SECTION 4.0

REACTOR COOLANT SYSTEM

4.1 SUMMARY DESCRIPTION

This section describes the systems and components that form the major portions of the nuclear system process barrier. These systems and components contain or transport the fluids coming from or going to the reactor core.

Subsection 4.2, "Reactor Vessel and Appurtenances Mechanical Design," describes the reactor vessel and the various fittings with which other systems are connected to the vessel. The major safety consideration for the reactor vessel is the ability of the vessel to function as a radioactive material barrier. Various combinations of loading are considered in the vessel design. The vessel meets the requirements of the various applicable codes and criteria. The possibility of brittle fracture is considered, and suitable limits are established that avoid conditions where brittle fracture is possible.

The reactor recirculation system varies coolant flow through the core, as described in subsection 4.3. Adjustment of the core coolant flow rate changes reactor power output, thus providing a means of following plant load demand without adjusting control rods. The recirculation system is designed to provide a slow coastdown of flow on loss of power so that fuel thermal limits are not exceeded. The arrangement of the recirculation system is such that a piping failure cannot compromise the integrity of the floodable inner volume of the reactor vessel.

The nuclear system pressure relief system, described in subsection 4.4, protects the nuclear system process barrier from damage due to overpressure. To accomplish overpressure protection, a number of pressure-operated relief valves are provided that can discharge steam from the nuclear system to the primary containment. A portion of the nuclear system pressure relief system also acts to automatically depressurize the nuclear system in the event of a LOCA in which the HPCIS fails to maintain reactor water level. The depressurization of the nuclear system allows low-pressure CSCS's to supply cooling water to adequately cool the fuel.

The main steam line flow restrictors, described in subsection 4.5, limit the loss of coolant resulting from a main steam line break outside the primary containment. The coolant loss is limited so that reactor vessel water level remains above the top of the core during the time required for the main steam line isolation valves (MSIV's) to close. This action protects the fuel barrier. The restrictors are venturi-type flow devices, and one restrictor is installed in each main steam line close to the reactor vessel but downstream from the pressure relief valves.

The MSIV's, described in subsection 4.6, automatically close off the nuclear system process barrier in the event a pipe break occurs downstream of the valves. This action limits the loss of coolant and the release of radioactive materials from the nuclear system. Two isolation valves are installed on each main steam line, one located inside, and the other outside, the primary containment. In the event a main steam line break occurs inside the primary containment, closure of the isolation valve outside the containment acts to seal the primary containment itself.

The RCICS, described in subsection 4.7, provides makeup water to the core during a reactor shutdown in which feedwater flow is not available. The system can be started manually by the operator, or automatically upon receipt of a low reactor water level signal. Water is pumped to the core by the turbine-pump driven by reactor steam.

The RHRS, described in subsection 4.8, includes pumps and heat exchangers that can be used to cool the nuclear system under a variety of situations. The shutdown cooling mode of the RHRS is used during normal shutdown and reactor servicing to remove residual heat. The LPCI mode of the RHRS is an engineered safeguard for use during a LOCA. This operation is described in Section 6.0, "Core Standby Cooling Systems." The containment spray mode of the RHRS allows the removal of heat from the primary containment following a LOCA.

The reactor water cleanup system, described in subsection 4.9, functions to maintain the required purity of reactor coolant by circulating coolant through a system of filter-demineralizers.

Subsection 4.10, "Nuclear System Leakage Detection and Leakage Rate Limits," establishes the limits on nuclear system leakage so that appropriate action can be taken before the nuclear system process barrier is impaired.

Subsection 4.11, "Main Steam Lines, Feedwater Piping, and Drains," describes the design requirements and arrangement of these piping systems and their associated drains.

4.2 REACTOR VESSEL AND APPURTENANCES MECHANICAL DESIGN

4.2.1 Power Generation Objective

The reactor vessel's power generation objective is to provide a volume in which the core can be submerged in coolant, thereby allowing power operation of the fuel. The design of the reactor vessel and appurtenances provides for the attachment of lines to the reactor vessel, and the means for the proper installation of vessel internal components.

4.2.2 Power Generation Design Basis

1. The location and design of the external and internal supports provided as an integral part of the reactor vessel are such that stresses in the reactor vessel and supports, due to reactions at these supports, are within ASME code limits.

2. The reactor vessel design life is 40 years.

3. The design of the reactor vessel and appurtenances allows for the accomplishment of a suitable program of in-service inspection.

4.2.3 Safety Design Basis

1. The reactor vessel and appurtenances are designed to withstand adverse combinations of loadings and forces resulting from operation under abnormal and accident conditions.

2. To minimize the possibility of brittle fracture failure of the nuclear system process barrier, the following are required: (a) the initial ductile-brittle transition temperature of materials used in the reactor vessel are known by reference or established empirically; (b) expected shifts in transition temperature during design service life due to environmental conditions, such as neutron fluence and elevated temperature, were determined and employed in the reactor vessel design; (c) operation margins to be observed with regard to the transition temperature are designated for each mode of operation.

4.2.4 Description

4.2.4.1 Reactor Vessel

The reactor vessel is a vertical, cylindrical pressure vessel with hemispherical heads and is of welded construction. The reactor vessel is designed and fabricated for a useful life of 40 yr based upon the specified design and operating conditions. The vessel is designed in accordance with the ASME Boiler and Pressure Vessel Code, Section III 1965 Edition, its interpretations, and applicable requirements (including 1965 Winter Addendum for Units 2 & 3 and additional code applications as stated on the N-1A Vessel Data Reports) for Class A vessels as defined therein. The design of the reactor vessel and its support system meets seismic Class I requirements.

The materials used in the design and fabrication of the reactor pressure vessel are shown in Table 4.2.1. The reactor vessel data is shown in Table 4.2.2.

The cylindrical shell and bottom hemispherical head of the reactor vessel are fabricated of low alloy steel plate. The shell is clad on the interior with a stainless steel overlay, and the bottom head with an Inconel overlay. The plates and forgings are ultrasonically and magnetic particle tested on 100 percent of their surfaces after forming and heat treatment. Preheat of vessel plate and forgings is maintained during welding until the weld joints are post-weld-heat treated. Full penetration welds are used at all joints, including nozzles, throughout the vessel except for nozzles of less than 3-in nominal size and the CRD housing-to-stub tube welds.

Although little corrosion of plain carbon or low alloy steels occurs at temperatures of 500°F to 600°F, higher corrosion rates occur at lower temperatures around 140°F. The 0.125-in minimum thickness stainless steel cladding provides the necessary corrosion resistance during reactor shutdown, and also helps maintain water clarity during refueling operations. Since the vessel head is exposed to a saturated steam environment throughout its operating lifetime, stainless steel cladding is not required over its interior surfaces. Exterior exposed ferritic surfaces of pressure-containing parts have a minimum corrosion allowance of 1/32-in. The interior surfaces of the top head and all carbon and low alloy steel nozzles exposed to the reactor coolant have a corrosion allowance of 1/16-in. The vessel shape is designed to limit coolant retention pockets and crevices.

The nil-ductility transition (NDT) temperature is defined as the temperature below which ferritic steel fails in a brittle rather than ductile manner. The NDT temperature increases, as a function of neutron exposure, at integrated neutron exposures greater than about 1×10^{17} n/cm^2, with neutrons of energies in excess of 1 Mev. Since the material NDT temperature dictates the minimum operating temperature at which the reactor vessel can be pressurized, it is desirable to keep the NDT temperature as low as possible. The as-fabricated initial NDT temperature for all carbon and low alloy steel used in the main closure flanges, and the shell and head materials connecting to these flanges, are limited to a maximum of 10°F as determined by ASTM E208. For all other carbon and low alloy steel pressure-containing materials and the vessel support skirt material, the as-fabricated initial NDT temperature is no higher than 40°F. A grain size of 5 or finer, as determined by the method in ASTM E112, is maintained.

NDT temperature may also be minimized by reducing the integrated neutron exposure at the vessel wall where a quarter thickness (1/4 T) flaw is assumed for ASME Code analysis. The updated predicted peak fluence at the 1/4 T of the RPV beltline is 1.14×10^{18} n/cm^2 for Unit 2, and 1.09×10^{18} n/cm^2 for Unit 3 after 60 years of service (54 EFPY), based on the analysis results documented in NEDC-33873P, Safety Analysis Report for Peach Bottom Atomic Power Station Units 2 and 3, Thermal Power Optimization (Reference 4). The revised analysis performed as part of the GNF3 fuel introduction does not result in a change to the peak fluence values at the 1/4 T of the RPV beltline but does reduce the P-T Curve EFPY to 49.7 for Unit 2 and 47.5 for Unit 3 (Reference 6). Pressure and temperature limits for various operating states are contained in the Technical Requirements Manual (TRM). The RPV fracture toughness evaluation is contained in Reference 5.

The vessel top head is secured to the reactor vessel by studs, nuts, and bushings which are designed to be tightened with a stud tensioner. The vessel flanges are sealed by two concentric, stainless steel seal-rings designed for no detectable leakage through the inner or outer seal at any operating condition, including hydrostatic test pressures and heating to operating pressure and temperature at maximum rate of 100°F/hr. To detect a lack of seal integrity, a 1-in. vent tap is provided in the area between the two seal rings, and a monitor line is attached to the tap to provide an indication of leakage from the inner seal-ring seal. A 1-in. tap is also provided in the area outside the outer seal-ring.

The head and vessel flanges are low alloy steel forgings. The sealing surfaces of the reactor vessel head and shell flanges are weld overlay clad with nickel-chromium-iron alloy which consists of a minimum of two layers and a minimum of 0.25 in. total thickness after all machining, including the area under the seal grooves. The nickel-chromium-iron alloy weld metal is in accordance with ASME SFA 5.11, type ENiCrFe-3, UNS No. W86182.

The vessel nozzles and penetrations (Figure 4.2.2) are low alloy steel forgings made in accordance with ASTM A508 as modified by ASME Code Case 1332-2, Paragraph 5. Nozzles of 3-in. nominal size or larger are full penetration welded to the vessel. Nozzles of less than 3-in. nominal size may be partial penetration welded as permitted by ASME Boiler and Pressure Vessel Code, Section III. Nozzles which are partial penetration welded are nickel-chromium-iron forgings made in accordance with ASME SB-166 as modified by Code Case 1336 or in accordance with ASME SB-167. The CRD stub tubes which form the CRD penetrations are welded using either a full or partial penetration weld in accordance with Section III of the ASME Boiler and Pressure Vessel Code (Winter Addenda 1965).

The vessel top head nozzles are provided with flanges with small groove facing. The drain nozzle is of a full penetration weld design and extends 16 in. below the bottom outside surface of the vessel. The recirculation inlet nozzles (located as shown in Figure 4.2.1), feedwater inlet nozzles, and core spray inlet nozzles have thermal sleeves similar to those shown in the detail of Figure 4.2.2.

Nozzles connecting to stainless steel piping are clad to a minimum thickness of 0.125-in. with stainless steel weld overlay equivalent to that used in the vessel, and provided with a solution, heat-treated, stainless steel transition piece. Nozzles for connecting carbon steel piping (except the top head nozzles and the feedwater nozzles which are unclad) are clad through at least the thickness of the vessel wall or one-half the diameter of the nozzle bore, whichever is less.

The nozzle for the core differential pressure and liquid control pipe is designed with a transition so that the stainless steel outer pipe of the differential pressure and liquid control line (subsection 3.3, "Reactor Vessel Internals Mechanical Design") can be socket welded to the inner end of the nozzle, and so that the inner pipe passes through the nozzle. This design provides an annular region between the nozzle and the inner liquid control line to minimize thermal shock effects on the reactor vessel in the event that use of the standby liquid control system is required.

The jet pump instrumentation penetration seal, the stainless steel recirculation loop piping, and the main steam line piping are welded directly to their respective nozzles.

Thermocouple pads are located on the exterior of the vessel (Table 4.2.3). At each thermocouple location, two 3/4-in diameter pads are provided: an end pad to hold the end of a 3/16-in diameter thermocouple, and a clamp pad equipped with a set screw to secure the thermocouple.

Security-Related Information Withheld under 10 CFR 2.390

NOTES:
1. Includes volume of jet pumps.
2. Inside shroud vessel only.
3. Includes volume outside of shroud vessel.
4. Includes steam separator returning water volume.
5. Does not include steam volume leaving steam separators.
6. Includes steam volume leaving steam separators.

4.2.4.2 Shroud Support

The reactor vessel shroud support is a radial cylindrical shell that surrounds the reactor core assembly and is designed so that stresses due to reactions at the shroud support for normal, upset, emergency, and faulted loading conditions are within ASME Boiler and Pressure Vessel Code, Section III requirements. The design pressure differential across the core shroud support (higher pressure under the support) occurs at the vessel design temperature. The design of the shroud support also takes into account the restraining effect of the components attached to the support, weight, and seismic loadings. The vessel shroud support and other internal attachments are as shown in Figure 4.2.1 and Table 4.2.3.

4.2.4.3 Reactor Vessel Support Assembly

The reactor vessel is laterally and vertically supported and braced to make it as rigid as possible without impairing the movements required for thermal expansion. Where thermal requirements prohibit the use of rigid supports, spring snubbers are employed to resist seismic forces while allowing sufficient flexibility for thermal expansion.

The reactor vessel support assembly consists of a ring girder and the various bolts, shims, and set screws necessary to position and secure the assembly between the reactor vessel support skirt and the support pedestal. The concrete and steel support pedestal is constructed integrally with the building foundation. Steel anchor bolts are set in the concrete with the threads extending above the surface. The anchor bolts extend through the ring girder bottom flange. High strength bolts are used to bolt the flange of the reactor vessel support skirt to the top flange of the ring girder. The ring girder is fabricated of ASTM A36 structural steel according to AISC specifications.

4.2.4.4 Vessel Stabilizers

Disc spring-type vessel stabilizers connect the reactor vessel to the top of the shield wall surrounding the vessel to provide lateral stability. Eight stabilizer brackets are attached, by full penetration welds, to the reactor vessel at evenly spaced locations around the vessel below the flange. Two stabilizers are attached to each bracket and are pretensioned. The stabilizers are designed to permit radial and axial vessel expansion, to limit horizontal vibration, and to resist seismic and jet reaction forces.

4.2.4.5 Refueling Bellows

The refueling bellows forms a seal between the reactor vessel and the drywell to permit flooding of the space (reactor well) above the vessel during refueling operations. The refueling bellows assembly (Figure 4.2.1) consists of a type 304 stainless steel bellows, a backing plate, a spring seal, and a removable guard ring. The backing plate surrounds the outer circumference of the bellows to protect it, and is equipped with a tap for testing and for monitoring leakage. The self-energizing spring seal is located in the area between the bellows and the backing plate, and is designed to limit water loss in the event of a bellows rupture by yielding to make a tight fit to the backing plate when subjected to full hydrostatic pressure. The guard ring attaches to the assembly and protects the inner circumference of the bellows. The guard ring can be removed from above to inspect the bellows. The assembly is welded to the reactor bellows support skirt and the reactor well seal bulkhead plate. The reactor bellows support skirt is welded to the reactor vessel shell flange (Figure 4.2.1), and the reactor well seal bulkhead plate bridges the distance to the drywell wall. The bulkhead plate is penetrated by six ventilation ducts. These penetrations are sealed by six watertight, hinged covers during refueling. For normal operation, these covers are opened and removable air supply ducts and air return ducts permit circulation of ventilation air in the region above the reactor well seal.

4.2.4.6 Control Rod Drive Housings

The CRD housings are inserted through the CRD penetrations in the reactor vessel bottom head and are welded to the stub tubes extending into the reactor vessel[1] (Figure 4.2.2). Each housing transmits a number of loads to the bottom head of the reactor. These loads include the weight of a control rod and control rod drive, which are bolted to the housing from below (subsection 3.4, "Reactivity Control Mechanical Design"), the weight of a control rod guide tube, one four-lobed fuel support piece, and the four fuel assemblies which rest on the top of the fuel support piece (subsection 3.3, "Reactor Vessel Internals Mechanical Design"). The housings are fabricated of type 304 austenitic stainless steel.

4.2.4.7 Control Rod Drive Housing Supports

The CRD housing support is designed to prevent a nuclear transient in the unlikely event that there is a CRD housing failure. This device consists of a grid structure located below the reactor vessel, from which housing supports are suspended. The supports allow only slight movement of the CRD or housing in the event of failure. The CRD housing support is treated in detail in subsection 3.5, "Control Rod Drive Housing Supports."

4.2.4.8 In-Core Neutron Flux Monitor Housings

The in-core neutron flux monitor housings are inserted up through the in-core penetrations in the bottom head of the reactor vessel and are welded to the inner surface of the bottom head (Figure 4.2.2). An in-core flux monitor guide tube is welded to the top of each housing (subsection 3.3, "Reactor Vessel Internals Mechanical Design") and either an WRNM or an LPRM is bolted to the seal-ring flange at the bottom of the housing (subsection 7.5, "Neutron Monitoring System").

4.2.4.9 Reactor Vessel Insulation

The reactor vessel insulation is the all-metal reflective type and has an average maximum heat transfer rate of approximately 80 Btu/hr-sq ft at the average operating conditions of 550°F for the vessel and 135°F for the outside air. The nominal insulation thicknesses are 4 in. for the upper head, 3 1/2 in. for the cylindrical shell and nozzles, and 3 in. for the bottom head. The upper head insulation is designed to permit complete submersion in water during shutdown without loss of insulating material, or adverse effect on the insulation efficiency of the insulation assembly after draining and drying. The lower head and cylindrical shell insulation is designed for the 40-year life of the vessel. The insulation panels for the cylindrical shell of the vessel are held in place by vessel insulation supports located at two elevations on the vessel. The support brackets for each support are full penetration welded to the vessel at 12 evenly spaced locations around the circumference. The upper lead insulation frame is seismically designed to support the piping passing through the insulation.

