and E upstream of the suction valves. The supply line to the Shutdown Cooling System (SCS) is connected to the suction side of the recirculation loop E at the Isolation Condenser line. The return line from the SCS is connected to the discharge line of recirculation loop E, downstream of the discharge valve.

The supply line to the Reactor Cleanup System is connected to the suction line of recirculation loop B and the return is to the discharge line of the source loop.

5.4.1.2.6 Controls and Instrumentation

The Reactor Recirculation System is provided with a speed control unit. The unit consists of a pneumatic operator for each fluid drive scoop tube, an electric tachometer on each generator shaft, a remote manual speed controller for each M-G set (with speed and scoop tube position indicators), a master remote control device for all five pumps, and all necessary electronic equipment.

Operating speeds of all five pumps are normally adjusted in unison by the master speed controller. The individual speed controller is used for taking a pump out of service or returning it to normal operation. The tachometer on each generator shaft provides a feedback signal for comparison of actual versus selected speeds.

Instrumentation is provided for monitoring the recirculation pump motors, seals and seal cooling water parameters, loop temperature and loop flow. The necessary interlocking features are provided for protection of the system components and equipment. Recirculation flow through each loop and total recirculation flow are indicated in the Control Room, as well as loop temperature, valve position and pressure change across the pumps.

A modification installed during 1986 provides an alarm upon isolation of more than three recirculation loops. The alarm logic includes a reflash capability upon loss of all loops. This modification alerts the operator to a condition which potentially impacts the functioning of reactor level instrumentation.

5.4.1.3 System Evaluation

The Reactor Recirculation System is a Seismic Class I system. The operability requirements of, and limitations on, the system are included in the Technical Specifications. The system is part of the Reactor Coolant Pressure Boundary.

In addition to the system conditions which would cause the pumps to trip, the Recirculation Pumps trip on reactor vessel high pressure or low low reactor water level. A modification installed during 1996 would trip recirculation pumps A, B and E on reactor vessel high pressure and would trip recirculation pumps C and D on a persistent high pressure within 12 seconds (time delay setpoints will provide margin for calibration and accuracy of the time delay relays). This trip significantly limits the consequences of an anticipated transient without scram.

The transient responses of the plant to a trip of all the Recirculation Pumps and to the trip of one recirculation pump while operating at rated power are discussed in Chapter 15. Other transients, analyzed in Chapter 15 which relate to the Reactor Recirculation System, include:

a. Recirculation Pump Stall

b. Flow Controller Malfunction (Decreased Flow)

c. Maximum Flow Demand from Low Power (Increased Flow)

d. Startup of a Cold Recirculation Loop

The analysis of pipe breaks in the Reactor Recirculation System are presented in Subsection 3.6.2. The transient analysis for a recirculation pipe break is discussed in Chapter 15.

5.4.2 Steam Generators

Not applicable to BWR.

5.4.3 Reactor Coolant Piping

The OCNGS reactor coolant piping includes the Reactor Recirculation System piping (Subsection 5.4.1.2.5), the main steam line and feedwater piping (Subsection 5.4.9), core spray piping, closure head piping, reactor water cleanup system piping, isolation condenser system piping and the shutdown cooling system piping. OCNGS is utilizing Hydrogen Water Chemistry (HWC), pipe replacement, and stress improvement, on a partial basis for each method, to mitigate the potential for Intergranular Stress Corrosion Cracking (IGSCC). The HWC System at OCNGS mitigates the potential for IGSCC in the Reactor Coolant Piping by reducing the dissolved oxygen level in the reactor coolant with the addition of hydrogen gas into the feedwater. See Section 10.4.7.2.

The Isolation Condenser piping within the penetrations and outside the drywell were replaced with IGSCC-resistant material. Also, the piping within the reactor water cleanup system 2penetrations were replaced with the same IGSCC-resistant material.

Stress improvement has been used to minimize IGSCC in most of the original reactor coolant piping within the drywell. The schedule for implementation is provided in Reference 3, Section 3.9.7.

5.4.4 Main Steam Line Flow Restrictors

5.4.4.1 Design Bases

Each of the two 24 inch main steam headers (Subsection 5.4.9) is provided with a flow restrictor designed to serve as constrictions which, in conjunction with the Main Steam Isolation Valves (Subsection 5.4.5), will, in the event of a break in the main steam line downstream of the isolation valve:

a. Limit loss of coolant from the reactor vessel to the extent that level in the vessel will not fall below the top of the core within the time it takes to close the Main Steam Isolation Valves (MSIV).

b. Reduce the amount of moisture carryover before closure of the MSIVs.

c. Reduce the probability of forming water slugs of high velocity in the steam lines.

The design accident under which the flow restrictors are evaluated is the postulated complete severance of a main steam line outside the drywell.