4.2.5 Safety Evaluation

The reactor vessel design pressure of 1,250 psig is determined by an analysis of margins required to provide a reasonable operating range, with additional allowances to accommodate transients above the operating pressure (1,035 psig at the level of the top head flange), without causing operation of the safety valves. The design temperature for the reactor vessel (575°F) is based on the saturation temperature of water corresponding to the design pressure.

To withstand external and internal loadings while maintaining a high degree of corrosion resistance, a high strength carbon alloy steel is used as the base metal with an internal cladding of stainless steel applied by weld overlay. Use of the ASME Boiler and Pressure Vessel Code, Section III, Class A, pressure vessel design criteria provides assurance that a vessel designed, built, and operated within its design limits has an extremely low probability of failure due to any known failure mechanism.

Table 4.2.4 lists the fatigue monitoring locations throughout a 60-year life for which the reactor vessel fatigue has been evaluated.

Stress analysis and load combinations for the reactor vessel are evaluated for the monitoring locations listed in Table 4.2.4, with the conclusion that ASME Boiler and Pressure Vessel Code limits are satisfied. The details of assumed loading combinations are described in Appendix C, Section C.5.3.

The reactor vessel was originally designed for a 40-year life and to not be exposed to more than $1x10^{19}$ n/cm^2 of neutrons with energies exceeding 1 MeV. Extensive tests have established the magnitude of changes in the NDT temperature as a function of the integrated neutron dosage. Regulatory Guide 1.99, Revision 2, gives correlations of NDT change versus fluence based on statistical evaluation of the commercial reactor surveillance database. It represents the best method currently available to predict irradiation changes in fracture toughness. The regulatory guide requires that two surveillance capsules be tested before factoring the surveillance Charpy data into plant-specific shift predictions. Therefore, the generic approach presented in the regulatory guide is used for Peach Bottom 2 and 3.

The reactor assembly is designed such that the average annular distance from the outermost fuel assemblies to the inner surface of the reactor vessel is approximately 80 cm. This annular volume, which contains the core shroud, the jet pump assemblies, and the reactor coolant, serves to attenuate the fast neutron flux incident upon the reactor vessel wall. The neutron fluence at the vessel wall was calculated. The location of interest for ASME Code Appendix G analysis is the 1/4 T depth, as this is the assumed flaw size in the Appendix G analysis.

The updated predicted peak fluence at the 1/4 T of the RPV beltline is 1.14×10^{18} n/cm^2 for Unit 2, and 1.09×10^{18} n/cm^2 for Unit 3 after 60 years of service (54 EFPY), based on the analysis results documented in NEDC-33873P, Safety Analysis

Report for Peach Bottom Atomic Power Station Units 2 and 3, Thermal Power Optimization (Reference 4). The revised analysis performed as part of the GNF3 fuel introduction does not result in a change to the peak fluence values at the 1/4 T of the RPV beltline but does reduce the P-T Curve EFPY to 49.7 for Unit 2 and 47.5 for Unit 3 (Reference 6).

For the chemical properties of the beltline materials, the shifts in RT_{NDT} predicted by Regulatory Guide 1.99 Revision 2 are less than 91°F. The predicted 60-year adjusted reference temperature provides a substantial margin for brittle fracture prevention, since during operation the vessel need not be pressurized until coolant temperatures exceed 212°F; during hydrostatic testing, vessel temperature is controlled. For further information, see subsection 4.2.5.1, Appendix D of the PBAPS Unit 2 and 3 TRM, or Technical Specification 3.4.9 for PBAPS.

In addition to the minimum requirements of the ASME Boiler and Pressure Vessel Code, the following precautions are taken, and tests made, either to ensure that the initial NDT temperature of the reactor vessel material is low, or to reduce the sensitivity of the material to irradiation effects.

1. The material is selected and fabrication procedures are controlled to produce as fine a grain size as practical. It is an objective in fabrication to maintain a grain size of 5 or finer.

2. Drop weight impact tests are performed on each heat and heat treatment charge of all low alloy steel plate material in its "as-fabricated" condition.

3. Drop weight impact tests are made on the weld metal, the heat-affected zone of the base metal, and the base metal of the weld test plates' simulating seams. If different welding procedures are used for nozzle welds, drop weight tests of similarly prepared coupons are made. The NDT temperature test criteria for the weld and heat-affected zone of the base material are the same as for the unaffected base metal.

4. The actual NDT temperature of the plate opposite the center of the reactor core is determined. The area of the vessel located opposite the core is fabricated entirely of plate and is not penetrated by nozzles, nor are there any other structural discontinuities in this area which would act as stress risers.

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

The reactor coolant system was cleaned and flushed before fuel was initially loaded. During the pre-operational test program, the reactor vessel and reactor coolant system were given a hydrostatic test in accordance with code requirements at 125 percent of design pressure. The vessel temperature is maintained at a minimum of 60°F above the NDT temperature prior to pressuring the vessel for hydrostatic test. An ASME boiler and pressure vessel code, Section XI hydrostatic, or leakage test is performed following each refuel outage. Other pre-operational tests include calibrating and testing the reactor vessel flange seal-ring leakage detection instrumentation, adjusting reactor vessel stabilizers, checking all vessel thermocouples, and checking the operation of the vessel flange stud tensioner.

During the startup test program, the reactor vessel temperatures were monitored during vessel heatup and cooldown to assure that thermal stress on the reactor vessel is not excessive.

4.2.5.1 Brittle Fracture Considerations

4.2.5.1.1 Compliance with 10CFR50 Appendix G

A major condition necessary for full compliance with 10CFR50 Appendix G is satisfaction of the requirements of the Summer 1972 or later Addenda to Section III of the ASME Code. This is not possible with components which were purchased to earlier Code requirements, such as the Peach Bottom vessels with construction code dates of 1968 or earlier.

Ferritic materials complying with current requirements of 10CFR50 Appendix G must have both drop weight tests and Charpy V-Notch (CVN) tests with the CVN specimens oriented transverse to the principal material working direction to establish the reference temperature RT_{NDT}. The CVN tests must be evaluated against both an absorbed energy and a lateral expansion criteria. The maximum acceptable RT_{NDT} must be determined in accordance with the analytical procedures of the ASME Code Section III, NB-2300. Appendix G of 10CFR50 requires a minimum of 75 ft-lb transverse upper-shelf CVN energy for unirradiated beltline materials, and at least 50 ft-lb transverse upper-shelf CVN energy at the end-of-life unless it is demonstrated and approved by the NRC that lower values than 50 ft-lb will provide the necessary margin of safety against fracture. It also requires at least 45 ft-lb CVN energy and 25 mils lateral expansion for bolting material at the lower of the preload or lowest service temperature (LST).

By comparison, materials for the Peach Bottom RPVs were qualified by CVN tests on longitudinal specimens and for some materials by drop weight tests. Charpy tests were generally conducted at only one temperature, confirming that the material met 30 ft-lb CVN energy at the required test temperature of 10°F. There was no upper-shelf CVN energy requirement on the beltline materials. The bolting materials were qualified to a 30 ft-lb CVN energy requirement at 60°F below the minimum preload temperature.

From the above comparison it can be seen that the fracture toughness testing performed on the Peach Bottom RPV materials cannot be shown to comply directly with the requirements of ASME Code Section III NB-2300. However, paragraph III.A of 10CFR50 Appendix G states that an approved method may be used to demonstrate equivalence of pre-1972 Code fracture toughness data with post-1972 Code requirements. The method used to develop RT_{NDT} values is described in Section 4.2.5.1.2.

4.2.5.1.2 Method of Initial RT_{NDT} Evaluation

For the purpose of setting the operating limits the initial RT_{NDT} was determined from the toughness test data taken in accordance with requirements of the Code and General Electric purchase specification to which the RPV was designed and manufactured. These toughness test data, CVN energy and dropweight nil-ductility temperature (NDT), were analyzed to establish compliance with the intent of 10CFR50 Appendix G. Because all toughness testing needed for strict compliance was not required at the time of RPV procurement, some toughness results are not available. To substitute for this absence of certain data, toughness property correlations were derived by General Electric in 1978 for the BWR vessel materials in order to operate upon the available data to give a conservative estimate of RT_{NDT}, compliant with the intent of 10CFR50 Appendix G criteria. These toughness correlations vary, depending upon the specific material analyzed, and were derived from the results of WRC Bulletin 217, "Properties of Heavy Section Nuclear Reactor Steels," and from toughness data for other BWR reactors.

In the case of vessel plate material (SA-302 Grade B, modified), the predicted limiting toughness property is either NDT or transverse CVN 50 ft-lb temperature minus 60°F, whichever is greater. As a matter of practice where NDT results are missing, NDT is estimated as the longitudinal CVN 35 ft-lb transition temperature. However, for the Peach Bottom vessel plates, dropweight NDT information was available. The transverse CVN 50 ft-lb transition temperature was estimated from longitudinal CVN data in the following manner. The lowest longitudinal CVN energy, if below 50 ft-lb, was adjusted to derive a longitudinal CVN 50 ft-lb transition temperature by adding 2°F per ft-lb to the test temperature. If the actual data equaled or exceeded 50 ft-lb, the test temperature was used. Once the longitudinal 50 ft-lb temperature was derived, an additional 30°F was added to account for the orientation change from longitudinal 50 ft-lb to transverse 50 ft-lb.

For forgings (ASTM A508, Class 2), the predicted limiting property is the same as for vessel plates, and the RT_{NDT} was estimated in the same way.

For the vessel submerged arc weld metal the predicted limiting property is the CVN 50 ft-lb transition temperature minus 60°F, as BWR materials experience indicates that dropweight NDT values are typically -50°F or lower for weld materials. The CVN 50 ft-lb temperature was derived in the same way as for the vessel plate material, except the 30°F addition for orientation effects was omitted since there is no principal working direction in weld metal. NDT values were not available, so the RT_{NDT} was the transverse CVN 50 ft-lb transition temperature minus 60°F.
For the vessel electroslag weld, data from numerous weld prolongation were used to characterize the weld properties. There was significant variation in the Charpy curves from each prolongation, all of which were made with the same weld wire heat. Therefore, the 50 ft-lb index temperature for each of nine prolongations was determined and mean RT_{NDT} and standard deviation values were calculated. This approach is consistent with the analysis procedure in Regulatory Guide 1.99, Revision 2.

For the vessel weld heat affected zone (HAZ) material the RT_{NDT} was assumed the same as for the base material as ASME Code weld procedure qualification test requirements and post weld heat treatment data indicate this assumption is valid.

Original closure bolting material (ASTM A540, Grade B24) toughness test requirements were for CVN 30 ft-lb energy at 60°F below the bolt preload temperature. Current 10CFR50 Appendix G requirements are for 45 ft-lb and 25 mil lateral expansion (MLE) at the bolt preload or lowest service temperature (LST). Some closure stud materials do not meet 45 ft-lb absorbed energy at +10°F, and mils lateral expansion results were not reported. Since compliance with current requirements could not be shown, the original requirements were used to establish the closure bolting material LST.

4.2.5.1.3. Calculated Values of Initial RT_{NDT}

The methods of subsection 4.2.5.1.2 were used to calculate initial RT_{NDT} values for the core beltline plates and welds, closure flange region, nozzles and other discontinuities, and LST for the closure bolting material. The calculation methods conservatively estimate RT_{NDT} in order to meet the intent of 10CFR50 Appendix G criteria. Calculated values of initial RT_{NDT} are presented in Table 4.2.5 for Unit 2 and Table 4.2.6 for Unit 3.

4.2.5.1.4 Irradiation Effects on Core Beltline

Estimated maximum changes in RT_{NDT} for the 60 year predicted fluence at the one-quarter thickness (1/4 T) of the vessel beltline materials are listed in Table 4.2.7 for Unit 2 and Table 4.2.8 for Unit 3. The updated predicted peak fluence at the 1/4 T of the RPV beltline is 1.14×10^{18} n/cm^2 for Unit 2, and 1.09×10^{18} n/cm^2 for Unit 3 after 60 years of service (54 EFPY), based on the analysis results documented in NEDC-33873P, Safety Analysis Report for Peach Bottom Atomic Power Station Units 2 and 3, Thermal Power Optimization (Reference 4). The revised analysis performed as part of the GNF3 fuel introduction does not result in a change to the peak fluence values at the 1/4 T of the RPV beltline but does reduce the P-T Curve EFPY to 49.7 for Unit 2 and 47.5 for Unit 3 (Reference 6). The updated fluence prediction is based on a combination of experimental and analytical results.Flux at the surveillance capsule location in the RPV was evaluated by testing flux wires removed after about 7.5 EFPY. The relationship between the capsule location and the peak flux location (lead factor) was determined by a combination of two-dimensional and one-dimensional flux distribution computer analyses. Reference temperature shifts were calculated using the methods of Regulatory Guide 1.99, Revision 2.

Following the original fluence calculations, the NRC issued Regulatory Guide (RG) 1.190, which provides state of the art calculation and measurement procedures that are acceptable to the NRC for determining Reactor Pressure Vessel (RPV) neutron fluence. PBAPS RPV fluence has been evaluated using a method in accordance with the recommendations of RG 1.190. Future evaluations of RPV fluence will be completed using a method in accordance with the recommendations of RG 1.190 (as noted in Reference 2).

Since the predicted 54 EFPY reference temperatures are below 200°F, provisions to permit thermal annealing of the RPV in accordance with Paragraph IV.B of 10CFR50 Appendix G are not required. Upper shelf energy (USE) data available for the electroslag welds, for the beltline plates used in the surveillance program and for some of the submerged arc welds show greater than 50 ft-lb impact energy at 54 EFPY. There are no USE data for the other beltline plates, or for some of the submerged arc welds. However, considering USE data for similar materials, the levels of copper in the plates and weld, and the relatively low fluences predicated at 54 EFPY, it is expected that the beltline materials have adequate USE. These conclusions are not impacted by the revised P-T Curve EFPY values determined for the GNF3 fuel introduction for Unit 2 and 3 (as noted in Reference 6).

An equivalent margin analysis was performed and has been approved by the NRC which confirms that, even in the absence of initial USE data to demonstrate 50 ft-lb USE per previously established NRC methods, adequate margin of safety against fracture equivalent to 10CFR50 Appendix G requirements does exist. This analysis is documented in NEDO-32205-A, Revision 1, "10CFR50 Appendix G Equivalent Margin Analysis for Low Upper Shelf Energy in BWR/2 Through BWR/6 Vessels," and is the basis for demonstrating compliance to Appendix G for Peach Bottom reactor vessels.

4.2.5.1.5 Operating Pressure-Temperature Limits Curves

Operating limit curves are required for the Technical Specifications for three reactor conditions. They are system hydrostatic and leakage tests, non-nuclear heatup/ cooldown and core critical operation. The curves are established by requirements of Section III, Appendix G of the ASME Code and by 10CFR50, Appendix G. Figures 1, 2 and 3 of the PBAPS Unit 2 and Unit 3 TRM Appendix D show these respective operating limit curves for each Unit, including irradiation shift of the core beltline region curves to their positions at 40 years. The Unit 2 operating Limit curve is limited by the region outside of the core beltline region, specifically the upper vessel region. The Unit 3 operating limit curve is non-beltline limited for most of the pressure range and beltline limited for a small range at higher pressures.

4.2.5.1.6 Temperature Limit for Boltup and Pressurization

The minimum temperature for boltup and pressurization of 70°F was established by adding 60°F to the RT_{NDT} for the limiting closure flange region component. The 60°F added to the RT_{NDT} for boltup and pressurization is a requirement of the ASME Code applicable to the original RPV design work. However, Appendix G of the 1998 ASME Code with 2000 Addenda requires a minimum permissible temperature equal only to the RT_{NDT} for boltup and pressurization up to 20% of hydrotest pressure (Paragraph G-2222c). The 60°F added to the RT_{NDT} is extra margin, included because boltup of the closure flange region is one of the most limiting operating conditions for brittle fracture considerations.

4.2.5.1.7 Temperature Limits for Hydrostatic or Leakage Tests

The fracture toughness analysis for pressure tests results in the curves depicted in Figure 1 of the Unit 2 and Unit 3 TRM Appendix D. The curve branches off from a vertical line which represents a requirement of 10CFR50 Appendix G, Paragraph IV.A.2, which says that at pressures above 20% hydrotest, or 312 psig, the minimum temperature for the limiting closure flange region must be at least RT_{NDT} + 90°F. A detailed BWR/6 analysis of the feedwater nozzle, assuming a 1/4 T flaw was adjusted to the feedwater nozzle RT_{NDT}s of 48°F for Unit 2 and 44°F for Unit 3. The Unit 2 pressure-temperature curve is limited by the region outside of the core beltline region, specifically the upper vessel region. The Unit 3 pressure-temperature curve is non-beltline limited for most of the pressure range and beltline limited for a small range at higher pressures.

4.2.5.1.8 Temperature Limits for Non-Nuclear Heatup/Cooldown

The fracture toughness analysis for non-nuclear heatup and cooldown and for low level physics tests results in the curves depicted in Figure 2 of the Unit 2 and Unit 3 TRM Appendix D. The curve was derived from a detailed BWR/6 feedwater nozzle, assuming a 1/4 T flaw, adjusted to the RT_{NDT}s for the feedwater nozzle.

The 10CFR50 Appendix G requirement at 312 psig, which is twenty percent of the preservice system hydrostatic test pressure, is obvious on the pressure-temperature curve as presented in Figure 2 of the Unit 2 and 3 TRM Appendix D. The analysis assumes a normal heatup or cooldown rate of 100°F/hour and it also includes other operation transients such as the effects of cold water injection into the nozzles. The resulting temperature gradients and thermal stress effects are included. The pressure temperature curves in

Figure 2 of the PBAPS Unit 2 and 3 TRM Appendix D depict the RT_{NDT} + 120°F at 312 psig per the requirements of 10CFR50 Appendix G. The Unit 2 pressure-temperature curve is limited by the region outside of the core beltline region, specifically the upper vessel region. The Unit 3 pressure-temperature curve is non-beltline limited for most of the pressure range and beltline limited for a small range at higher pressure.

4.2.5.1.9 Temperature Limits for Core Critical Operation

Appendix G of 10CFR50, Table 1, requires that when the reactor core is critical, the temperature of the RPV metal must be 40°F higher than the highest temperature derived from the same pressure on the other operating limits curves. The resulting curves, depicted in Figure 3 of the Unit 2 and 3 TRM Appendix D, are 40°F above the B curves. The minimum permissible temperature of 70°F is based on the requirement for BWRs in Paragraph IV.A.3 of 10CFR50 Appendix G which allows core critical operation at less than 20% of operating pressure at temperatures of flange region (RT_{NDT} + 60°F) or greater.

4.2.5.1.10 Operating Transients versus Pressure-Temperature Limits

An examination of operating transients was made to determine if any transient conditions violate the curves of Figures 1, 2 and 3 of the PBAPS Unit 2 and Unit 3 TRM Appendix D. Past experience has demonstrated that at high pressures in the vessel (pressures up to 1,180 psig), the bottom head temperature is typically above 250°F. The design basis thermal cycle diagram for Peach Bottom Units 2 and 3, however, indicates that the bottom head temperature can be lower. The worst transient identified is an upset condition, where loss of feedwater pumps and isolation valve closure, cause the reactor to scram.

The worst pressure-temperature combination for this transient is a peak of 1,163 psig with a bottom head temperature of 100°F. This transient would violate the pressure-temperature curves. Operating experience has demonstrated that thermal stratification can occur in the bottom head resulting in temperatures as low as 100°F, but these temperatures typically occur at lower pressures. There have been isolated cases where the bottom head pressure-temperature conditions have exceeded the P-T curve limits, but when actual cooldown rates were considered, the bottom head materials met the requirements of 10CFR50 Appendix G.

4.2.6 Inspection and Testing

Consideration of working space and access for inspection during the design of the reactor vessel allows inspection based on the intent of Section XI of the ASME Boiler and Pressure Vessel Code. The criteria for selecting the components and locations to be inspected are based insofar as practical on this code. Direct visual examination, wherever possible, is sensitive, fast, and positive. Aids to direct visual examination, such as magnetic particle and liquid penetrant inspection, whenever practical, give added sensitivity. Ultrasonic testing and radiography further aid detection where defects can occur on concealed surfaces, as do borescopes or TV cameras in remote locations. The insulation for the cylindrical shell of the vessel is of the standoff type to allow maximum accessibility for inspection. Some insulation panels or portions of panels outside the vessel support are removable to permit inspection of the vessel and vessel support surfaces. All nozzles (except those nozzles inside the vessel support, such as the CRD in-core instrument, and drain nozzles in the bottom head) have insulation designed so that it may be removed to expose the entire exterior of the nozzle and the adjacent vessel shell, and be readily replaced. Original components and locations to be examined were selected taking into account the probability of a defect occurring or enlarging at a certain location, the available accessibility; and Section XI of the ASME boiler and pressure vessel code. For inspection intervals subsequent to the original interval, examinations are performed to satisfy the ASME XI code requirements as specified in 10CFR50.55A. (Ref. Appendix I)

The surveillance test program provides for the preparation of a series of Charpy V-notch impact specimens and tensile specimens from the base metal of the reactor vessel, weld heat-affected zone metal, and weld metal from a reactor steel joint which simulates a welded joint in the reactor vessel. The specimens and neutron monitor wires were placed near core mid-height adjacent to the reactor vessel wall where the neutron exposure is similar to that of the vessel wall. The specimens were installed at startup or just prior to full-power operation. The first of three available capsules were withdrawn from Units 2 and 3 and tested in 1988 and 1989, respectively. The results are documented in GE report SASR 88-24, part of DRF B13-01445, for Unit 2 and in GE report SASR 90-50, part of DRF B11-00494, for Unit 3.

In 2003, the NRC approved PBAPS participation in the BWR Vessel and Internals Project (BWRVIP) Integrated Surveillance Program (ISP) as described in BWRVIP-86-A(Reference 1). The NRC approved the ISP for the industry (Reference 1) and approved PBAPS participation (Reference 2). The ISP meets the requirements of 10 CFR 50 Appendix H and provides several advantages over the original program. The surveillance materials in many plant-specific programs do not represent the best match with the limiting vessel beltline materials since some were established prior to 10 CFR 50 Appendix H requirements. Also, the ISP allows for better comparison to unirradiated material data to determine actual shifts in toughness. Finally, for many plants, ISP data will be available sooner to factor into plant operations since there are more sources of data.

The current withdrawal schedule for both units is as follows and is based on the NRC-approved revision of BWRVIP-86-R1-A (Reference 3).

Unit No. 2	Unit No. 3
1. 7-9 EFPY (7.53 EFPY Actual)	1. 7-9 EFPY (7.57 EFPY Actual)
2. 33.7 EFPY	2. Standby
3. 40 EFPY (~2030)	3. Standby

4.2 REACTOR VESSEL AND APPURTENANCES MACHANICAL DESIGN

REFERENCES

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

2. G. F. Wunder, NRC, to J. L. Skolds, Exelon, Peach Bottom Atomic Power Station, Units 2 and 3 – Issuance of Amendment Re: Revision to the Reactor Pressure Vessel Material Surveillance Program (TAC Nos. MB7006 and MB7007).

3. BWRVIP-86-R1-A, "BWR Vessel and Internals Project, Updated BWR Integrated Surveillance Program (ISP) Implementation Plan," Final Report, October 2012.

4. NEDC-33873P, Safety Analysis Report for Peach Bottom Atomic Power Station Units 2 and 3, Thermal Power Optimization, February 2017.

5. GE Hitachi Nuclear Energy, "PBAPS Units 2 & 3 TPO Task T0301 – RPV Fracture Toughness Evaluation," GEH, Wilmington, NC, 003N5577, October 2016 (PEAM-MUR-0301).

6. GE Hitachi Nuclear Energy, "Peach Bottom Atomic Power Station, Units 2 & 3, GNF3 New Fuel Introduction Statement for the Analysis of Record P-T Curves Applicability," 006N6272, August 2021 (PEAM-MUR-0301, Revision 0A).

TABLE 4.2.1

REACTOR PRESSURE VESSEL MATERIALS

Component	Form	Material	Spec. (ASTM/ASME)
Heads, shell	Rolled plate	Low alloy steel	SA533 Gr B cc 1339
Closure flange	Forged rings	Low alloy steel	SA508 C1 2 cc 1332
Cladding	Weld overlay	Austenitic stainless steel	SA371 type ER309-type ER308 (and carbon content <0.08 w/o)
Nozzles	Forged shapes	Low alloy steel	SA508 C1 2 cc 1332
CRD stub tubes	Forged tubes	Inconel	SB166 cc 1336
CRD housings	Pipe	Austenitic stainless steel	
In-core housings	Pipe	Austenitic stainless steel	

TABLE 4.2.2

REACTOR VESSEL DATA

Reactor Vessel

Security-Related Information Withheld under 10 CFR 2.390

Design pressure and temperature	1,250 psig/575°F

Vessel Nozzles (number and size)

Recirculation outlet	2-28 in
Steam outlet	4-26 in
Recirculation inlet	10-12 in
Feedwater inlet	6-12 in
Core spray inlet	2-10 in
Instrument (one of these is head spray)	2- 6 in
CRD	185- 6 in
Jet pump instrumentation	2- 4 in
Vent	1- 4 in
Instrumentation	6- 2 in
CRD hydraulic system return	1- 4 in
Core differential pressure and liquid control	1- 2 in
Drain	1- 2 in
In-core flux instrumentation	55- 2 in
Head seal leak detection	2- 1 in

Weights (lb)

Bottom head	207,500
Vessel shell	842,300
Vessel flange	105,800
Support skirt	28,200
Other vessel components	65,000
Total Vessel without Top Head	1,248,800
Top Head	252,200
Total Vessel	1,501,000

TABLE 4.2.3

REACTOR VESSEL ATTACHMENTS

Internal Attachments	Number
Guide rod bracket	2
Steam dryer support bracket	4
Dryer holddown bracket	4
Feedwater sparger bracket	12
Jet pump riser support pads	1 ea; 20 places
Jet pump diffuser bracket	1 ea; 20 places
Core spray bracket	8
External Attachments	
Stabilizer bracket	8
Top head lifting lug	4
Insulation supports	2
Insulation support brackets	12 ea; 2 places
Thermocouple pad	36

TABLE 4.2.4

REACTOR DESIGN MONITORING LOCATIONS (60-YEAR LIFE)

LOCATIONS SELECTED FOR CYCLE BASED FATIGUE MONITORING

Component	Includes Environmental Effects
RPV Locations:	
Vessel Shell	Yes
Closure Bolts/Studs	No(2)
Shroud Support	Yes(1)
Jet Pump Shroud Support	No(2)
Core Spray Nozzle	Yes
Recirculation Inlet Nozzle	Yes(1)
Recirculation Outlet Nozzle	Yes(1)
Refueling Containment Skirt	Yes
'A' Feedwater Nozzle	Yes
'D' Feedwater Nozzle	Yes
RPV Support Skirt	No
Torus Locations:	
Torus Shell (both Units)	No(2)
Unit 3 Torus Penetrations	No(2)
Unit 2 Torus Penetrations	No(2)
Piping Locations:	
RHR Return Piping (Stainless Steel)	Yes(1)
RHR Return Piping (Carbon Steel)	Yes(1)
RHR Supply Piping (Stainless Steel)	Yes(1)
Recirculation Piping, RHR Tee	Yes(1)
Feedwater Piping	Yes(1)
Main Steam Piping	Yes

Notes:

(1) Location required per NUREG/CR-6260

(2) Environmental effects do not need to be considered per SIA Calculation 1101048.306 Rev 0.

(3) Cyclic and Transient occurrences are monitored at the above locations to ensure they remain under their design allowable Cumulative Usage Factor (CUF). The following occurrences have been historically expected to contribute to CUF:

TABLE 4.2.4 (continued)

Blowdown Scram
Bolt-up
Chugging
Cooldown
Core Spray Injection
Excessive Cooldown
Excessive Heat-up
FW Temperature Reduction
HPCI Injection
Heatup
Hot Standby
Hydro Test (1563 psi)
Hydrostatic Test
Improper Start Cold Loop
Liquid Control Operation
Loss of Feedpumps
Normal Operation
Operating Basis Earthquake
Partial FW Heater Bypass
RCIC Injection
RHR Injection
Rx OverPRS Delayed Scram
Scrams (All other and Turbine Generator Trip)
SRV Lift
SSE
Sudden Recirc Loop Start
Turbine Bypass
Turbine Roll
Unbolt
Vessel Floodup (Shutdown Cooling in Service)
RWCU Pump Loss/Restart Bolt-up / Unbolt
Hydrostatic Tests
Heat-up / Cooldown
Turbine Roll to 100% Power
Power Reduction
Loss of Feedwater Heaters
Turbine Trip 25% Power
Feedwater Heater Bypass
Scrams (including transients involving Loss of Feedwater Pumps, Isolation Valves Close, Turbine Trip, Reactor Overpressure, Delayed Scram, SRV/SV actuations, etc.)
Improper Start of Cold Recirculation Loop
Feedwater Temperature Reduction
HPCI/ RCIC Injection
Shutdown Cooling Services

Table 4.2.5

EQUIVALENT TRANSVERSE RT_{NDT} VALUES FOR REACTOR VESSEL MATERIALS

FOR UNIT 2

Component	Transverse CVN 50 ft-lb Temperature (°F)	Dropweight NDT Temperature (°F)	RT_{NDT} (°F)
Closure Flanges	40	10	10
Plate Connecting to Closure Flanges	54	10	10
Closure Bolting Material	meets 30 ft-lb at 10°F		10
Beltline Plates	54	-20	- 6
Beltline Welds	15[a]	-	-45
Bottom Head Plate	112106	40	5246

[a] Value based on statistical fit of nine Charpy curve values. Standard deviation equals 16.4°F.

Table 4.2.6

EQUIVALENT TRANSVERSE RT_{NDT} VALUES FOR REACTOR VESSEL MATERIALS

FOR UNIT 3

Component	Transverse CVN 50 ft-lb Temperature (°F)	Dropweight NDT Temperature (°F)	RT_{NDT} (°F)
Closure Flanges	40	10	10
Plate Connecting to Closure Flanges	56	10	10
Closure Bolting Material	meets 30 ft-lb at 10°F		10
Beltline Plates	52	-1010	10
Beltline Welds	15[a]	-	-45
Bottom Head Plate	114	40	54

[a] Value based on statistical fit of nine Charpy curve values. Standard deviation equals 16.4°F.

Table 4.2.7

Adjusted Reference Temperatures 60-Year License (54 49.7 EFPY[5]) - Unit 2

Lower-Intermediate Shell Plates and Axial Welds

Thickness in inches = 6.125

54 49.7 EFPY Peak I.D. fluence = 1.65E+18 n/cm^2
54 49.7 EFPY Peak 1/4 T fluence = 1.14E+18 n/cm^2

Lower Shell Plates, Circumferential Weld and Axial Welds

Thickness in inches = 6.125

54 49.7 EFPY Peak I.D. fluence = 1.24E+18 n/cm^2
54 49.7 EFPY Peak 1/4 T fluence = 8.59E+17 n/cm^2

Water Level Instrumentation Nozzle (Lower-Intermediate Shell)

Thickness in inches = 6.125

54 49.7 EFPY Peak I.D. fluence = 4.81E+17 n/cm^2
54 49.7 EFPY Peak 1/4 T fluence = 3.33E+17 n/cm^2

COMPONENT	HEAT	%Cu	%Ni	CF	Initial RT_{ndt} °F	1/4 T Fluence n/cm^2	54 49.7 EFPY ΔRT_{NDT} °F	σ_1	σ_Δ	Margin °F	54 49.7 EFPY Shift °F	54 49.7 EFPY ART °F
PLANT-SPECIFIC CHEMISTRIES												
PLATES:												
	C2791-2	0.12	0.52	81.4	-8	8.59E+17	31.5	0	15.8	31.5	63.0	55.0
Lower Shell	C2761-1	0.11	0.54	73.4	-14	8.59E+17	28.4	0	14.2	28.4	56.8	42.8
Mark 57	C2873-2	0.12	0.57	82.4	-20	8.59E+17	31.9	0	15.9	31.9	63.8	43.8
	C2894-2	0.13	0.42	85.6	-20	1.14E+18	38.0	0	17.0	34.0	72.0	52.0
Lower-Intermediate Shell Mark 58	C2873-1	0.12	0.57	82.4	-6	1.14E+18	36.6	0	17.0	34.0	70.6	64.6
	C2761-2	0.11	0.54	73.4	-20	1.14E+18	32.6	0	16.3	32.6	65.2	45.2
AXIAL WELDS:												
Lower Shell B1,B2,B3	37C065	0.182	0.181	94.5	-45	8.59E+17	36.6	16	18.3	48.6	75.2	40.2
Lower-Int Shell C1,C2,C3	37C065	0.182	0.181	94.5	-45	1.14E+18	42.0	16	21.0	52.8	94.7	49.7
CIRCUMFERENTIAL WELDS:	S-3986											
	Linde 124											
BC	Lot 3876	0.056	0.96	76.4	-32	8.59E+17	29.6	0	14.8	29.6	59.1	27.1
NOZZLES:												
N16 [1]	C2873-1	0.12	0.57	82.4	-6	3.33E+17	19.2	0	9.6	19.2	38.5	32.5
BEST ESTIMATE CHEMISTRIES from BWRVIP-135 R1												
BC	S-3986	0.058	0.949	79.2	-32	8.59E+17	30.7	0	15.3	30.7	61.3	29.3
INTEGRATED SURVEILLANCE PROGRAM (BWRVIP-135 R1):												
Plate [2]	C2761-2	0.10	0.54	65.0	-20	1.14E+18	28.9	0	14.4	28.9	57.7	37.7
Weld [3]	PB2 ESW	0.10	0.32	84.2	-45	1.14E+18	37.4	0	18.7	37.4	74.8	29.8

Notes:

[1] The N16 Water Level Instrumentation Nozzle occurs in the beltline region. Because the forging is fabricated from Alloy 600 material, the ART is calculated using the plate heats where the nozzles occur.

[2] The ISP plate material is not the vessel target material, but does occur within the Unit 2 beltline region (Lower-Intermediate Shell). Therefore, this material is considered in determining the limiting ART. Only one set of surveillance data is currently available; therefore, upon testing of a second ISP capsule scheduled for 2018, the CF can be reviewed.

[3] The ISP weld material is not the vessel target material and does not occur within the Unit 2 beltline region. Therefore, this material is not considered in determining the limiting ART. The CF is determined using RG 1.99 for the ISP chemistry.

[4] Pressure and temperature limits for various operating states are contained in the Technical Requirements Manual (TRM). The RPV fracture toughness evaluation is contained in Reference 4.2.5.

[5] The results and conclusions of Reference 4.2.5 remain valid after the introduction of GNF3 fuel up to a reduced EFPY of 49.7 as described in Reference 4.2.6.