5.4.4.2 Description

The flow restrictor is a simple venturi type tube welded into each main steam line between the reactor vessel and the first MSIV. The restrictor has no moving parts and is located as close to the vessel as practical. The ratio of the venturi throat area to steam line flow area is about 0.6, which results in an irreversible pressure drop of 5 to 10 psi. The design limits the steam flow in a severed line to about 191 percent of the full design rated steam flow, yet results in negligible increase in steam moisture content during normal operation. The restrictor is designed to withstand the maximum pressure difference expected following complete severance of a main steam line.

5.4.4.3 Design Analysis

In the event of a steam line break downstream of the restrictor, flow chokes in the decreased area by a two phase mechanism similar to the critical flow phenomena in gas dynamics. This limits the steam flow, thus reducing reactor coolant blowdown and limiting fuel clad temperature rise. The probability of fuel failure is thereby reduced.

Pressure surges caused by water-steam slugs impacting the flow limiter are within design limits; while beyond the restrictor velocities are reduced and pressure surges are of no consequence. The throat section of the flow limiting venturi is fitted inside a section of the 24 inch main steam line and held in place with a full circumferential 5/8 inch fillet weld. A calculated impact pressure of 1510 psia results in a 314,000 lb shear force on this weld and a resulting stress of 10,800 psi. Design code allowable stress for the material was 15,000 psi at 600 F and yield strength 60,000 psi. The analysis of the steam line rupture accident is presented in Chapter 15.

The flow restrictors have no moving parts and require no maintenance. Tests conducted to determine final design and performance characteristics of the restrictor have shown that:

a. Restrictor performance is in agreement with ASME correlations.

b. The loss of pressure is consistently about ten percent of total restrictor differential pressure.

c. Operation at critical throat velocities is stable and predictable.

5.4.5 Main Steam Line Isolation System

5.4.5.1 Design Bases

Main steam line isolation is accomplished by means of the Main Steam Isolation Valves (MSIVs). The MSIVs are containment isolation valves designed to minimize coolant loss from the vessel and thus offsite doses in the event of a main steam line break accident.

The valves are designed to close within three to ten seconds. The minimum closing time is chosen to minimize pressure buildup in the reactor vessel due to a quick cutoff of steam flow from the vessel. A three second valve closure limits vessel pressure to 1138 psia at the core midplane. The ten second maximum closing time is based upon the steam line break accident, the resulting loss of coolant and the resulting offsite dose rate (Chapter 15).

5.4.5.2 Description

Two isolation valves are installed in each of the two 24 inch main steam lines in parallel horizontal runs that penetrate the drywell through 36 inch diameter openings at El. 27'-0" and azimuths 171°15' and 180°45'. The penetrations have expansion joints to allow for steam line movement. One valve is located inside Containment, and the other outside Containment. Both sets of valves are located as close as possible to the drywell penetrations.

The basic design of the four valves is identical. Cross sectional views of the valve are shown in Drawing 20451-H. The valves are 24 inch angled globe valves of "Y" configuration. The cup shaped poppet moves on a centerline that is 45° upward from the horizontal centerline of the piping run. The valves in the inboard and outboard sets are rotated inward toward each other at 22°30' from vertical so that the air cylinders clear downcoming steam lines and other neighboring lines. Refer to Section 6.7 for discussion on leakage.

The diameter of the main valve seat is approximately the same as the inside diameter of the pipe and entrance and exit are streamlined to minimize pressure drop through the valve during normal steam flow. (Normal pressure drop is approximately 5.8 psi.)

Because of a history of LLRT failures of OCNGS MSIV's, the MSIV manufacturer, Atwood and Morrill, subsequently (Cycle 12R outage) modified the original MSIV design to provide assurance that the MSIV's would meet the LLRT requirements and to ensure reliable and acceptable valve seat tightness. The valve manufacturer's modification to the original valve components consists of the following:

- A new and improved main poppet with extended nose for proper seating.

- Internal poppet design with "self-aligning" feature to minimize seat leakage of the pilot poppet.
- New bottom spring plate design to allow separate disassembly of actuator assembly from cover poppet and stem assembly.
- A poppet backseat has been added to the valve cover to minimize flow induced poppet vibration.

The modified valve parts are designed to meet ASME Code allowable stresses requirements and the system design rating.

Each valve is controlled independently using a pneumatic system. The controls are capable of opening and slow speed exercising and fast closing the MSIVs one at a time.

Four differential pressure switches sensing high flow at the flow limiting venturi in each main steam line open on high differential pressure causing relays in the Reactor Protection System to be de-energized thus closing the MSIV. Critical flow corresponds to a venturi pressure loss of at least 400 psi. If the steam line break were near the turbine, the venturi pressure loss would be closer to 20 psi. Either case would cause a sufficient increase in venturi pressure loss over that at rated flow to initiate isolation valve closure within 5.0 seconds after the accident. The 5.0 second value includes the transmission time of a pressure wave propagating from the break location as well as any additional delay associated with instrument snubbers or signal processing. One-out-of-two-twice logic is used to prevent spurious closure of the isolation valves.