Table 4.2.8

Adjusted Reference Temperatures 60-Year License (54 47.5 EFPY) - Unit 3

Intermediate Shell Plates and Axial Welds

Thickness in inches = 6.125 — *54 47.5 EFPY Peak I.D. fluence* = 9.65E+17 n/cm^2

54 47.5 EFPY Peak 1/4 T fluence = 6.68E+17 n/cm^2

Lower-Intermediate Shell Plates and Axial Welds

Thickness in inches = 6.125 — 54 47.5 EFPY Peak I.D. fluence = 1.57E+18 n/cm^2

54 47.5 EFPY Peak 1/4 T fluence = 1.09E+18 n/cm^2

Lower Shell Plates, Circumferential Weld and Axial Welds

Thickness in inches = 6.125 — 54 47.5 EFPY Peak I.D. fluence = 9.32E+17 n/cm^2

54 47.5 EFPY Peak 1/4 T fluence = 6.45E+17 n/cm^2

Water Level Instrumentation Nozzle (Lower-Intermediate Shell)

Thickness in inches = 6.125 — 54 47.5 EFPY Peak I.D. fluence = 4.57E+17 n/cm^2

54 47.5 EFPY Peak 1/4 T fluence = 3.16E+17 n/cm^2

COMPONENT	HEAT	%Cu	%Ni	CF	Initial RT_{ndt} °F	1/4 T Fluence n/cm^2	54 47.5 EFPY ΔRT_{NDT} °F	σ_1	σ_Δ	Margin °F	54 47.5 EFPY Shift °F	54 47.5 EFPY ART °F
PLANT-SPECIFPLATES: Lower Shell												
6-146-1												
6-146-3	C4689-2	0.12	0.56	82.2	-10	6.45E+17	27.5	0	13.8	27.5	55.1	45.1
6-146-7	C4684-2	0.13	0.58	90.4	-20	6.45E+17	30.2	0	15.1	30.3	60.6	40.6
	C4627-1	0.12	0.57	82.4	-20	6.45E+17	27.6	0	13.8	27.6	55.2	35.2
Lower-Intermediate Shell												
6-139-10	C2773-2	0.15	0.49	104.0	10	1.09E+18	45.1	0	17.0	34.0	79.1	89.1
6-139-11	C2775-1	0.13	0.46	86.8	10	1.09E+18	37.7	0	17.0	34.0	71.7	81.7
6-139-12	C3103-1	0.14	0.6	100.0	10	1.09E+18	43.4	0	17.0	34.0	77.4	87.4
Intermediate Shell												
6-146-5	C4608-1	0.12	0.55	82.0	10	6.687.41E+17	28.029.5	0	14.07	28.029.5	55.959.0	65.969.0
6-146-4	C4689-1	0.12	0.56	82.2	10	6.687.41E+17	28.029.6	0	14.08	28.029.6	56.159.1	66.169.1
6-146-2	C4654-1	0.11	0.55	73.5	10	6.687.41E+17	25.126.4	0	12.513.2	25.126.4	50.152.9	60.162.9
AXIAL WELDS:												
Lower Shell D1,D2,D3	37C065	0.182	0.181	94.5	-45	6.45E+17	31.7	16	15.8	45.0	76.7	31.7
Lower-Int Shell E1,E2,E3	37C065	0.182	0.181	94.5	-45	1.09E+18	41.0	16	20.5	52.0	93.0	48.0
Intermediate Shell F1,F2,F3	37C065	0.182	0.181	94.5	-45	6.687.41E+17	32.234.0	16	16.117.0	45.446.7	77.780.6	32.735.6
CIRCUMFERENTIAL WELDS:												
Lower to Lower-Int DE	3P4000 Linde 124 Lot 3932	0.020	0.934	27.0	-50	6.45E+17	9.0	0	4.5	9.0	18.1	-31.9
Lower-Int to Int EF	1P4217 Linde 124 Lot 3929	0.102	0.942	136.9	-50	6.687.41E+17	46.749.2	0	23.324.6	46.749.2	93.498.5	43.448.5
NOZZLES:												
N16 [1]	C4689-1	0.12	0.56	82.2	10	3.16E+17	18.6	0	9.3	18.6	37.2	47.2
BEST EST. CHEMISTRIES from BWRVIP-135 R1												
DE	3P4000	0.020	0.935	27.0	-50	6.45E+17	9.0	0	4.5	9.0	18.1	-31.9
EF	1P4217	0.104	0.938	139.3	-50	6.687.41E+17	47.550.1	0	23.825.1	47.550.1	95.0100.2	45.050.2

INTEGRATED SURVEILLANCE PROGRAM (BWRVIP-135 R1):												
Plate [2]	B0673-1	0.15	0.65	111.25	10	1.09E+18	48.3	0	17.0	34.0	82.3	92.3
Weld [3]	5P6756	0.06	0.93	82.0	-45	1.09E+18	35.6	0	17.8	35.6	71.1	26.1
Weld [3]	5P6756 [4]	0.08	0.94	108.0	-45	1.09E+18	46.9	0	23.4	46.9	93.7	48.7

Table 4.2.8 (continued)

Notes:

[1] The N16 Water Level Instrumentation Nozzle occurs in the beltline region. Because the forging is fabricated from Alloy 600 material, the ART is calculated using the plate heats where the nozzles occur.

[2] The ISP plate material is not the vessel target material and does not occur within the Unit 3 beltline region. Therefore, this material is not considered in determining the limiting ART. The CF is determined using RG 1.99 for the ISP chemistry.

[3] The ISP weld material is not the vessel target material and does not occur within the Unit 3 beltline region. Therefore, this material is not considered in determining the limiting ART. The CF is determined using RG 1.99 for the ISP chemistry.

[4] The ISP best estimate chemistry is used.

[5] Pressure and temperature limits for various operating states are contained in the Technical Requirements Manual (TRM). The RPV fracture toughness evaluation is contained in Reference 4.2.5.

[6] The results and conclusions of Reference 4.2.5 remain valid after the introduction of GNF3 fuel up to a reduced EFPY of 47.5 as described in Reference 4.2.6.

4.3 REACTOR RECIRCULATION SYSTEM

4.3.1 Power Generation Objective

The power generation objective of the reactor recirculation system is to provide a variable moderator (coolant) flow to the reactor core for adjusting reactor power level.

4.3.2 Power Generation Design Basis

1. The reactor recirculation system provides sufficient coolant circulation through the core during normal power operation to maintain normal operating temperatures.

2. The reactor recirculation system operates over a flow control range of 20 percent to 100 percent flow to allow power variation.

3. The reactor recirculation system is designed to minimize maintenance situations that would require fuel removal.

4.3.3 Safety Design Basis

1. The reactor recirculation system is designed so that adequate fuel barrier thermal margin is assured following recirculation pump system malfunctions.

2. The reactor recirculation system is designed so that failure of piping integrity does not compromise the ability of the reactor vessel internals to provide a refloodable volume.

3. The reactor recirculation system is designed to withstand adverse combinations of loading and forces resulting from operation during abnormal, accident, and special event conditions.

4. The reactor recirculation system is designed to insert negative reactivity by tripping the recirculation pumps to protect the fuel cladding fission product barrier during the following operational transients:

   a. Turbine Trip with No Bypass
   b. Load Reject with No Bypass
   c. Feedwater Controller Failure Maximum Demand

4.3.4 Description

The reactor recirculation system consists of the two recirculation pump loops external to the reactor vessel which provide the driving flow of water to the reactor vessel jet pumps (Figure 4.3.1, Drawings M-351 and M-353). Each external loop contains one high-capacity, motor-driven recirculation pump and two motor-operated gate valves for pump maintenance. Each pump discharge line contains a venturi-type flowmeter.

The recirculation loops are a part of the nuclear system process pressure boundary and are located inside the drywell. The jet pumps are reactor vessel internals and their location and mechanical design are discussed in subsection 3.3, "Reactor Vessel Internals Mechanical Design"; however, certain operational characteristics of the jet pumps are discussed in this subsection. A summary of the characteristics of the reactor recirculation system is presented in Table 4.3.1. The values provided in Table 4.3.1 are the original design characteristics of the Reactor Recirculation system. Operational parameters such as flow, pump horsepower, and Adjustable Speed Drive (ASD) power requirements will vary throughout the fuel cycle based on core conditions and jet pump efficiencies.

The recirculated coolant consists of saturated water from the steam separators and dryers which has been subcooled by incoming feedwater. This water passes down the annulus between the reactor vessel wall and the core shroud. A portion of the coolant exits from the vessel and passes through the two external recirculation loops to become the driving flow for the jet pumps. The two external recirculation loops each discharge high-pressure flow into an external manifold from which individual recirculation inlet lines are routed to the jet pump risers within the reactor vessel. The remaining portion of the coolant mixture in the annulus becomes the driven flow for the jet pumps. This flow enters the jet pumps at the suction inlet and is accelerated by the driving flow. The driving and driven flows are mixed in the jet pump throat section, resulting in partial pressure recovery. The balance of recovery is obtained in the jet pump diffusing section (Figure 4.3.3). The adequacy of the total flow to the core is discussed in subsection 3.7, "Thermal and Hydraulic Design." Tests have been conducted and documented to show that the jet pump design is sound and that jet pump operation is stable and predictable.

Since the removal of reactor recirculation system valve internals requires unloading the nuclear fuel, the valves are provided with high quality backseats and trim to facilitate stem packing renewal and to provide adequate leak tightness. The design objective of the backseats and trim is to provide a 20-yr service life.

For Unit 3 only, the two pump discharge valves MO-53A and MO-53B have a small relief hole in the downstream disc to prevent pressure locking of the valves.

It is possible to operate with one recirculation pump. The idle pump loop is not completely valved off if it is desired to return the idle loop to service prior to the next reactor cooldown. The recirculation pump casing's allowable heatup rate is 100°F/hr, the same as the reactor vessel. It is possible to keep the idle loop hot with the idle loop valves left open, permitting the pressure head created by reverse flow through the idle jet pumps to cause reverse flow through the idle loop. Also, if it is desired to return the idle loop to service, PBAPS Technical Specifications requires that the temperature of an idle recirculation loop be within limits as specified in the Pressure Temperature Limits Report (PTLR).

The feedwater flowing into the reactor vessel annulus during operation provides subcooling for the fluid passing to the recirculation pumps, thus providing the additional NPSH available beyond that provided by the pump location below the reactor vessel water level. If feedwater flow is below 20 percent, the recirculation pump speed is automatically limited. Automatic protection against recirculation pump cavitation is not provided, and operating procedures are relied upon for pump protection. With one pump delivering 100 percent of its rated flow, approximately 50 percent feedwater flow is required to provide adequate NPSH, due to the subcooling.

The recirculation pumps can be operated during nuclear system heatup for hydrostatic tests. At this time, they act in conjunction with any contribution from reactor core decay heat to raise nuclear system temperature above the limit imposed on the reactor vessel by NDT temperature considerations so that the hydrostatic test can be conducted.

Decontamination connections are provided in the piping on the suction and discharge sides of the pump for Unit 3 as shown in Drawing M-353, Sheets 3 and 4, to permit flushing and decontamination of the pump and adjacent piping. These connections are arranged for the convenient and rapid connection of temporary piping. The piping low point drain is used, during flushing or decontamination, to conduct crud away from the piping low point, and is also designed for the connection of temporary piping.

Each recirculation pump is a single-stage, variable speed, vertical, centrifugal pump equipped with mechanical shaft seal assemblies. The pump is capable of stable and satisfactory performance while operating continuously at any speed corresponding to a power supply frequency range of 11.5 to 57.5 Hz. For loop startup, each pump operates at a pump speed of approximately 28 to 30 percent of rated speed corresponding to a power supply frequency of approximately 17 Hz.

The recirculation pump shaft seal assembly consists of two seals built into a cartridge which can be readily replaced without removing the motor from the pump. The seal assembly is designed to require minimum maintenance over a long period of time, regardless of whether the pump is stopped or operating. Each individual seal in the cartridge is capable of sealing against pump design pressure so that any one seal can adequately limit leakage in the event that the other seal should fail. A breakdown orifice is provided in the pump casing to reduce leakage in the event of a gross failure of both shaft seals. Provision is made for monitoring the pressure drop across each individual seal, as well as the cavity temperature of each seal. The measurement of the seal leakage is performed by monitoring the drywell sump, pump out cycles and pump out cycle times. Reactor recirculation pump seal integrity is maintained by monitoring seal cavity pressure, seal cavity temperature and staging flow.

The recirculation seal assembly is provided with a purge flow to keep the seals clean by maintaining clean water flowing out of the seal area, along the pump shaft, and into the recirculation system. The recirculation pump seal purge system takes condensate water from the CRDS upstream of the CRD flow control valves, FCV-19A and FCV-19B, and routes it through a motor-operated valve, a flow regulating valve, and a rotometer, to the recirculation pump seal cavities. The flow enters the seal cavities downstream of the existing flow check valves on the pressure indication taps. Seal purge system isolation valve operation can be initiated automatically upon a recirculation pump start or stop signal, or placed in service manually.

Each recirculation pump motor is a variable speed ac motor capable of driving the pump over a controlled range of 20 percent to 102 percent of rated pump speed. The motor is designed to operate continuously at any speed within the power supply frequency range of 11.5 to 57.5 Hz. A variable frequency ASD located outside the drywell supplies power to each recirculation pump motor. Electrical equipment is designed, constructed, and tested in accordance with the applicable sections of the NEMA standards.

The ASD provides power to drive the Recirculation Pump motors using static power cell. Each power cells consists of a three-phase rectifier bridge that feeds a DC-capacitor bank. There is no inherent inertia driven coastdown power or braking force applied to the recirculation pump motor upon ASD stop or trip. The GNF2 ECCS-LOCA analysis demonstrated compliance with the 10 CFR 50.46 acceptance criteria, and concluded that the coastdown rate of the ASD is acceptable.

To ensure that the pump casing for both units can withstand a pressure equivalent to that inside the reactor vessel, the pump casing is designed using the ASME Boiler and Pressure Vessel Code, Section III, for Class C vessels as a guide. This class is used because the pump casing does not experience temperature transients as severe as those that portions of the reactor vessel and certain piping connections experience; therefore, it is not necessary to make the cyclic analysis required for Class A equipment.

The recirculation pumps' nonpressure retaining parts are classified as machinery and, as such, are specifically exempt from the jurisdiction of any section of the ASME Boiler and Pressure Vessel Code or the ANSI code for pressure piping. The Standards of the Hydraulic Institute are used for performance testing of the pumps.

The design objective for the recirculation pump casing is a useful life of 40 yr, accounting for corrosion, erosion, and material fatigue. The pump drive motor, impeller, and wear rings are designed for as long a life as is practical. The castings and forgings of the pumps are fabricated from austenitic stainless steel. The design objective is to provide a unit which will not require removal from the system for rework or overhaul at intervals of less than 5 yr.

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

The coolant in the nuclear process system is at high pressure and contains a large amount of energy. Substantial failure of the nuclear process system could result in a rapid loss of coolant. Although loss of the moderator (coolant) would render the reactor core subcritical, lack of cooling could cause overheating of the reactor core, from residual heat, leading to fuel damage and fission product release. The core standby cooling systems (which adequately cool the reactor core following a design basis LOCA), and the primary containment and containment cooling systems (which control the release of fission products and absorb the energy released by the accident), are not intended to diminish the overall design objective of the entire nuclear system (to design and construct a nuclear system which will not fail). The intent of using Section III of the ASME Boiler and Pressure Vessel Code for the recirculation system is to provide piping systems of a quality equivalent to the reactor pressure vessel to which it is attached.

The reactor recirculation system, except for ASDs, is designed in accordance with seismic Class 1 criteria. Vibration snubbers, located at the top of the motor and at the bottom of the pump casing, are designed to resist seismic forces.

The recirculation piping, valves, and pumps are supported by constant and variable support hangers, to avoid the use of piping expansion loops which would be required if the pumps were anchored. In addition, the recirculation loops are provided with a system of supports designed to limit pipe motion so that reaction forces associated with any split or circumferential break do not jeopardize containment integrity. This support system provides adequate clearance for normal thermal expansion movement of the loop. The spacing between limit stops is set on the basis that a split pipe retains its structural load resisting characteristics. Impact loading is not considered on limit stops since possible pipe movement is limited to slightly more than the clearance required for thermal expansion movement.

The recirculation system piping, valves, and pump casings are covered with thermal insulation having an average maximum heat transfer rate of 65 Btu/hr-sq ft with the system at rated operating conditions. The insulation is prefabricated into sections for field installation. The insulation is removable for inspection of the equipment and piping.

4.3.5 Safety Evaluation

Reactor recirculation system malfunctions that pose threats of damage to the fuel barrier are described and evaluated in Section 14.0, "Plant Safety Analysis." It is shown that none of the malfunctions results in fuel damage; thus, the recirculation system has sufficient flow coast down characteristics to maintain fuel thermal margins during operational transients. This satisfies safety design basis 1.

The core flooding capability which is provided by a jet pump design plant is pictured in Figure 4.3.4. There is no recirculation line break which can prevent reflooding of the core to the level of the jet pump suction inlet. The core flooding capability of a jet pump design plant is discussed in a General Electric topical report. This satisfies safety design basis 2.

The reactor recirculation system piping and pump design pressures are based on peak steam pressure in the reactor dome plus the static head above the lowest point in the recirculation loop. Piping and related equipment pressure parts are chosen in accordance with applicable codes. Use of the listed code design criteria provides assurance that a system designed, built, and operated within design limits has an extremely low probability of failure due to any known failure mechanism.

4.3.6 Inspection and Testing

Quality control methods are used during the fabrication and assembly of the reactor recirculation system to ensure that the design specifications are met. The reactor coolant system is thoroughly cleaned and flushed before fuel is loaded initially. During the pre-operational test program, the Peach Bottom 3 and Peach Bottom 2 initial reactor recirculation system was given a hydrostatic test at 125 percent of reactor vessel design pressure. A hydrostatic test, at a pressure not to exceed system operational pressure, is made following each removal and replacement of the reactor vessel head. During the pre-operational test subsequent to recirculation pipe replacement, the piping was pressure tested at 105 percent of the system operational pressure. Other pre-operational tests on the reactor recirculation system included operating valves and verifying that seat leakage is small enough to permit pump maintenance work, operating pumps and ASDs, and checking flow control transient operation.