In the event of failure of the air supply, closing springs on the valve provide sufficient force to close the valve. The valve stroke time of 3-10 seconds is based on springs and accumulator air supply.

Two limit switches open when the valve has closed 10 percent (90 percent open position) to initiate reactor scram. These switches remain open during the rest of the closing stroke. Redundant full open and full closed limit switches operate lights in the Control Room to inform the plant operator regarding valve position. These switches can also be used to time valve closure.

During plant operation, the MSIVs can be tested and exercised individually to the 90% open position. The MSIVs can also be tested and exercised individually to the fully closed position if reactor power is reduced sufficiently to avoid scram from reactor overpressure or high flow through the steam line flow restrictor in the remaining steam line. Continued operation with one or both MSIVs closed on one Main Steam Line is permitted, provided that reactor thermal power is maintained at or below 50% of rated and reactor pressure is maintained at or below 1020 psig (rated operating pressure).

The Main Steam Isolation Valves no longer require a stem leakoff due to a change in their packing configuration. The seal leakoff connection on these valves has been capped.

The valves were designed to the requirements of ASA B16.5 as covered by Section VIII of the ASME B&PV Code. All wall thicknesses, flanges, cover bolting, and other pressure containing parts, were calculated to cover criteria of ASA B16.5. A basic design pressure of 1250 psig at 575°F was used. This is an interpolated ASA design of 655 psi, but all calculations were based on a minimum of 675 psi design rating. The valves were also designed for a hydrostatic shell test of 2450 psig.

Wall thicknesses were calculated for the design rating with a corrosion allowance of 0.088 for minimum wall thicknesses, and all castings were specified to be + 3/16", - 0 on the minimum wall thicknesses.

5.4.5.3 Design Evaluation

The conditions inside the reactor vessel, at the flow limiting venturis and at the Main Steam Isolation Valves were analyzed for the accident conditions involving a main steam line break outside the Reactor Building for various reactor conditions, namely: maximum power, startup and hot standby (Chapter 15). Steam line rupture at maximum power produces the worst conditions at the venturis and isolation valves, i.e., impact pressures due to level swell in the reactor vessel and mixture slugs (12 percent quality or greater) hitting the steam line restrictions at the venturi and the partially closed isolation valve. Calculations have shown that for 2 percent quality and 400 fps oncoming velocity at the venturi, total impact pressure is 1510 psia at the venturi and 380 psia at the valve.

Analyses of the isolation valve design reveal significant margin for the pressure containing parts. No damage to the main poppet or its guide surfaces is foreseen due to impact pressure loading. Closing force margin is sufficient to assure closure of the valve. Sufficient force is applied to the valve poppet to assure that it remains seated and seals following the accident.

Analysis of the dryer assembly and associated hardware reveals that all components can withstand the higher than normal loading due to flooding so that no parts are expected to break loose, enter the steam line and interfere with isolation valve closing.

The transient response of the plant to the closure of all MSIVs, from full power operation is discussed in Chapter 15.

The valve design has been analyzed for the effect of earthquake loading. These loads are small compared to the pressure and operating loads the valve components are designed to withstand. The cantilevered support of the air cylinder and the hydraulic cylinder was the key area investigated. The increase in loading at the joints between the support rods and the cover plate on the valve due to earthquake loading is negligible.

The valves are self supporting, with no external supports required under maximum stress conditions. Nevertheless, a spring loaded support is provided at the end of the actuator cover plate, and a vertical support is provided between the supporting floor and the valve body.

The main steam lines are anchored approximately 10 ft downstream from the outboard isolation valve. Tie rods extend from lugs on the main steam line just upstream from the outboard isolation valve to the building structure at the steam line anchor point. These tie rods are designed to restrain the steam line in the event of a break between the anchor and the outboard isolation valve. The outboard isolation valve is mounted on top of a constant support mechanism. The vertical run of steam line upstream from the flow limiting venturi is hung from two constant support hangers. Snubbers with stop features are provided for three axis (x, y and z) restraint at the elbow just upstream from the inboard isolation valve. Therefore, adequate support and snubbing has been provided for each of the main steam lines in the vicinity of the Main Steam Line Isolation Valves.

5.4.6 Reactor Core Isolation Cooling System

Not applicable to the OCNGS.

5.4.7 Residual Heat Removal System (Shutdown Cooling System)

5.4.7.1 Design Bases

The Shutdown Cooling System (SCS) is designed to remove fission product decay heat during shutdown, and to provide a means of cooling the reactor pressure vessel from approximately 350°F to 125°F to allow for refueling or maintenance following a period of operation.

The initial cooling and removal of decay heat, immediately following a shutdown of the turbine and reactor, is accomplished by means of the Turbine Bypass System. At approximately 350°F, steam pressure is too low to maintain the turbine shaft seals, so the Main Condenser is taken out of service to avoid pulling cold air in across the turbine shaft. The SCS is then placed in operation to complete the cooling and reduce reactor coolant temperature.

The refueling schedule requires that the vessel be cooled to a reasonable temperature (140°F) in about 20 hours for removal of insulation and head bolt nuts. Under normal refueling conditions, vessel head removal would begin 36 hours following reactor shutdown. The reactor coolant temperature should be decreased to 125°F by this time.

5.4.7.2 System Description

The SCS is comprised of three heat exchangers, three pumps and associated controls, instrumentation, motors and valves (Drawing GE148F711). The segment of the system from the reactor vessel to the isolation valve is designed for 1250 psig at 575°F, and the rest of the primary side of the system is designed for 1250 psig at 350°F. The pumps and heat exchangers are designed to the ASME B&PV Code, Section III, Class C. The heat exchangers are designed to ASME B&PV Code Section VIII on the shell (cooling water) side. The system up to the isolation valves is stainless steel, designed to the same specifications as the reactor vessel. The rest of the system is carbon steel. Hot reactor water at approximately 350°F is taken from the suction line of reactor Recirculation Pump E, upstream of the pump suction valve. A 14 inch line directs the flow to a motor operated isolation valve inside Containment and through the drywell wall.

The 14 inch line then branches into three 10 inch lines. Each of the three 10 inch shutdown lines has motor operated isolation valves at the pump suctions and at the heat exchanger outlet lines. Throttle valves are provided downstream of the heat exchangers.

Downstream of the discharge isolation valves, the 8 inch lines join into a 14 inch header which enters the Containment. One motor operated isolation valve is provided inside the drywell. The 14 inch line is connected to the discharge pipe of recirculation loop E, downstream of the Recirculation Pump discharge valve. The valves are interlocked to prevent opening if the reactor water temperature at the suction of any Recirculation Pump exceeds 350°F. Valve interlocks also prevent opening, and automatically close, the valves on either:

a. Low-low Reactor Water level (one-out-of-two-twice)

or

b. High Drywell pressure (one-out-of-two-twice)

The system is normally isolated. Flow elements are installed downstream of the heat exchangers to provide local indication for each SCS loop. The shutdown pumps have a design capacity of 3000 gpm at a total developed head of 225 feet and 350°F. The minimum Net Positive Suction Head (NPSH) required at the design point is 10.55 ft. The pumps are provided with interlocks which prevent operation unless the suction pressure exceeds 4 psig, the suction temperature is below 350°F, and the suction line isolation valve (V-17-19) is open. This assures that suction pressure is greater than the minimum NPSH and protects the system against inadvertent operation at excessive temperature. Each pump is protected by a 500 gpm maximum flow recirculation line connected from the outlet of the heat exchanger to the pump suction. The pump bearings and seals are cooled by the Reactor Building Closed Cooling Water (RBCCW) System.

The heat exchangers are of the horizontal U-Tube type with a heat removal capacity of 11 x 10^6 Btu/hr at a design flow of 3000 gpm, a reactor coolant temperature of 125°F, and a RBCCW flow of 1500 gpm at 75°F.

Relief valves are provided on the shell and tube sides of the heat exchangers. Temperatures at inlet and outlet locations are indicated and a high temperature condition is alarmed.

The system is designed to be drained to the RBEDT and inerted with nitrogen when not in use to reduce corrosion of metal surfaces; however normal procedure during system shutdown is to fill the system with condensate (Drawing 148F711).

Shutdown Cooling System safety-related motor-operated valves are included in the Generic Letter (GL) 89-10 Motor-Operated Valve (MOV) Program as noted in the OCNGS Program Description for NRC Generic Letter 89-10 Motor-Operated Valve Program. This program has reestablished the design basis for safety-related motor-operated valves. Critical design bases assumptions such as design bases differential pressure, safety function - open vs. close, minimum available AC/DC voltages, actuator gearing, torque switch control logic, valve factors, stem friction coefficients, and valve stroke times have been established in assessing GL 89-10 design bases capability. Plant changes or activities which can affect these design bases assumptions must consider the affect on the capability of GL 89-10 motor-operated valves to perform their safety function and on safety margins established for these valves.