During the initial and post pipe replacement startup test programs, the horizontal and vertical motions of the reactor recirculation system piping and equipment were observed, and adjustments of supports were made as necessary to assure that components are free to move as designed. Nuclear system responses to recirculation pump trips at rated temperatures and pressure are evaluated during the startup tests, and the plant power response to recirculation flow control is determined.

4.3 REACTOR RECIRCULATION SYSTEM

REFERENCES

1. "Design Performance of G.E. BWR Jet Pumps," General Electric Company, Atomic Power Equipment Department, APED-5460, September 1968.

2. Ianni, P. W., "Core Standby Cooling Systems for Boiling Water Reactors," General Electric Company, Atomic Power Equipment Department, APED-5458, March 1968.

TABLE 4.3.1

REACTOR RECIRCULATION SYSTEM
DESIGN CHARACTERISTICS

External Loops	
Number of Loops	2
Pipe Sizes (nominal O.D.)	
Pump Suction, in	28
Pump Discharge, in	28
Discharge Manifold, in	22
Recirculation Inlet Line, in	12
Cross-Tie Line, in	22
Design Pressure (psig)/Design Temperature (°F)	
Suction Piping	1250/575
Discharge Piping	1500/575
Pumps	1500/575
Operation at Rated Conditions	
Recirculation Pump	
Flow gpm (nominal), gpm	48,106
Maximum Drive Flow (gpm)	51,000
Flow, lb/hr	17.1×10^6
Total Developed Head, ft	750
Suction Pressure (static), psia	1054
Available NPSH* (min.), ft	500
Water Temperature (max.), °F	532
Pump Hydraulic HP (min.), hp	6130
Flow Velocity at Pump Suction, fps (approximate)	27.5
Drive Motor and Power Supply	
Frequency (at rated), Hz	56
Frequency (operating range), Hz	11.5-57.5
Total Required Power to M-G Sets	
kW/set	6730
kW total	13,460
Total Required Power to the ASDs	
Efficiency	96.5% Rated
Power (kW) Output at 57.5 Hz	6752
Clamped Power (kW) Output at 55.9 Hz	6189
Maximum Power (kW) Output at 58.1 Hz	6960
kW/ASD	6997
kW total	13,994
Jet Pumps	
Number	20
Total Jet Pump Flow, lb/hr (rated conditions)	102.5×10^6
Total Jet Pump Flow, lb/hr	112.75×10^6

(during recirc rerate conditions)	
Throat I.D., in	8.18

TABLE 4.3.1 (continued)

Diffuser I.D., in	19.0
Nozzle I.D., in (representative)	3.14
Diffuser Exit Velocity, fps	15.3
Jet Pump Head, ft	76.1

* Includes velocity head.

4.4 NUCLEAR SYSTEM PRESSURE RELIEF SYSTEM

NOTE: The nuclear system pressure relief system is also known as the automatic blowdown system.

4.4.1 Safety Objective

The safety objective of the nuclear system pressure relief system is to prevent overpressurization of the nuclear system, which protects the nuclear system process barrier from failure and could result in the uncontrolled release of fission products. In addition, the automatic depressurization feature of the nuclear system pressure relief system acts in conjunction with the core standby cooling systems for reflooding the core following small breaks in the nuclear system process barrier. This protects the reactor fuel cladding from failure due to overheating.

4.4.2 Safety Design Basis

1. The nuclear system pressure relief system prevents overpressurization of the nuclear system thus preventing failure of the nuclear system process barrier due to pressure.

2. The nuclear system pressure relief system provides automatic nuclear system depressurization for small breaks in the nuclear system so that the LPCI and the core spray systems can operate to protect the fuel barrier. This depressurization is dependent on the availability of LPCI or core spray.

3. The relief valve discharge piping is designed to accommodate forces resulting from relief action, and is supported for reactions due to flow at maximum relief valve discharge capacity so that system integrity is maintained. Relief valve piping has also been designed to withstand suppression pool loads including pool swell resulting from a LOCA.

4. The nuclear system pressure relief system is designed for testing prior to nuclear system operation and for verification of the operability of the nuclear system pressure relief system.

5. The nuclear system pressure relief system is designed to withstand adverse combinations of loadings and forces

resulting from operation during abnormal, accident, or special-event conditions.

6. The nuclear system pressure relief valves are designed to accommodate liquid flow in the event that the "alternate shutdown cooling mode" is used.

   The relief valves would be used as part of the coolant flow path, as discussed in paragraph G.5.3, event 21.

4.4.3 Power Generation Objective

The power generation objective of the nuclear system pressure relief system is to relieve any overpressure which occurs during abnormal operational transients.

4.4.4 Power Generation Design Basis

1. The nuclear system relief valves relieve overpressure without the assistance of the safety valves during selected plant transients, as described in Section 14.5.1, "Plant Safety Analysis."

2. The nuclear system relief valves discharge to the primary containment suppression pool.

3. The relief valves properly reclose following operation.

4.4.5 Description

The nuclear system pressure relief system includes three safety and eleven relief valves, all of which are located on the main steam lines within the drywell, between the reactor vessel and the first isolation valve (Table 4.4.1).

The safety and relief valves are mounted on the four main steam lines so that a single accident cannot completely disable a safety, relief, or automatic depressurization function. See Drawing M-351, Sheets 1 through 4 for schematic location and layout details of the valves and piping. The relief valves, which discharge to the suppression pool, provide three main protection functions:

1. Overpressure relief operation. The valves are opened (by overpressure or remote manual operation) to limit nuclear system pressure rise and preclude safety valve opening.

2. Overpressure safety operation. The valves augment the safety valves by opening (self-actuated operation only) in order to prevent nuclear system overpressurization.

3. Depressurization operation. When required as part of the core standby cooling systems, the relief valves operate to provide automatic or manual depressurization.

The relief valves are designed, constructed, and marked with data in accordance with the ASME Boiler and Pressure Vessel Code, Section III, Article 9 (1968), and in accordance with ANSI B31.1 (1968) and B16.5 (1967). Popping-point tolerance is in accordance with ASME Boiler and Pressure Vessel Code, Section I (1965), paragraph PG-72 (c). Each valve is self-actuating at the set relieving pressure, but may also be actuated by remotely operated devices that permit remote manual or automatic opening at lower pressures. For depressurization operation, each relief valve is provided with a power-actuated device capable of opening the valve at any steam pressure above 100 psig, and capable of holding the valve open until the steam pressure decreases to about 50 psig. The control system for the actuator is described in subsection 7.4, "Core Standby Cooling Systems Control and Instrumentation." Pressure-containing parts of the valve body are fabricated of ASTM A216, Grade WCB. The relief valve is designed for operation with saturated steam containing less than 1-percent moisture. The relieving pressures for overpressure relief and safety operating modes are adjustable between 1,025 and 1,155 psig with a maximum back pressure of 40 percent of the set pressure. The delay time (maximum elapsed time between overpressure signal and actual valve opening) is equal to or less than 0.55 sec.

Criteria for the design and installation of relief valves and safety valves include the following:

1. Discharge tees are provided on safety valves to equalize the discharge thrust force.

2. Flanges are installed to ensure vertical tolerances.

3. Clearance of at least 6 in is provided between valves and other equipment.

4. Space greater than (2t + 2 in) is provided between welds of sweepolet on header for inspection.

5. Clearance is provided between header and bottom of flange for bolt removal from installed valve.

6. Flatness tolerance on surface of groove is provided on the safety and relief valve flanges.

7. A flange rating of 1,500 lb is provided for structural stability rather than the lower rating corresponding to the pressure temperature rating.

8. A pipe schedule of 160 was used for structural stability rather than the Schedule 80 required for pressure/temperature.

For analysis, the special loadings listed below are considered in addition to the usual design loads such as weight, pressure, temperature, and earthquake.

1. The jet force exerted on the relief and safety valves during the first millisecond when the valve is open and steady-state flow has not yet been established. (With steady-state flow, the dynamic flow reaction forces will be self-equilibrated by the relief valve discharge piping or the tee at the safety valve discharge.)

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

All code-allowable stresses are met with these special loads acting concurrently with other design loads. The highest stress is at the branch connection below the valve. The results of these analyses are contained in Appendix C, Table C.5.7.

The safety valves provide additional protection against overpressure of the nuclear system and discharge directly to the interior space of the drywell.

The <u>safety valves</u> are spring-loaded valves which are designed, constructed, and marked with data in accordance with the ASME Boiler and Pressure Vessel Code, Section III, Article 9 (1968), and in accordance with ANSI B31.1 (1968) and B16.5 (1967). Popping-point tolerance is in accordance with ASME Boiler and Pressure Vessel Code, Section I (1965), Paragraph PG-72 (c). The valves are designed for operation with saturated steam containing less than 1 percent moisture and are designed to have an opening response time equal to or less than 0.2 sec.

The materials used in the valve fabrication are as follows:

Safety Valves

Body	ASTM A216 Gr. WCB
Nozzle	ASTM A182 Gr. A347
Disc	A1S1 422 ASME SB-637 nickel alloy N07750 Type 3

Relief Valves

Body	ASTM A216 Gr. WCB
Base	ASTM A216 Gr. WCB
Disc	ASTM A276 Type 304 or approved alternate
Seat	ASTM A107 Gr. 1018 or approved alternate

The only component which may have become sensitized during manufacture is the disc of the relief valves. This component is made of Type 304 stainless steel and has been hardfaced.

Each relief valve consists of a main valve disc and piston, operated by a second stage disc and piston displaced by either a pressure-sensing pilot or a pneumatically operated mechanical push rod. A typical sequence of operation for overpressure relief self-actuation can be described as follows (refer to Figures 4.4.1 and 4.4.2):

a. In the closed position (Figure 4.4.1), the bellows is mechanically extended a slight amount by the preload spacer to provide a preload force on the pilot disc. This seats the pilot valve tightly and prevents reverse leakage at low system pressures or high back pressures. The main valve disc is tightly seated by the combined forces exerted by the main valve preload spring and the system internal pressure acting over the area of the main valve disc. In the closed position, the static pressures will be equal in the valve body and in the chamber over the main valve piston. This pressure equalization is made possible by leakage through the piston orifice.

b. As system pressure increases, the preload force on the pilot disc is reduced to zero as the bellows is extended farther, and the disc is held closed by the internal pressure acting over the pilot valve seat area. The upstream seating force, which is significantly greater than the initial preload, increases with increasing system pressure and prevents leakage or "simmering" at pressures near the valve set pressure.

c. As system pressure further increases, bellows expansion reduces the abutment gap between the stem and the disc yoke. When the stem abuts against the yoke, further pressure increase reduces the net pilot seating force to zero and lifts the first stage pilot valve from its seat.

d. Once the pilot valve starts to open, the upstream seating force is eliminated, resulting in a net increase in the force tending to open the pilot valve. This increase in net force produces the "popping" action during pilot valve opening (Figure 4.4.2).

e. Opening of the first stage pilot valve admits fluid to the operating piston of the second stage valve, causing it also to open.

f. Opening of the second stage pilot valve vents the chamber over the main valve piston to the downstream side of the valve. This venting action creates a differential pressure across the main valve piston almost equal to the system pressure and in a direction tending to open the valve. The main valve piston is sized so that the resulting opening force is greater than the combined preload and upstream seating force. Therefore, opening the pilot opens the main valve.

g. As in the case of the pilot valve, once the main valve disc starts to open, the upstream seating force is reduced, causing a significant increase in opening force and the characteristic full opening or "popping" action.

h. When the pressure has been reduced sufficiently to permit the pilot valve to close, leakage of system fluid past the main valve piston repressurizes chamber over the piston, eliminates the upstream opening force, and permits the preload spring to close the valve. Once closed, the additional upstream seating force due to system pressure acting on the main valve disc seats the main valve tightly and prevents leakage.

The relief valves are installed so that each valve discharge is piped through its own discharge line to a point below the minimum water level in the primary containment suppression pool to permit the steam to condense in the pool. Each relief valve discharge line terminates in a T-quencher (sparger) which serves to reduce pressures on the torus shell during the initial blowdown of air and water into the pool after a relief valve opens. The quencher also provides uniform and stable condensation of steam in the suppression pool. The relief valves are located on the main steam line piping, rather than on the reactor vessel top head, primarily to simplify the discharge piping to the pool and to avoid the necessity for removing sections of this piping when the reactor head is removed for refueling. A vacuum relief valve is provided on each relief valve discharge line to prevent drawing water up into the line due to steam condensation following termination of relief valve operation.

A pressure-actuated switch is provided to indicate by an indicating light and alarm that a leak has occurred in the pilot valve bellows.

The automatic depressurization feature of the nuclear system pressure relief system serves as a backup to the HPCIS under LOCA conditions. If the HPCIS does not operate, and one of the LPCI or two of the core spray pumps are available, as determined by sensing discharge pressure, the nuclear steam system is depressurized sufficiently to permit the LPCI and core spray systems to operate to protect the fuel barrier. Depressurization is accomplished through automatic opening of any or all of the five ADS valves (RV-71A, B, C, G, K) to vent steam to the suppression pool. For small line breaks, if the HPCIS fails, the nuclear steam system is depressurized in sufficient time to allow the core spray or LPCI systems to provide core cooling to prevent excessive fuel cladding temperatures. The signal for the relief valves to open and remain open is based upon signals from: (1) drywell high pressure, (2) reactor vessel low water level, and (3) availability of one of the LPCI or two core spray pumps. Any combination of CS pumps running except A and B or C and D will satisfy the requirement. The high drywell permissive is bypassed after an extended time period if conditions result in low reactor level without the presence of high drywell pressure. For large breaks, the vessel depressurizes rapidly through the break without assistance.

Normal pneumatic supply is from the plant instrument nitrogen system. This system takes suction from the drywell atmosphere, compresses the gas, and returns it as a supply to all pneumatic services within the primary containment in order to prevent degradation of the inert atmosphere. The instrument nitrogen compressors are not capable of being powered from on-site electric power. Refer to Subsection 10.17 for a discussion of the instrument nitrogen system.

When low pressure is sensed in the instrument nitrogen receiving tanks, the station instrument air system is automatically switched into service as a backup (subsection 10.17).

A long-term, safety grade, pneumatic supply has been provided for the ADS valves. A separate and split ring header is located inside containment with three ADS valves connected to one section of the split header and the remaining two ADS valves connected to the other section of the split header. The source of safety grade pneumatic pressure is a series of nitrogen cylinders located within the reactor building with a connection provided outside the reactor building for the installation of additional bottles, as required.

Each of the five relief valves provided for automatic depressurization is equipped with an accumulator and check valve arrangement. Unit 2 values have one accumulator, Unit 3 valves have two accumulators. These accumulators are provided to ensure that the valves can be held open following failure of the supply to the accumulators, and are sized for a minimum of five valve operations at a drywell pressure of one atmosphere.

Containment isolation is provided for safety grade pneumatic supply lines into containment by use of check valves and other automatic valves outside primary containment. The outer, automatic valves are manually controlled from the control room and automatically close on low differential pressure between pneumatic supply pressure and containment pressure or if after a time delay, gas flow becomes excessively high. For purposes of containment isolation, a pressure transmitter is located in each pneumatic supply and its signal is compared to that of a drywell pressure transmitter. Flow transmitters are also located in the supply piping to monitor for excessive flow (subsection 7.3).

A long-term, backup, safety grade pneumatic nitrogen supply is available to enable remote operation of the safety relief valves for a period of 72 hours following a design basis fire. The source of the pneumatic nitrogen supply is the Safety Grade Instrument Gas (SGIG) system. The SGIG system is tied into the 6000 gallon liquid nitrogen tank which supplies the Containment

Atmospheric Dilution (CAD) system. See FPP Table A-4 for actions required to credit this pneumatic supply.

Further descriptions of the operation of the automatic depressurization feature are found in Section 6.0, "Core Standby Cooling Systems," and subsection 7.4, "Core Standby Cooling Systems Control and Instrumentation."

A manual depressurization of the nuclear system can be effected in the event the main condenser is not available as a heat sink after reactor shutdown. The steam generated by core decay heat is discharged to the suppression pool. The relief valves are operated by remote-manual controls from the main control room.

A direct method of indicating the position of safety/relief and safety valves is provided. The system is based on acoustical monitoring techniques (paragragh 7.20.4.9).

The number, and set pressures of the relief and safety valves, and the capacities of the relief valves are shown in Table 4.4.1.

4.4.6 Safety Evaluation

The ASME Boiler and Pressure Vessel Code requires that a vessel designed to meet Section III be protected from pressure in excess of the vessel design pressure. A peak allowable pressure of 110 percent of the vessel design pressure is allowed by the code. The code specifications for safety valves require that (1) the lowest safety valve be set at or below vessel design pressure, and (2) the highest safety valve be set to open at or below 105 percent of vessel design pressure.

The reactor dome pressure remains unchanged at 1035 psig in conjunction with rerating the plant to 4016 MWt. The relief valve nominal setpoints (1135 to 1155 psig) retain the simmer margin, which provided satisfactory leak tightness performance, and remain unchanged. The safety valve setpoint also remains the same at 1260 psig and retains differential pressure between the steam dome operating pressure and nominal safety valve setpoints. The setpoints on these valves satisfy the ASME code specifications for safety valves since the lowest set valve opens below the 1250 psig nuclear system design pressure, and the highest set valve opens below 105% of the vessel design pressure (1313 psig). [Note that this analysis remains valid for 4016 MWt, as the reactor dome pressure does not change.]

Since the same valves are being used, a new valve capacity analysis is not performed for 4016 MWt. The original valve capacity determination was made by analyzing the pressure rise accompanying the main steam flow stoppage resulting from a 3-sec main steam line isolation valve closure initiated from operating conditions with a vessel dome pressure of 1020 psig. This original analysis indicated that there was approximately 78 psi margin below the ASME overpressure limit of 1375 psig.