5.4.7.3 <u>System Evaluation</u>

The SCS has components classified as Safety Related (Q), Augmented Quality (A), and Non-Safety Related (N). These are either seismic I or II. The SCS is classified as Seismic Category I inside the Drywell up to and including the normally closed valves outside of the Drywell (V-17-1, V-17-2, V-17-3 on the pump suction piping, and V-17-55, V-17-56, V-17-57 on the pump discharge piping). This includes the local drain and vent piping to the normally closed valve(s). The portion of the system outside of the Drywell beyond the listed normally closed drain/vent valves, including the pumps and heat exchangers and piping in between, is Seismic Category II. The SCS outside the containment beyond the normally closed valves will remain mechanically intact following a design basis seismic event. Therefore, while not Seismic Class I it is considered seismically capable. Access to components of the system is controlled because of high radiation conditions.

5.4.8 <u>Reactor Water Cleanup System</u> (Reactor Cleanup System)

5.4.8.1 Design Basis

The Reactor Cleanup System is a filtration and demineralization system for maintaining the purity of the water in the Reactor Coolant System. The system is operated to:

a. Reduce the deposition of water impurities on fuel surfaces, thus minimizing heat transfer surface fouling.

b. Reduce secondary sources of beta and gamma radiation by removing corrosion products, impurities and fission products from the reactor coolant.

c. Reduce the concentration of $C1^-$ ions to protect steel components from chloride stress corrosion.

d. Maintain or lower water level in the reactor vessel during startup, shutdown and refueling operations, in order to accommodate reactor coolant swell during heatup and to accommodate water inputs from the Control Rod Drive System and the Head Cooling System.

The system is designed to perform its function with minimum heat loss from the Reactor Coolant System, and to be operated during all phases of normal plant operation.

5.4.8.2 System Description

The system includes a regenerative heat exchanger, a non regenerative heat exchanger, a pressure reducing station, cleanup filters and auxiliaries, a cleanup demineralizer, cleanup pumps, a surge tank, a flow control station, a reactor drain station, isolation valves, piping, instrumentation and controls (Drawing GE148F444).

Under normal operation, reactor coolant flows under reactor pressure from the suction of reactor Recirculation Pump B, is cooled to 120°F in the regenerative and non regenerative heat exchangers (in series), its pressure reduced to 110 psig, filtered, demineralized, and pumped through a flow control valve and the regenerative heat exchanger to the discharge of reactor Recirculation Pump B. When reactor pressure is insufficient to maintain the required suction pressure at the cleanup recirculation pump, an auxiliary cleanup pump is placed in operation.

For draining the reactor, some of the cleanup system effluent flow is directed to the hotwell or to radwaste via a second flow control valve at the reactor drain station. The normal drain path is through the recirculation loop and cleanup system. There is also a drain line, from the bottom of the reactor vessel, which has normally open manual valves to the cleanup system.

The system is operated to maintain low levels of reactor water conductivity and undissolved solids. Conductivity is monitored at the influent (two cells) and at the effluent (one cell) of the demineralizer. Recording capability is provided, and abnormal conditions alarmed in the Control Room. The design flow rate is 380,000 lbs/hr.

The system supply line has a motor operated isolation valve inside the drywell and two parallel motor operated valves outside the drywell. The return line has one motor operated valve outside the drywell and one check valve inside the drywell. The isolation valves will close, and the cleanup pumps will stop automatically under any of the following conditions (refer to Table

6.2-12 for isolation signals for each valve):

a. Low flow, or outlet valve shut, for the cleanup filter in service.

b. High auxiliary pump cooling water outlet temperature.

c. High non regenerative heat exchanger outlet temperature (reactor coolant).

d. High pressure from the pressure reducing station.

e. Liquid poison system flow into the reactor vessel.

f. High drywell pressure.

g. Low-low reactor water level.

h. High area temperature (RWCU HELB isolation signal).

An exception to the above is valve V-16-61 which closes only on low-low reactor water level, high drywell pressure or High area temperature (RWCU HELB isolation signal).

A backup isolation on high pressure from the pressure reducing station isolates V-16-2 and V-16-14 only.

Reactor Water Cleanup System safety-related motor-operated valves are included in the Generic Letter (GL) 89-10 Motor-Operated Valve (MOV) Program as noted in the OCNGS Program Description for NRC Generic Letter 89-10 Motor-Operated Valve Program. This program has reestablished the design basis for safety-related motor-operated valves. Critical design bases assumptions such as design bases differential pressure, safety function - open vs. close, minimum available AC/DC voltages, actuator gearing, torque switch control logic, valve factors, stem friction coefficients, and valve stroke times have been established in assessing GL 89-10 design bases capability. Plant changes or activities which can affect these design bases assumptions must consider the affect on the capability of GL 89-10 motor-operated valves to perform their safety function and on safety margins established for these valves.

The major characteristics of the Reactor Cleanup System components are presented in Table 5.4-1. The equipment is designed to ASME B&PV Code, Section III, Class C on the primary side. The tube side of the heat exchanger is ASME VIII.

The pressure reducing station consists of a pressure control valve and a bypass control valve, and relief valves. Pressure is maintained at or below 110 psig in the filter and demineralizer portion of the cleanup system. High pressure from the pressure reducing station trips the cleanup system isolation valves and pumps.