The case of MSIV closure with neutron flux scram (MSIVF) was analyzed using TRACG methodology with statistical pressure adder and is also analyzed during for every cycle-specific reload and was re-evaluated at 4030 MWt, which bounds 4016 MWt, to assure that the ASME code allowable value for peak vessel pressure is not violated. The analysis conservatively assumes that the position scram fails and the event terminates on a high neutron flux scram signal. The closure of all MSIVs causes a rapid pressure increase in the reactor vessel. The pressure increase is mitigated by the actuation of the relief valves operation and by the safety valves. The MSIVF event is analyzed with two relief valves out of service, which are assumed to be an SRV with the lowest pressure setpoint. This assumption is conservative since this minimizes the initial pressure relief capacity and results in the highest peak pressure value for the overpressure analyses.

The MSIVF event was analyzed at 110% of ratedincreased core flow and normal feedwater temperatureMELLLA+ conditions, which is are the bounding conditions for vessel overpressure calculations. The peak bottom pressure for 4030 MWt with MELLLA+ conditions is 1352 psig which is below the ASME Code limit of 1375 psig. With the SRV/SV configuration at a +3% tolerance setting and two relief valves out of service, the resulting peak vessel pressure is 1352 1354 psig, corresponding to a margin of 23 21 psi to the ASME upset code limit. The peak dome pressure reached is 1324 1327 psig, with 1 13 psi margin to the Tech Spec safety limit. These pressure margins are adequate to account for any differences in the unit- and cycle-specific reload transient analyses. This analysis was performed assuming one relief valve out-of-serviceSee Reference 2 for additional information.

System malfunctions which pose threats to the radioactive material containment barriers are presented in Section 14.0, "Plant Safety Analysis."

Evaluations of the automatic depressurization capability of the nuclear system pressure relief system are presented in Section

6.0, "Core Standby Cooling Systems," and subsection 7.4, "Core Standby Cooling Systems Control and Instrumentation."

As stated in the operational analyses presented in Appendix G, the ADS valves may be used to provide an alternative shutdown cooling path in the event of a failure of the normal suction flow path. For further discussions of this ADS feature see Appendix G.

The relief valve discharge piping is designed as described in Appendix A.

4.4.7 Power Generation Evaluation

The relief valves are designed to relieve energy from the nuclear system rapidly enough to prevent operation of spring safety valves (SSVs) during pressure transients which are reasonably expected during the lifetime of the plant. A variety of turbine system malfunctions can initiate a turbine stop valve closure, normally called a turbine trip. This event represents the fastest possible steam flow shutoff (about 0.1 sec) and, therefore, represents the potential for a severe pressure transient.

An inadvertent closure of all main steam isolation valves (MSIVs) also results in a steam flow shutoff, although more slowly. However, since the MSIVs are upstream of the steam bypass valves which help alleviate the pressure increase resulting from the closure of the turbine stop valves, the MSIV closure event produces a more severe pressure increase than the turbine trip with bypass event.

The MSIV closure event was analyzed for 4016 MWt conditions with the relief valve setpoints given in paragraph 4.4.6 and using nominal assumptions and assuming all relief valves are operational. The analysis shows (Fig. 14.5.1B) that relief valves open fully to limit the pressure rise, then sequentially close as the stored energy is dissipated. The peak pressure in the steam line at any SSV is below the lowest nominal setpoint (1,260 psig) for the SSVs with a margin of approximately 92 psi (Reference 4.4.1). A sensitivity study assuming one relief valve out-of-service was also performed with a resulting reduction in available margin of less than 5 psi. Although an operational consideration only (not safety related), the SSV pressure margin is evaluated each reload to track available margin.

4.4.8 Inspection and Testing

The safety and relief valves are tested to detect defects and prove operability.

Test and vent connections at the ADS pneumatic supply check valves provide for leak rate testing prior to startup and also periodic local leak rate testing. Test and vent connections are provided for periodic testing of the containment isolation valves in the safety grade pneumatic supply to the ADS valves.

It is not feasible to test the safety and relief valve set points while the valves are in place or during normal plant operation. The valves are mounted on flanges so that they may be removed for maintenance or bench checks and reinstalled during normal plant shutdowns. The external surface and flange mating surface of all relief valves and safety valves are 100 percent visually inspected when the valves are removed for maintenance or bench checks.

Surveillance testing of the eleven relief and three safety valves will be as follows:

1. Each relief and safety valve will be bench tested every second refueling outage. Approximately one-half the valves will be tested at each refueling outage.

2. The pneumatic actuator of each ADS valve is stroked to verify that the second stage pilot disc rod is mechanically displaced when the actuator strokes. Second stage pilot rod movement is determined by the measurement of actuator rod travel. The total amount of movement of the second stage pilot rod from the valve closed position to the open position shall meet criteria established by the S/RV supplier.

In-service inspection is conducted in accordance with the In-service Inspection Program.

4.4 NUCLEAR SYSTEM PRESSURE RELIEF SYSTEM

REFERENCES

1. "Peach Bottom 3 Cycle 21 Spring Safety Valve Lift Margin Evaluation," GEH Report 002N5709-R0, April 15, 2015.

2. Peach Bottom Units 2 & 3 Two Safety Relief Valves Out-of-Service Evaluation," GEH Report 004N6240-R1, March 2018 (G-080-VC-468).

TABLE 4.4.1

NUCLEAR SYSTEM SAFETY AND RELIEF VALVES

	Number of Valves	Set Pressure, (psig)	Certified Capacity at Reference Pressure* (each)
Relief Valves	4	1,135	800,000 lbm/hr
	4	1,145	800,000 lbm/hr
	3	1,155	800,000 lbm/hr
Total	11(5)		
Safety Valves	3 (Unit 2) 2 (Unit 3)	1,260	925,700 lbm/hr

* Reference pressure = 1080 psig (relief valves) and 1230 psig (safety valves). Valve flow capacities at each specific set pressure are based on the certified capacity values, considering a 3% accuracy factor. See M-1-U-116 for specific predicted valve capacities in relief and safety functions.

4.5 MAIN STEAM LINE FLOW RESTRICTORS

4.5.1 Safety Objective

The safety objective of the main steam line flow restrictors is to protect the fuel barrier by limiting the loss of coolant from the reactor vessel before main steam line isolation valve closure in case of a main steam line rupture outside the primary containment.

4.5.2 Safety Design Basis

1. The main steam line flow restrictor is designed to limit the loss of coolant from the reactor vessel, following a steam line rupture outside of the primary containment, to the extent that the reactor vessel water level does not fall below the top of the core within the time required to close the main steam line isolation valves.

2. The main steam line flow restrictor is designed to withstand the maximum pressure difference expected across the restrictor following complete severance of a main steam line.

4.5.3 Description

A main steam line flow restrictor (shown in Figure 4.5.1) is provided for each of the four main steam lines. The restrictor is a complete assembly welded into the main steam line between the reactor vessel and the first main steam line isolation valve, and downstream of the main steam line safety and relief valves. The restrictor limits the coolant blowdown rate from the reactor vessel, in the event of a main steam line break outside of the primary containment. The restrictor assembly consists of a venturi-type nozzle insert welded into a carbon steel pipe. The venturi-type nozzle insert is constructed utilizing all austenitic stainless steel and is held in place with a full circumferential fillet weld. The restrictor assembly is self-draining (low point pockets are internally drained to steam line).

The flow restrictor is designed and fabricated in accordance with ANSI B31.1. It has no moving parts, and the mechanical structure of the restrictor is capable of withstanding the velocities and forces under main steam line break conditions, where the maximum differential pressure is conservatively assumed to be equal to the maximum allowed pressure for the reactor vessel, in accordance with the ASME Boiler & Pressure Vessel Code, Section III (1965).

The ratio of the venturi throat diameter to steam line diameter is approximately 0.6. This results in a small unrecoverable pressure difference at rated flow. This design limits the steam flow in a severed line to about 200 157 percent of its rated flow, yet it results in a negligible increase in steam moisture content during normal operation. The restrictor is also used in the measurement of steam flow to initiate closure of the MSIV's in the event the steam flow exceeds preselected operational limits and to provide steam flow information to the feedwater control system.

4.5.4 Safety Evaluation

In the event of a main steam line break outside the primary containment, steam flow rate is restricted in the venturi throat by a two-phase mechanism similar to the critical flow phenomena in gas dynamics. This limits the steam flow rate, thereby reducing the reactor vessel coolant blowdown and the fuel clad temperature increase subsequent to the blowdown. The probability of fuel failure and its consequences are therefore decreased.

Analysis of the steam line rupture accident (Section 14.0, "Plant Safety Analysis") shows that the core remains covered with water and that the amount of radioactive materials released to the environs through the main steam line break does not exceed the guideline values of published regulations.

Pressure surges caused by a two-phase mixture impinging on the flow restrictor result in stresses which do not exceed code allowable limits. There is adequate margin in the code for withstanding the pressure load due to impact pressure from the possible oncoming two-phase mixture predicted during main steam line break accident conditions.

Tests were conducted on a scale model to determine final design and performance characteristics of the flow restrictor, including maximum flow rate of the restrictor corresponding to the accident conditions, irreversible losses under normal plant operating conditions, and discharge moisture level. The tests showed that the flow restrictor operation at critical throat velocities is stable and predictable. Unrecovered differential pressure across the scale model restrictor is consistently about 10 percent of the total nozzle pressure differentials, and the restrictor performance is in agreement with existing ASME correlations. Full size restrictors have a slightly different hydraulic shape and a differential pressure loss of approximately 15 percent.

4.5.5 Inspection and Testing

Because the flow restrictor forms a permanent part of the main steam line piping and has no moving components, the restrictor is not routinely tested or inspected. Only very slow erosion will occur with time, and such a slight enlargement will not have safety significance.

4.6 MAIN STEAM LINE ISOLATION VALVES

4.6.1 Safety Objective

The safety objective of the two MSIV's, one on each side of the primary containment barrier in each of the main steam lines, is to close automatically to:

1. Prevent damage to the fuel barrier by limiting the loss of reactor coolant in case of a major leak from the steam piping outside the primary containment.

2. Limit release of radioactive materials by closing the nuclear system process barrier in case of gross release of radioactive materials from the reactor fuel to the reactor cooling water and steam.

3. Limit release of radioactive materials by closing the primary containment barrier in case of a major leak from the nuclear system inside the primary containment.

4.6.2 Safety Design Basis

The MSIV's, individually or collectively, will:

1. Close within the time established by design basis accident analysis to limit the release of reactor coolant or radioactive materials.

2. Close at a speed slow enough so that simultaneous (inadvertent) closure of all steam lines will not induce a more severe transient on the nuclear system than closure of the turbine stop valves while the turbine bypass valves remain closed.

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

4. Use valve closure motive force energy sources which are independent from the energy sources used to close the redundant valve in each steam line, to allow independent closure of redundant valves.

5. Use local stored energy (compressed air and springs) to close at least one isolation valve in each stem line without relying on continuity of any variety of

electrical power for the motive force to achieve closure.

6. Close during or after design basis seismic loadings to assure isolation.

7. Be testable during normal operating conditions to demonstrate that the valves will function.

4.6.3 Description

Two isolation valves are welded in a horizontal run of each of the four main steam lines, with one valve as close as possible to the primary containment barrier inside, and the other just outside the barrier. The valves, when closed, form part of the nuclear system process barrier for openings outside the primary containment, and part of the primary containment barrier for nuclear system breaks inside the containment.

The description and testing of the controls for the MSIV's are included in subsection 7.3, "Primary Containment and Reactor Vessel Isolation Control System."

A drawing of an MSIV is shown in Figure 4.6.1.

Each valve is a "Y"-shaped, 26-in globe valve and is installed in a matching 26-in pipe. The MSIV flow rate is 4.119 x 10^6 lb/hr at 1050 psia. The valve is designed for saturated steam at 1,250 psig and 575°F with a moisture content of approximately 0.23 percent, an oxygen content of 30 ppm, and a hydrogen content of 4 ppm. The main disc, or poppet, is attached to the lower end of the stem and moves in guides at a 45° angle from the inlet pipe. Normal steam flow tends to close the valve and higher inlet pressure tends to hold the valve closed. The bottom end of the valve stem closes a small pressure-balancing hole in the poppet; when open, it acts as a pilot valve to relieve differential pressure forces on the poppet. The valve stem travel is approximately 10 in; the main disc travels 9 in; and the last 1.0 in closes the pilot hole. A helical spring between the stem and the poppet keeps the pilot hole open when the poppet is off its seat, but failure of the spring will not prevent closure of the valve. An air cylinder with spring assist is part of the valve assembly and provides the motive capability for the valve. The cylinder can open the poppet with at least 200-psi differential pressure across the isolation valve in either direction. The poppet can open from higher differential pressures when pressure balancing occurs from opening the pilot.

A description of closing devices including accumulators for the isolation valves is discussed in paragraph 7.3.4.6, "Isolation Valve Closing Devices and Circuits."

The diameter of the poppet seat is approximately the same size as the inside diameter of the pipe, and the 45° angle permits streamlining of the inlet and outlet passage to minimize pressure drop during normal steam flow, and to avoid blockage by debris. The pressure drop at rated flow is less than 6 psi. [For Unit 2 "A" and "C" outboard and Unit 3 "A" and "D" outboard only, the valve backseats in the fully open position to prevent leakage through the stem packing.] The bonnet has provisions for seal welding in case leaks develop after the valve has extensive service.

The upper end of the stem is attached to a combination air cylinder and hydraulic dashpot that are used for opening and closing the valve and for speed control, respectively. Speed is adjusted by a valve in the hydraulic return line alongside the dashpot; the valve closing time is adjustable between 3 and 10 sec.

The cylinder is supported on large shafts threaded and pinned in to the valve bonnet. The shafts are also used as guides for the helical springs used to assist in closing the valve. The springs exert downward force on the spring seat member which is attached to the stem. Spring guides prevent scoring in normal operation, and prevent binding if a spring breaks. The spring seat member is also closely guided on the support shafts and rigidly attached to the stem to control any eccentric force in case of a broken spring.

A pair of reactor protection system switches, a switch to indicate when valve position is less than full open and a switch to indicate when valve position is less than full closed, are actuated by the motion of the spring seat member. The pair of reactor protection system switches have an analytical setpoint of $\leq$ 15 percent closed MSIV position to support an allowable setpoint of $\leq$ 10 percent closed MSIV position. The reactor protection system switches initiate a reactor scram if several valves close (subsection 7.2, "Reactor Protection System").

The valve is operated by pneumatic pressure and by the action of compressed springs. The instrument air system (subsection 10.17) supplies air to the main steam isolation valves external to the containment. The inner isolation valves are supplied from the instrument nitrogen system (subsection 10.17). The control unit is attached to the pneumatic supply and contains the pneumatic, ac and dc solenoid valves for opening, closing, and slow speed exercising of the main valve. The control power for each valve is supplied from two separate buses at 120 V AC, 60 Hz, or 125 V DC. Remote manual switches in the control room enable the operator to close each valve at fast speed (adjustable between 3 to 10 sec) or at slow speed (adjustable between 45 to 60 sec) for exercising or testing.

In the event that the main steam line should rupture downstream from the valve, the steam flow quickly increases to 157 percent of rated flow, being limited from further increase by the venturi flow restrictor upstream of the valves.

During approximately the first 75 percent of closing, the valve has little effect in reducing flow because the flow is choked by the venturi restrictor upstream from the valves. After the valve is more than about 75 percent closed, flow is reduced as a function of the valve area versus travel characteristic.

The design objective for the valve is a minimum of 40 years of service at the specified operating conditions with consideration of MSIV operating frequency.

In addition to the minimum wall thickness, which was calculated by interpolation between ANSI B16.5 pressure classes, a corrosion allowance of 0.088 inches minimum is included to provide for 40 years of service.

MSIV electrical components important to its operation are qualified for the inboard and outboard MSIV environmental conditions as documented in the PBAPS Environmental Qualification (EQ) Program.

The main steam line valve installations are designed to seismic Class I criteria to resist sufficiently the response motion from the maximum credible earthquake at the installed location within the supporting building. The valve assembly is manufactured to withstand the design basis seismic forces applied at the mass center, assuming the cylinder/spring operator is cantilevered from the valve body and the valve is located in a horizontal run of pipe. The stresses caused by horizontal and vertical seismic forces are considered to act simultaneously and are added directly. The seismic coefficients are specified as 0.53g horizontal and 0.04g vertical. The stresses in the actuator supports caused by seismic loads are combined with the stresses caused by other live and dead loads, including the operating loads. The allowable stress for this combination of loads is based on the allowable stress as set forth in the applicable codes. The parts of the MSIV's which constitute a process fluid pressure boundary are designed, fabricated, inspected, and tested in accordance with Appendix A.

The control valves and other equipment provided in the valve assembly are designed, manufactured, and shop tested in accordance with the revision of the following codes and standards in effect at the time of manufacturing, where applicable:

1. American National Standards Institute (ANSI) B31.1 and B16.5

2. American Society for Testing Materials (ASTM)

3. American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Sections 1, III and VIII

4. Institute of Electronic and Electrical Engineers (IEEE)

5. Pipe Fabrication Institute

6. National Electrical Manufacturers' Association.

4.6.4 Safety Evaluation

The safety objectives of the MSIV's are to limit the release of radioactive material by closing the nuclear system process barrier and the primary containment barrier, and to limit the loss of reactor cooling water in case of a major steam leak outside the primary containment.

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

The analysis of a complete, sudden steam line break outside the primary containment is described in Section 14.0, "Plant Safety Analysis." It shows that the fuel barrier is protected against loss of cooling if MSIV closure takes as long as 10.5 sec (which includes up to 0.5 sec for the instrumentation to initiate valve closure after the break). The calculated radiological effects of the radioactive material assumed released with the steam are shown to be well within the guideline values for such an accident. Thus, safety design basis 1 is shown to be satisfied with considerable margin.