The pressure control valve is a 4 inch, globe, single seat valve, air operated, air to open, fail close, with pressure controller. A pressure relief valve located just downstream of the pressure control valves protect the low pressure portions of the cleanup system. One 6 inch valve can discharge up to 125 pounds per second through a 20 inch line and one isolation check valve to the torus. A remote operated solenoid leakoff valve is used to detect relief valve leakage. One 1 inch valve is provided in line to the Reactor Building Equipment Drain Tank. The filters, the demineralizers, and other isolable portions of the system have relief valves which discharge to the Reactor Building Equipment Drain Tank.

Cleanup system flow is normally maintained at approximately 400 gpm. The system flow is measured by a flow element at the cleanup demineralizer inlet. Low flow at this element automatically starts the filter precoat pump to recycle, in order to hold the filter cake.

The flow control valve is a 4 inch globe, air diaphragm operated valve, air to open, spring and flow to close. This valve is located between the cleanup pumps and the regenerative heat exchanger. An electropneumatic converter supplies control air to the valve diaphragm, and is controlled by the flow controller, with feedback from the flow element. Cleanup pump suction header low pressure results in a flow reduction demand to the flow controller. A manually operated bypass valve is installed around the valve as backup to the automatic flow control. A flow recorder along with digital display is provided to monitor flow and system performance. The system is normally operated in the manual mode.

The reactor drain station consists of an auxiliary pressure-reducing valve and orifice in series, a motor operated orifice bypass valve, a flow element, and motor-operated shutoff valves to the hot well and to radwaste. The takeoff line for this station is located upstream of the cleanup recirculation pumps.

A normally locked open 6 inch manual valve (V-16-63) is located inside the drywell downstream of the return line check valve and provides a means of isolating the return line from the RPV. In order to implement repairs to V-16-63 with the plant mode switch in either the SHUTDOWN or REFUELING position, the use of a freeze seal positioned between the Reactor Vessel and V-16-63 may be necessary, subject to the following restrictions.

a. RPV is reassembled and fuel is in the vessel.

b. The reactor coolant system bulk temperature, as measured by TE-31D, shall not exceed a temperature of 99ºF prior to the initiation of the valve repair activity and no evolutions involving removal of the valve disc shall occur if the reactor coolant system bulk temperature exceeds 99ºF.

c. RPV water level shall not exceed 210 inches above TAF.

d. Decay heat from the RPV will be removed by the Shutdown Cooling System via the RBCCW/Service Water Systems.

e. Both Core Spray Systems shall be operable in accordance with the Technical Specifications.

f. Fire Protection System shall be operable and capable of delivering water to the Core Spray System.

g. At least one loop of the Reactor Recirculation System must remain open with the suction and discharge valves in the full open position.

h. The Emergency Diesel Generators and their respective NSR electrical distribution systems are returned to service and available to support emergency operation.

i. Offsite power shall be available.

j. Secondary Containment shall be operable in accordance with the Technical Specifications.

k. Both Standby Gas Treatment Systems will be operable.

l. The RPV will be vented to the atmosphere and the established vent path will be tagged in the open position.

m. RWCU System pumps P-16-001A and P-16-001B will be electrically deenergized.

n. RWCU System valves V-16-1, V-16-2, V-16-14 and V-16-61 will be tagged in the closed position.

o. Provisions will be in place to remove any equipment traversing the Drywell Airlock to ensure that the airlock can be closed in a timely fashion should the need arise.

p. The seal plate assembly for V-16-63 shall be staged and ready for fit-up and installation.

5.4.8.3 System Evaluation

The system is normally operated continuously during all phases of reactor operation. Cleanup system operation is necessary to maintain reactor coolant purity, and reactor operation without the cleanup system is limited to relatively short periods of time.

The system is automatically isolated from the Reactor Coolant System under abnormal operating conditions. Relief valves and instrumentation are provided to protect the system against overpressurization.

5.4.9. Main Steam and Feedwater Piping

5.4.9.1 Main Steam Piping

The dry steam in the reactor vessel head cavity is removed through two 24 inch nozzles on the sides of the reactor vessel. The nozzles are located due north and due south on the vessel, approximately 5'-8" below the vessel head flange. The headers curve downward to a horizontal header containing the safety and relief valves. Each header leaves the drywell through a 36 inch guard pipe with a flexible pressure seal to the header on the outside of the drywell. Each header enters the tunnel to the Turbine Building through a second pressure seal designed to prevent steam leaks from the valve space to the tunnel.

There are 14 safety/relief valves on the main steam piping. Nine safety valves discharge directly into the drywell (Subsection 5.4.10). The remaining five are the Electromatic Relief Valves (EMRVs), which are described as part of the Automatic Depressurization System in Section 6.3.