The shortest closing time (approximately 3 sec) of the MSIV's is also shown to be satisfactory in Section 14.0, "Plant Safety Analysis." The switches on the valves initiate reactor scram when several valves are more than 10 percent closed. The pressure rise in the system from stored and decay heat may cause the nuclear system relief valves to open briefly, but the rise in fuel cladding temperature will be insignificant. The transient is less than that from sudden closure of the turbine stop valves (in approximately 0.1 sec) coincident with postulated failure of the turbine bypass valves to open. No fuel damage results. Thus, safety design basis 2 is shown to be satisfied with considerable margin.

The ability of this 45°, Y-type, globe valve design to close in a few seconds after a steam line break, under conditions of high-pressure differentials and fluid flows, with fluid mixtures ranging from mostly steam to mostly water, has been demonstrated in a series of tests in dynamic test facilities. Dynamic tests with a 1-in valve showed that the analytical method was valid. A full-size, 20-in valve was tested in a range of steam/water blowdown conditions simulating postulated accident conditions(1).

Specified hydrostatic, leakage, and stroking tests, as a minimum were performed by the valve manufacturer in shop tests. Each valve was tested at normal operating pressure of 1,000 psig, and no flow, to verify capability to close between 3 and 10 sec. The valve was stroked several times and the closing time recorded. The valve was closed by the pneumatic cylinder and springs, and also could be closed by the springs only against external atmospheric backpressure. The closing time was usually slightly greater when closed by springs only but well within required closing times.

Adjustability of the closing time of each valve between 3 and 10 sec was tested at normal operating pressure and no flow. The valve was stroked several times at the fastest setting, intermediate setting, and the slowest setting. Closing times were recorded.

Valves were shop tested at atmospheric pressure and room temperature to verify closing time and adjustability of closing time.

Leakage with the valve seated and back seated was measured. Seat leakage was measured by pressurizing the upstream side of the valve to 1,250 psig. The specified maximum seat leakage, using cold water at design pressure, is 2 cc/hr/in of seat diameter. There must be no visible leakage from either set of stem packing at design pressure. The valve stem was operated a minimum of three times from the closed to open position, and the packing leakage was still zero by visual examination. In addition, an air seal leakage test was conducted using 50 psig air pressure.

Each valve was hydrostatically tested at a pressure of 2,380 psig calculated by interpolation between ANSI B16.5 pressure class test requirements.

During valve fabrication, extensive non-destructive tests and examinations were made, including radiographic, liquid penetrant, and/or magnetic particle examinations of casting, forgings, welds, hardfacings, and bolts.

Shop tests prove that the valve sealing surfaces can withstand impact loadings from closing even in the event that the hydraulic buffer is inoperative. The spring guides, the guiding of the spring seat member on the support shafts, and rigid attachment of the seat member to the stem assure that the valve closes when only spring-driven against external atmospheric backpressure. Three sets of springs, or two opposite sets, will close the valve against external atmospheric backpressure. The pneumatic cylinder, alone or with the remaining spring, will close the valve against external atmospheric backpressure even if a spring set jams, since the springs are not attached to the spring seat member. Binding of the valve poppet in the guides is prevented by making the poppet in the form of a cylinder longer than its diameter, and by applying the stem force near the bottom of the poppet. Clearance is provided between the poppet and its guides so that some cocking of the poppet, or warpage of the seat, can be tolerated while still achieving a seal.

After the valves were installed in the nuclear system, each valve was tested in accordance with the pre-operational and startup test procedures.

Redundancy is provided by two isolation valves in each steam line so that either can perform the isolation function, and either can be tested for leakage after closing the other. The inside valve and the outside valve and their control systems are separated physically. Considering the redundancy, the mechanical strength, the closing forces, and the leakage tests discussed above, the MSIV's satisfy the safety design basis to limit the release of reactor coolant or radioactive materials, within the margins evaluated in Section 14.0, "Plant Safety Analysis."

The isolation valves and their installation are designed to seismic Class I criteria.

The design of the isolation valve has been analyzed for earthquake loading. These loads are small compared with the pressure and operating loads the valve components are designed to withstand. The cantilevered support of the pneumatic cylinder, hydraulic cylinder, springs, and controls, is the key area. The increase in loading at the joints between the support shafts and the valve bonnet caused by the specified earthquake loading is negligible. Therefore, the seismic loading requirement of design basis 6 is met.

Electrical equipment, associated with the isolation valves, that operates in an accident environment is limited to the wiring, solenoid valve, and position switches on the isolation valves. As described in subsection 7.19, "Class 1E Equipment Environmental Qualification," this equipment is qualified to operate in the accident environment.

Operation of the valves in the normal operating conditions is ensured by the requirements of the purchase specifications, review and approval of equipment design and vendor drawings, extensive control of materials, fabrication procedures, fabrication tests, non-destructive examinations, shop tests, pre-operational and startup tests of the installed valves, and prescribed inspections and tests during plant life.

4.6.5 Inspection, Testing, and Operation with MSIV Closed

The MSIV's are may be tested during plant operation, and are tested and inspected during refueling outages.

The MSIV's are routinely may be tested and exercised individually to the allowable setpoint of $\leq$ 10 percent closed position, without reducing reactor power, since the valves still pass rated steam flow when $\leq$ 10 percent closed.

The MSIV's may be tested and exercised individually to the fully closed position by reducing reactor power to 75 percent of referenced power (3514 MWt). Continued operation with MSIV's closed on one Main Steam Line is permitted provided that reactor thermal power is maintained at or below 75 percent of 3514 MWt (Ref. 4).

The leak rate through the steam valve seats (pilot and poppet seats) is measured accurately during shutdown by pressurizing between the closed valves.

4.6 MAIN STEAM ISOLATION VALVES

REFERENCE

1. Van Zylstra, E.; Sutherland, W.; and Rockwell, D., "Design and Performance of GE BWR Main Steam Isolation Valves," General Electric Company, Atomic Power Equipment Department, APED-5750 March 1969.

2. NEDC-33566P, “Safety Analysis Report for Exelon Peach Bottom Atomic Power Station, Units 2 and 3, Constant Pressure Power Uprate,” Revision 0, September 2012.

3. UFSAR Change Request 2016-027 (ECR 16-00346), "MSIV Poppet Skirt Modification," November 2016Deleted.

4. PBAPS Units 2 & 3 TPO Task Report T0900, "Transient Analysis," PEAM-MPLUS-18, June 17, 2014.

4.7 REACTOR CORE ISOLATION COOLING SYSTEM

4.7.1 Safety Objective

The safety objective of the RCICS is to provide makeup water to the reactor vessel during shutdown and reactor isolation in order to prevent the release of radioactive materials to the environs as a result of inadequate core cooling.

4.7.2 Safety Design Basis

1. The system operates automatically to maintain sufficient coolant in the reactor vessel so that the integrity of the radioactive material barrier is not compromised.

2. The functional components of the RCICS satisfy seismic Class I criteria.

4.7.3 Description

The RCICS consists of a steam-driven turbine-pump unit and associated valves and piping capable of delivering makeup water to the reactor vessel. A summary of the design requirements of the turbine-pump unit is shown on Table 4.7.1. The RCIC system is shown in Drawings M-359 (Sheets 1 and 2) and M-360 (Sheets 1 through 4) and the logic diagrams are shown in Drawing M-1-CC-38 (Sheets 1 through 12).

The steam supply to the turbine comes from the "C" main steam line between the reactor and inboard MSIV and exhausts to the suppression pool. The pump can take suction from the condensate storage tank or from the suppression pool. The pump discharges either to the feedwater line or to a full flow return test line which will discharge to the condensate storage tank (CST) or suppression pool. This full flow test line is shared with HPCI. A minimum flow line to the suppression pool is provided. The makeup water is delivered into the reactor vessel through a connection to the "B" feedwater line and is distributed within the reactor vessel through the feedwater spargers. Cooling water for the RCICS turbine lube oil cooler and barometric condenser is supplied from the discharge of the pump.

Following any reactor shutdown, steam generation continues due to decay heat. Initially, the rate of steam generation can be as much as 6 percent of rated flow. The steam normally flows to the main condenser through the turbine bypass or, if the condenser is isolated, through the relief valves to the suppression pool. The fluid removed from the reactor vessel either can be furnished entirely by the feedwater system or can be partially furnished by the CRDS, which is supplied by the CRD pumps. If makeup water is required to supplement these sources of water, the RCICS turbine-pump unit either starts automatically upon receipt of a reactor vessel low-low water level signal or is started by the operator from the control room by remote manual controls. The same low-low level signal also energizes the HPCIS (Section 6.0, "Core Standby Cooling Systems"). The RCICS delivers its design flow within 50 sec after actuation.

For events other than pipe breaks, the RCICS has a makeup capacity sufficient to prevent the reactor vessel water level from decreasing to the level where the core is uncovered without the use of the CSCS's (Section 14.0, "Plant Safety Analysis"). Two sources of water (CST/suppression pool) are available. Initially the condensate storage tank is used. A low water level alarm from the condensate storage tank provided in the control room is energized when the level in the CST falls to a predetermined point. This is to ensure the NPSH requirement of the RCIC pump is met.

Should the condensate storage tank be unavailable, pump suction is automatically transferred from the condensate storage tank to the suppression pool (see NRC SER dated 4/6/83, "Auto Switchover of RCIC"). During periods of time when the RCICS is lined up to take suction from the suppression pool, the system is verified full by use of a hose connection from the condensate storage tank. The turbine-pump assembly is located below the level of the condensate storage tank, and below the minimum water level in the suppression pool, to assure a positive suction head to the pump. Pump NPSH requirements are met by providing adequate suction head and adequate suction line size.

All components necessary for initiating operation of the RCICS are completely independent of auxiliary ac power, plant service air, and external cooling water systems, requiring only dc power from the station battery. The power source for the turbine-pump unit is the steam generated in the reactor pressure vessel by the decay heat in the core. The steam is piped directly to the turbine, and the turbine exhaust is piped to the suppression pool. Vacuum breakers are connected between the RCICS turbine exhaust line upstream of the penetration through the torus shell and the torus air space. The vacuum breakers are arranged to prevent a negative pressure in the section of turbine exhaust piping between the torus and the upstream stop check valve. To provide single failure protection against a vacuum breaker valve failing in the open position there are four vacuum breakers installed in a one-out-of-two-twice arrangement.

If for any reason the reactor vessel is isolated from the main condenser, pressure in the reactor vessel increases but is limited by automatic or remote-manual actuation of the main steam relief valves (MSRV). MSRV discharge is piped to the suppression pool. Throughout the period of RCICS operation, the exhaust from the RCICS turbine and the relief valve discharge is being condensed in the suppression pool resulting in a temperature rise in the pool. During this period, RHR heat exchangers are used to maintain pool water temperature within acceptable limits.

The RCICS turbine-pump unit is located in a shielded area to maintain low dose rates and assure that personnel access to adjacent areas is not restricted during RCICS operation. The turbine controls provide for automatic shutdown of the RCICS turbine upon receipt of the following trip signals:

1. Reactor vessel high water level-- indicating that core cooling requirements are satisfied.

2. Turbine over-speed-- to prevent damage to the turbine and turbine casing.

3. Pump low suction pressure-- to prevent damage to the turbine-pump unit due to cavitation or flashing at suction.Deleted.

4. Turbine high exhaust pressure-- indicating turbine or control malfunction.

5. Auto Isolation Signal

A probabilistic missile evaluation has been performed on the RCICS pump turbine and is described in subsection 11.2.

The steam supply valve secures steam flow on high water level. The steam supply valve automatically reopens on a subsequent low reactor water level signal to automatically restart the RCICS turbine (see NRC SER dated 3/16/83, "RCIC Automatic Restart"). The turbine trip valve closes on the remaining trip parameters. Turbine trip valve closure requires manual reset of the system. Turbine overspeed trip also requires a local reset of the trip valve.

Since the steam supply line to the RCICS turbine is a primary containment boundary, certain signals automatically isolate this line, causing shutdown of the RCICS turbine. Automatic shutdown of the steam supply is described in subsection 7.3, "Primary Containment and Reactor Vessel Isolation Control System."

The turbine control system is positioned initially by the ramp generator and signal converter and subsequently by the demand signal from a flow controller, and satisfies a twofold purpose:

1. To limit the turbine pump speed to its maximum normal operating value, and

2. To position the turbine governor valve as required to maintain constant pump discharge flow over the pressure range of system operation.

To ensure rapid delivery of water to the RPV and to minimize water hammer effects, the RCIC System discharge piping is kept full of water. The RCIC System is normally aligned to the CST. The height of water in the CST is sufficient to maintain the piping full of water up to the first isolation valve. The relative height of the feedwater line connection for RCIC is such that the water in the feedwater lines keeps the remaining portion of the RCIC discharge line full of water. Therefore, RCIC does not require a "keep fill" system to meet the requirements of Generic Letter 2008-01 and Technical Specifications (TS) Amendment Nos. 297/300.

The RCICS piping is designed as described in Appendix A.

Valve position indication and instrumentation alarms are displayed in the control room.

4.7.4 Safety Evaluation

To provide a high degree of assurance that the RCICS operates when necessary and in time to provide adequate core cooling, the power supply for the system is taken from energy sources of high reliability which are immediately available. The capability of testing during plant operation gives added assurance. Evaluation of instrumentation reliability for the RCICS shows that no failure of a single initiating sensor either prevents the operation of or falsely starts the system. The design of the RCICS is in accordance with Appendix A and ensures that the safety design bases are satisfied.

4.7.5 Inspection and Testing

A design flow functional test of the RCICS is performed during plant operation by taking suction from the condensate storage tank or the suppression pool and discharging through the full flow test return line back to the condensate storage tank or to the suppression pool. The discharge valve to the feedwater line remains closed during the test and reactor operation is undisturbed. Control system design provides automatic return from test to operating mode if system initiation is required during testing. Inspection and maintenance of the system are carried out generally in accordance with the manufacturer's instructions and good operating practice.

TABLE 4.7.1

REACTOR CORE ISOLATION COOLING SYSTEM TURBINE

PUMP DESIGN DATA

PUMP

Number Required	1
Capacity	100%
Design Temperature	40°F to 140°F
Design Pressure	1,500 psig
NPSH	20 ft (minimum)

Developed Head

2,7902,835 ft @ 1,1501,185 psia psig Reactor Pressure
525 ft @ 165 psia Reactor Pressure

Flow Rate

Injection Flow	600 gpm
Cooling Water Flow	16 gpm
Total Pump Discharge	616 gpm

TURBINE

Number Required	1
Capacity	100%
Steam Inlet Pressure Range (psia)	150 to 1,150 (saturated)
Steam Exhaust Pressure (psia)	25 (nominal)

System Design

Rx Pressure	Turbine Inlet	Head
165 psia (150 psig)	150 psia (135 psig)	525 ft
1165 1185 psia (1150 1170 psig)	1150 psia (1135 psig)	2870 2,790 2,835 ft

4.8 RESIDUAL HEAT REMOVAL SYSTEM

4.8.1 Safety Objective

The safety objective of the RHRS is to restore and maintain the coolant inventory in the reactor vessel so that the core is adequately cooled after a LOCA. The RHRS also provides cooling for the containment so that condensation of the steam resulting from the blowdown due to the design basis LOCA is ensured.

Post-LOCA RHR heat load increased under Extended Power Uprate (EPU3951 MWt) conditions due, in part, to an increase in reactor decay heat. (This analysis was performed at 4030 MWt, which remains valid for the MUR rated thermal power of 4016 MWt.). This would have increased suppression pool temperatures, decreased NPSH margin for the ECCS pumps, and increased the reliance on containment accident pressure (CAP). The need for CAP credit for the emergency core cooling system pumps, however, has been eliminated as a result of the installation of an RHR system heat exchanger cross-tie (UFSAR Section 4.8.5) and the maintenance of an increased K-factor in the RHR heat exchangers (UFSAR Section 6.4.5). The RHR system cross-tie enables the operator to align a second RHR heat exchanger for post-OOCA containment heat removal. A second HPSW pump supplies cooling water flow to the second RHR heat exchanger on the LOCA unit.

4.8.2 Safety Design Basis

1. The RHRS acts automatically, in combination with other CSCS's, to restore and maintain the coolant inventory in the reactor vessel such that the core is adequately cooled to preclude fuel clad temperature in excess of 2,200°F following a design basis LOCA.

2. The RHRS, in conjunction with other CSCS's, has such diversity and redundancy that only a highly improbable combination of events could result in the inability to provide adequate core cooling.

3. The source of water for restoration of reactor vessel coolant inventory is located within the primary containment in such a manner that a closed cooling water path is established.

4. To provide a high degree of assurance that the RHRS operates satisfactorily during a LOCA, each active

component is capable of being tested during operation of the nuclear system.

5. The functional components of the RHRS are designed to seismic Class I criteria.

4.8.3 Power Generation Objective

The power generation objective of the RHRS is to cool the nuclear system following reactor shutdown.

4.8.4 Power Generation Design Basis

1. The RHRS removes residual heat from the nuclear system so that refueling and nuclear system servicing can be performed.

2. The RHRS is capable of supplementing the fuel pool cooling system capacity when necessary to provide additional cooling capacity.

3. An additional source of water for post-accident containment flooding is provided by a cross-tie between the high-pressure service water system and the RHRS.

4.8.5 Summary Description

The RHRS is designed for three modes of operation to satisfy all objectives and bases. Each mode of operation is defined as a subsystem of the RHRS and is discussed separately. Each subsystem contributes toward satisfying all objectives and design bases of the RHRS.

The major equipment of the RHRS consists of four heat exchangers, four main system pumps, and four high-pressure service water pumps for each unit (subsection 10.7, "High-Pressure Service Water System"). The equipment is connected by associated valves and piping, and the controls and instrumentation are provided for proper system operation. A schematic diagram of the RHRS is shown in Drawing M-1-DD-9. A description of the controls and instrumentation is presented in subsection 7.4, "Core Standby Cooling Systems Control and Instrumentation." A description of how operation of the equipment in the RHRS, in conjunction with other CSCS's, protects the core in case of a LOCA, is presented in Section 6.0, "Core Standby Cooling Systems."