The main steam lines are provided with flow restrictors (Subsection 5.4.4) and with Main Steam Isolation Valves inside and outside Containment (Subsection 5.4.5). The main steam lines inside the drywell are shown in Drawing BR 2002. Piping was designed to ANSI B31.1.

Figure 5.4-7 shows the general arrangement of the main steam safety and relief valves on the main steam piping. It is anticipated that the safety valves will never be actuated since overpressure is relieved by the Isolation Condensers, the turbine bypass valves, and the Electromatic Relief Valves. The effects of "blowoff" of these valves have been studied and taken into account in the following way. First, the five relief valves are set to open at a pressure lower than those specified for the safety valves. The discharge from the relief valves is piped to the bottom of the torus - completely away from drywell equipment. Second, the discharge from each safety valve initially impinges on a tee which deflects the steam 90 degrees in two opposite directions. Each tee discharge is aimed specifically to prevent its direct discharge on equipment which could be affected from such forces.

In an overpressure situation, the relief valves would actuate before the safety valves, with subsequent discharge to the torus, followed by safety valve blowoff (if the overpressure situation persisted) which by virtue of the selected tee aiming will not adversely affect equipment in the drywell.

The transient response of the plant to inadvertent opening of a safety valve is discussed in Chapter 15.

An acoustic based Valve Monitoring System on the safety and relief valves provides valve position indication in the Control Room. The valve position indication is seismically qualified and capable of operation in its appropriate environment. Thermocouples provide backup monitoring capability.

5.4.9.2 Feedwater Piping

Downstream of the high pressure feedwater heaters, the 14 inch feedwater lines discharge into a common 24 inch header, which branches into two 18 inch lines to the Reactor Building. These run parallel to the main steam lines and have similar penetrations through the Containment. There are dual check valves in each line, one inside and one outside the drywell. Locked open manual isolation valves for each branch are located within the drywell downstream of the second check valve. There is a one inch drain to the RBEDT downstream of the first check valve in each branch. Tandem manual valves in each drain line are provided outside the drywell penetration.

The 18 inch branches divide into two 10 inch lines each of which penetrate the vessel at the feedwater nozzles and supply water to the sparger.

The feedwater piping was designed to ASA B31.1 except for the piping indicated on Table 5.1-1.

5.4.10 Safety Valves

The safety valves serve as a complementary means of pressure relief for the reactor vessel. This system has no function during normal operation. The pressure relief system was designed in accordance with the ASME B&PV Code, Section I (1962 edition). Under the provisions of Section I, the safety valves must limit the rise in the reactor vessel pressure to less than the

ASME Code limit. Nine safety valves were demonstrated sufficient to provide the reactor vessel overpressure protection (see Section 5.2.2.4.2.1).

The valves are spring released and flange mounted on the main steam lines to permit periodic removal for testing. Five valves are mounted on the north main steam header and four on the south main header. During 13R, seven safety valves were removed and replaced with blind flanges. The setpoints for the nine safety valves are delineated on Figure 5.4-7 and are listed in the Technical Specifications.

5.4.11 Head Cooling System

5.4.11.1 Design Basis

The Head Cooling System is used in conjunction with reactor vessel flooding and the Shutdown Cooling System, for condensing steam formed in the vessel head and for cooling the flanges and the upper portions of the reactor pressure vessel during shutdown operation.

The system is designed to meet the following objectives (with associated vessel flooding and Shutdown Cooling System operation):

a. Condense steam and condensible gases in the vessel dome to assist in vessel head cooling during shutdown.

b. Prevent repressurization as the vessel is flooded to levels above the vessel flange and main steam nozzles to cool the upper portions of the vessel metal.

c. Provide vessel head cooling under the direct control of the Control Room operator during shutdown, after local valve settings have been completed.

d. Permit reactor pressure to be reduced to atmospheric, and vessel head temperature to be reduced to approximately 140°F without causing metal temperature differentials which would affect the integrity of the reactor vessel during its designed lifetime.

5.4.11.2 Description

The Head Cooling System consists of a single fog spray nozzle located inside the top of the reactor pressure vessel head, which sprays a maximum of 170 gpm through a cone angle of 70 degrees. (The spray does not strike the head metal surface.) The head spray water is supplied from the Condensate Storage Tank by the standby Control Rod Drive (CRD) Hydraulic System pump. Head spray flow is measured by a flow element, indicated in the Control Room, and controlled by a pneumatically operated flow control valve.

The Head Cooling System is connected to the vessel head nozzle by a removable 2 inch stainless steel pipe spool piece. A check valve is installed as the isolation valve inside the drywell. The isolation valve outside the drywell is an air operated globe valve which is remotely controlled from the Control Room and which will close automatically from an isolation signal. There are manually operated stop valves in the head cooling system connections to the CRD Hydraulic System pumps. A leak-off line is used for system drainage to the Reactor Building equipment drain sump, and to observe for leakage through the isolation of stop valves.

Reactor vessel head temperatures are measured at the outer diameter of the head flange and at the outside surface of the hemispherical head above the transition weld.

A cross-connect line between the head spray line and the head vent line prevents accumulation of hydrogen and other non-condensable gases in the head cooling line above the reactor vessel during normal power operation. During reactor head cooling system operation, a small amount of water is diverted to the reactor head vent nozzle. A restriction orifice in the cross-connect line, limits flow to prevent thermal shock.

5.4.11.3 System Evaluation

The system requirements are satisfied as follows:

a. During reactor operation the manual stop valves connecting the Head Cooling System to the CRD hydraulic pumps are both closed; and the pneumatically operated valves in the Head Cooling System are both closed. Thus, water will not be injected through the head spray due to leaking valves.

b. The isolation check valve in the Head Cooling System line and the closed valves prevent reactor steam from entering the CRD Hydraulic System.

c. For normal operation of the CRD Hydraulic System, check valves in the discharge lines permit the operator to change the pumps from the Control Room without making any changes in the local valve positions.

d. During shutdown, when the spare CRD hydraulic pump is required for the Head Cooling System, local stop valves are repositioned.

e. If the CRD hydraulic pump supplying the rod drive system fails, the local valving is changed manually to give priority to the rod drives, thereby removing the spray system from service. Thus the Head Cooling System adds the requirements of manual valve control in the event of a CRD hydraulic pump failure during shutdown spraying.

f. The CRD pumps are interlocked on low pump suction and protected from low flow conditions by recirculation.

g. If the manual stop valve is accidentally left open when the spare CRD hydraulic pump is changed from head spray service to CRD system service, the flow would divide with approximately 25 percent to the CRD system and 75 percent to the Head Cooling System.

h. The head spray connection to the vessel is made through a removable spool piece to prevent interference with head removal. The cross-connect pipe joins the head spray pipe upstream of the removable spool piece and above the mirror insulation to the head vent pipe and, thus, will not interfere with head removal.

i. A restriction orifice provided in the cross-connect line will minimize cooling water diversion and will not affect head cooling system operation.

TABLE 5.4-1
(Sheet 1 of 3)

DESIGN CHARACTERISTICS - CLEANUP SYSTEM COMPONENTS

Auxiliary Cleanup Pump	
Type	Horizontal Centrifugal pump
Flow	840 gpm
Design Conditions	1250 psig, 575°F
Developed Head	300 ft at 120°F
Motor	100 hp, 440 volt, 3 phase, 60 cycle
Regenerative Heat Exchanger (3)	
Type	U-tube
Materials	Type 304 stainless steel
Design Pressure	1300 psig
Design Temperature	575°F
Nonregenerative Heat Exchangers (2)	
Type	U-tube
Materials	
Tubes	Type 304 stainless steel
Shells	Carbon steel
Tube Sheets	Type 304 solid stainless steel
Design Pressure	1300 psig, tubes; 150 psig, shell
Design Temperature	575°F, tubes; 300°F, shell

TABLE 5.4-1
(Sheet 2 of 3)

DESIGN CHARACTERISTICS - CLEANUP SYSTEM COMPONENTS

Cleanup Filters	
Type	Two full-flow, pressure-precoat
Precoat Material	Fiber and/or resin
Flow	760 gpm each filter
Design Conditions	150 psig, 200°F
Cleanup Demineralizer	
Type	Mixed bed
Flow Rate	760 gpm
Operating Temperature	120°F normal 140°F maximum
Design Conditions	150 psig, 150°F
Material	Type 304 stainless steel

TABLE 5.4-1
(Sheet 3 of 3)

DESIGN CHARACTERISTICS - CLEANUP SYSTEM COMPONENTS

Cleanup Pumps (2)

Type	Centrifugal
Flow	420 gpm, each pump -- two running in parallel
Developed Head	2650 ft at 120°F
Design Pressure	Discharge 1670 psig, suction 150 psig
Design Temperature	140°F
Motor	400 hp, 4000 volt, 3 phase, 60 cycle

TABLE 5.4-2
(Sheet 1 of 1)

HEAD COOLING SYSTEM DESIGN PARAMETERS

System Design Flow	140 gpm
Design Conditions	
Reactor Vessel to A.O. Isolation Valve	1250 psig, 575°F
A.O. Isolation Valve	1750 psig, 350°F
CRD Hydraulic Pump to A.O. Isolation Valve	1750 psig, 150°F
Spray Nozzle Location	Reactor-vessel head nozzle N7B
Spray Nozzle Pressure Drop	80-100 psi at 170 gpm
Flow Control Valve Characteristic	140 gpm with 1240 psi differential pressure 25 gpm minimum flow with 1650 psi