A summary of the design requirements of the main system pumps and the heat exchangers is presented in Table 4.8.1.

Large capacity passive pump suction strainers have been installed on each RHR suction line in the suppression pool, via plant modification, in response to NRC I.E. Bulletin 96-03 "Potential Plugging of Emergency Core Cooling Suction Strainers by Debris in Boiling Water Reactors."

Connections are provided on the shutdown cooling piping (Drawing M-361, Sheets 1 through 4) so that the RHRS heat exchangers may be used to assist fuel pool cooling when required (subsection 10.5, "Fuel Pool Cooling and Cleanup System").

One loop, consisting of two heat exchangers, two main system pumps in parallel, and associated piping, is located in one area of the reactor building. The other heat exchangers, pumps, and piping, forming a second loop, are located in another area of the reactor building to minimize the possibility of a single physical event causing the loss of the entire system. The two loops of the RHRS are cross-connected by a single header, making it possible to supply either loop from the pumps in the other loop. The flexibility provided by this arrangement satisfies safety design basis 2. Each loop is also provided with a normally closed cross-tie line connecting the RHR pump discharge headers upstream of the RHR heat exchangers. With a single RHR pump available, additional containment cooling capacity can be achieved by opening the cross-tie valve and aligning RHR flow to both RHR heat exchangers (a second HPSW pump is also required to provide cooling water to the second RHR heat exchanger). Flow control valves upstream of each of the RHR heat exchangers, in conjunction with flow elements with Main Control Room indication at the outlet of each heat exchanger, allow operators to balance flow for containment cooling and prevent RHR pump runout.

RHRS equipment is designed in accordance with seismic Class I criteria to resist sufficiently the response motion at the installed location within the supporting building from the maximum credible earthquake. The main system pumps are assumed to be filled with water for the seismic analysis.

The RHR and core spray piping have an automatic fill system to ensure the lines are filled with water. Two water makeup sources are provided: (1) condensate transfer system during plant operation and shutdown, and (2) condensate pump discharge.

The system piping, heat exchangers, and main system pumps are designed as described in Appendix A. The pumps are performance tested in accordance with the Standards of the Hydraulic Institute.

4.8.6 Description

4.8.6.1 Shutdown Cooling

The shutdown cooling subsystem is an integral part of the RHRS and is placed in operation during a normal shutdown and cooldown. The initial phase of nuclear system cooldown is accomplished by dumping steam from the reactor vessel to the main condenser, with the main condenser acting as the heat sink. When nuclear system temperature has decreased to a point where the steam supply pressure is not sufficient to maintain the turbine shaft gland seals, vacuum in the main condenser cannot be maintained and the RHRS is placed in the shutdown cooling mode of operation. The shutdown cooling subsystem (with two shutdown cooling subsystems in service) is capable of completing cooldown to 125°F in approximately 35 hours, and of maintaining the nuclear system at 125°F so that the reactor can be refueled and serviced.

Reactor coolant is pumped by the RHRS main system pumps from one of the recirculation loops through the RHRS heat exchangers, where cooling takes place by transferring heat to the service water. Radiation monitors are located in the high-pressure service water upstream and downstream of the RHR heat exchangers. These monitors alarm on high radiation if there is leakage from the RHR to the high-pressure service water. Reactor coolant is returned to the reactor vessel via either recirculation loop.

During a nuclear system shutdown and cooldown, when the shutdown cooling subsystem is initially placed in operation, decay heat levels can be high and operation of more than one RHRS heat exchanger may be required to remove the heat. When the decay heat level has decreased sufficiently, the entire shutdown cooling load can be shifted to one RHRS heat exchanger, leaving the other available for any other cooling loads.

The reactor pressure vessel shall not exceed 75 psig when a RHR pump is operating in the shutdown cooling mode.

An Alternate Decay Heat Removal (ADHR) method can be used when flooded-up during refueling. By aligning the RHR heat exchangers to the spent fuel pool skimmer surge tanks and returning flow to the reactor vessel, decay heat can be removed.

4.8.6.2 Containment Cooling

The RHR System provides a means to cool the Containment when operating in the Suppression Pool Cooling (SPC), Containment Spray Cooling (CSC), and Coolant Injection Cooling (CIC) modes.

The SPC mode of the RHR System operation provides the heat removal function for cooling the suppression pool during normal plant startup, power operation shutdown, refueling, and following a design basis event or accident.

The safety related function of this mode is to remove reactor core decay heat and sensible heat discharged to the suppression pool in the event of a design basis event or accident. This function is required to maintain the suppression pool temperature within an acceptable limit and the Containment pressure within an acceptable range during and following a design basis event or accident.

The non-safety related function of this mode is to operate occasionally to remove heat discharged to the suppression pool from sources such as reactor safety/relief valve leakage, HPCI and RCIC System turbine exhaust, and HPCI, RCIC, Core Spray, and RHR System pump heat generated during normal plant operation. This function is designed to maintain the temperature of the suppression pool water low enough to assure an adequate heat sink. The adequate heat sink, in conjunction with the Containment cooling function (i.e., the SPC or Containment Spray), is required to assure that an acceptable suppression pool temperature can be maintained throughout a plant transient, plant design basis event, or accident.

With the RHRS in the containment cooling mode of operation, the RHRS main system pumps are aligned to pump water from the suppression pool through the RHRS heat exchangers, where cooling takes place by transferring heat to the high-pressure service water. The flow returns to the suppression pool via the full flow test line (Drawing M-1-DD-9) or the reactor vessel via the LPCI injection line (CIC mode). A detailed discussion of the containment coding modes may be found in UFSAR Section 14.10.4.

A normally closed cross-tie line connects the RHR pump discharge headers upstream of the RHR heat exchangers within each loop. When only a single RHR pump is available, additional containment cooling capacity can be achieved by opening the cross-tie valve and aligning RHR flow to both RHR heat exchangers within a loop. This also requires starting a second HPSW pump to provide cooling to the second RHR heat exchanger. Flow control valves upstream of each of the RHR heat exchangers, in conjunction with flow elements with Main Control Room indication at the outlet of each heat exchanger, allow operators to balance flow for containment cooling and prevent RHR pump runout.

The containment cooling subsystem also provides additional redundancy to the CSCS's for post-accident conditions. The water pumped through the RHRS heat exchangers may be diverted to spray headers in the drywell and above the suppression pool. The spray headers in the drywell condense any steam that may exist in the drywell, thereby lowering containment pressure. The spray collects in the bottom of the drywell until the water level rises to the level of the pressure suppression vent lines, where it overflows and drains back to the suppression pool. Approximately 5 percent of this flow may be directed to the suppression chamber spray ring to cool any non-condensable gases collected in the free volume above the suppression pool.

The spray headers of the RHRS cannot be placed in operation unless the core cooling requirements of the LPCI subsystem have been satisfied. These requirements may be bypassed by the operator using a keylock switch in the control room (subsection 7.4, "Core Standby Cooling Systems Control and Instrumentation").

4.8.6.3 Low-Pressure Coolant Injection

The LPCI subsystem is an integral part of the RHRS. It operates to restore and, if necessary, maintain the coolant inventory in the reactor vessel after a LOCA so that the core is sufficiently cooled to preclude excessive fuel clad temperatures and subsequent energy release due to a metal-water reaction. A detailed discussion of the requirements and response of the equipment which operates during LPCI for a LOCA may be found in Section 6.0, "Core Standby Cooling Systems." A detailed discussion of the requirements and response of the controls and instrumentation of LPCI during a LOCA may be found in subsection 7.4, "Core Standby Cooling Systems Control and Instrumentation."

In general, LPCI operation involves restoring the water level in the reactor vessel to a sufficient height for adequate cooling after a LOCA. The LPCI subsystem operates in conjunction with the HPCIS, the ADS, and the core spray system to achieve this goal (Section 6.0, "Core Standby Cooling Systems"). This capability satisfies design basis 1.

The HPCIS is a high-head, low-flow system and pumps water into the reactor vessel when the nuclear system is at high pressure. If the

HPCIS flow is not sufficient to restore the level in the pressure vessel, the automatic depressurization feature of the nuclear system pressure relief system functions to reduce nuclear system pressure so that LPCI operates to inject water into the pressure vessel. LPCI is a low-head, high-flow subsystem and delivers rated flow to the reactor vessel when the differential pressure between the reactor vessel and the primary containment is 20 psi or less. The HPCIS turbine shuts down after pressure vessel level is restored, or when steam pressure is too low to support HPCIS turbine operation. All these operations are carried out automatically. LPCI is designed to reflood the reactor vessel to at least two-thirds core height and to maintain this level. After the core has been flooded to this height, the capacity of one RHRS main system pump is more than sufficient to maintain the level.

During LPCI operation, the main system pumps take suction from the suppression pool and discharge into the core region of the reactor vessel through the recirculation loops. Any spillage through a break in the lines within the primary containment returns to the suppression pool through the pressure suppression vent lines. A bypass line to the suppression pool is provided so that the pumps are not damaged by operating with the discharge valves shut. It is concluded that safety design basis 3 is satisfied.

Service water flow to the RHRS heat exchangers is not required immediately after a LOCA because heat rejection from the containment is not necessary during the time it takes to flood the reactor. Power for the main system pumps comes from an auxiliary power bus. Power for this bus normally comes from the auxiliary supply, but if this source is not available, power is available from the standby (diesel) ac power source.

To provide a source of water if any post-accident flooding of the primary containment is required, a cross-tie exists from the piping on the discharge of the high-pressure service water pumps to the discharge piping on the shell side of an RHRS heat exchanger. This connection is provided with redundant valving appropriate to a primary containment penetration. The valves are remotely operable from the control room. This connection provides access to a large source of water for post-accident flooding of the primary containment. The high-pressure service water pumps which provide this function can add water to either recirculation loop through the cross-connection between the piping of each RHRS loop.

4.8.7 Safety Evaluation

Since the RHRS includes the LPCI and containment cooling subsystems, and acts with other CSCS's to satisfy the safety objective, it is properly evaluated in conjunction with the other CSCS's. This safety evaluation is in Section 6.0, "Core Standby Cooling Systems." The safety evaluation of the controls and instrumentation of the LPCI subsystem is in subsection 7.4, "Core Standby Cooling Systems Control and Instrumentation."

4.8.8 Inspection and Testing

A design flow functional test of the RHRS main system pumps is performed for each pair of pumps during normal plant operation by taking suction from the suppression pool and discharging through the test lines back to the suppression pool. The LPCI injection valves to the reactor recirculation loops remain closed during this test and reactor operation is undisturbed. An operational test of these injection valves is performed by shutting the downstream valve, after it has been satisfactorily tested, and then operating the LPCI injection valve. The discharge valves to the containment spray headers are checked in a similar manner by operating the upstream and downstream valves individually. All these valves can be actuated from the control room using remote manual switches. Control system design provides automatic return from the design flow functional test to the operating mode if LPCI initiation is required during testing. It is concluded that safety design basis 4 is satisfied.

The RHR pump discharge flow control valves are verified during surveillance testing to be positioned for LPCI operability to achieve a LPCI flow rate which exceeds the Technical Specifications minimum flow and is less than the maximum runout flow for NPSH analysis. The RHR pump discharge flow control valves are not automatically positioned with LPCI initiation. A functional test of the pump discharge cross-tie valve and the flow control valves is performed by stroking the valves.

Inspection and maintenance of the main system pumps, pump motors, and heat exchangers are carried out generally in accordance with the manufacturer's instructions and good operating practice.

TABLE 4.8.1

RESIDUAL HEAT REMOVAL SYSTEM EQUIPMENT DESIGN DATA

RHRS PUMPS

Number Installed per Unit - 4
Capacity per Pump - 33 1/3% (LPCI)
Design Temperature - 350°F
Design Pressure - 450 psig

Design Conditions per pump at 20 psid* for 1 pump in operation

The original design point for the RHR pumps was a discharge head of 542 ft at a flow rate of 10,000 gpm. This was revised to 8,600 gpm for Unit 2 and 10,900 gpm for Unit 3, at a system head corresponding to a reactor pressure of 20 psig.

Operating Conditions per Pump

Discharge Flow – 0 to 10,600 gpm (Unit 2)
0 to 12,000 gpm (Unit 3)
Discharge Head - 840 to 400 ft
Differential Pressure - 295 to 0 psid

RHRS HEAT EXCHANGERS

Number Installed per Unit - 4
Shell Side Fluid - Reactor Water or Suppression Pool Water
Tube Side Fluid - High-Pressure Service Water
Design Temperature - 40 to 350°F
Design Pressure - 450 psig
Pressure Drop at Design Conditions -
Shell Side - 10 psi
Tube Side - 6 psi

Design Conditions

Shell Side Flow - 10,000 gpm
Inlet Temperature Shell Side - 165°F
Heat Exchanger Duty – 72.9 x 10^6 Btu/hr (with 5% tubes plugged)
Tube Side Flow - 4,500 gpm
Inlet Temperature Tube Side - 92°F

*psid - pounds per square inch difference between reactor vessel and drywell.

4.9 REACTOR WATER CLEANUP SYSTEM

4.9.1 Power Generation Objective

The power generation objective of the reactor water cleanup system (RWCUS) is to maintain high reactor water purity to limit chemical and corrosive action, thereby limiting fouling and deposition on heat transfer surfaces. The RWCUS also removes corrosion products to limit impurities available for neutron activation and resultant radiation from deposition of corrosion products.

4.9.2 Power Generation Design Basis

1. Provision is made for the discharge of reactor water at reduced activity during various modes of plant operation.

2. Provisions are made to minimize the heat loss and the fluid loss from the nuclear system.

4.9.3 Description

The RWCUS provides continuous purification of a portion of the recirculation flow. The processed fluid is returned to the nuclear system or to radwaste or to condensate storage tanks. Regenerative heat exchangers are provided to minimize heat loss from the nuclear system. The system can be operated at any time during planned operations. The RWCU system is operated to maintain the Reactor Water quality to within the limits specified in ERPI BWR Water Chemistry Guidelines.

The major equipment of the RWCUS is located in the reactor building and consists of two 100% capacity pumps, one three-shell regenerative and two two-shell non-regenerative heat exchangers and two filter-demineralizers with supporting equipment. The entire system is connected by associated valves and piping, and controls and instrumentation are provided for proper system operation (Drawing M-354, Sheets 1 and 2 and Drawing M-355, Sheets 1-4). Design data for the major pieces of equipment are presented in Table 4.9.1.

Reactor coolant is normally removed from the reactor coolant recirculation system and/or reactor vessel bottom head drain, cooled in the regenerative and non-regenerative heat exchangers, filtered and demineralized, and returned to the feedwater system through the shell side of the regenerative heat exchanger. A schematic diagram of the RWCUS is shown in Drawing M-1-DD-5.

Because the filter-demineralizer units are temperature limited (Table 4.9.1), the reactor coolant must be cooled prior to processing in the filter-demineralizer units. The regenerative heat exchanger transfers heat from the influent water to the effluent water, which is returned to the feedwater system. The non-regenerative heat exchanger cools the influent water further by transferring heat to the reactor building cooling water system. The non-regenerative heat exchanger is designed to maintain the lower temperature even when the effectiveness of the regenerative heat exchanger is reduced. The thermal effectiveness of the regenerative heat exchanger is reduced when excess water is being removed from the reactor vessel via the RWCUS. A part of the flow from the filter-demineralizer can be directed either to the main condenser, or the radwaste system or the condensate storage tank instead of returning to the regenerative heat exchanger. In the event the main condenser is at a high level, the condensate storage tank and radwaste system provide alternate flow paths.

The filter-demineralizer units (Drawing M-355, Sheets 1 through 4) are pressure precoat type filters using finely ground mixed ion exchange resins as a filter and ion exchange medium. Spent resins are not regenerable and are sluiced from a filter-demineralizer unit to a backwash receiver tank, from which they are processed in the radwaste system. A strainer is installed on the outlet of each filter-demineralizer unit to limit resins from entering the reactor system in the event of a resin support failure. Each strainer is provided with an alarm which is energized by high differential pressure and provide a high differential pressure filter-demineralizer isolation. A bypass line is provided around the filter-demineralizer units for bypassing the units when necessary.

Relief valves and instrumentation are provided to protect the equipment against overpressurization and the resin against overheating. The system is automatically isolated, for the reasons indicated, when signaled by any of the following occurrences:

1. High temperature downstream of the non-regenerative heat exchanger-- to protect the ion exchanger resin from damage due to high temperature.

2. Reactor vessel low water level-- to protect the core in case of a possible break in the RWCUS piping and equipment (subsection 7.3, "Primary Containment and Reactor Vessel Isolation Control System").

3. Standby liquid control system actuation-- to prevent removal of the boron by the ion exchange resin.

4. High flow in the pump suction line-- indicative of pipe rupture.

Flow is maintained through each filter-demineralizer by its own holding pump in the event of low flow or loss of flow. Sample points of the RWCUS are provided upstream and downstream of each filter-demineralizer unit. The influent sample point is also used as normal source of reactor coolant samples. Sample analysis provides an indication of the effectiveness of the filter-demineralizer units.

Operation of the RWCUS is controlled from the main control room and from a local control panel in the reactor building. Filter-demineralizer operations, which include backwashing and precoating, are controlled from the local control panel in the reactor building.

Control and instrumentation logic is presented in Drawing M-1-CC-35, Sheets 1 through 4.

The RWCU system also provides process flow to the noble metals monitoring system. The noble metals monitoring system provides indication of the effectiveness of noble metals to mitigate stress corrosion cracking (SCC) initiation and existing SCC growth on Reactor pressure vessel (RPV) wetted internal components and associated piping.

4.9.4 Inspection and Testing

Because the RWCUS is normally in service during operation of the nuclear plant, satisfactory performance is demonstrated by maintaining high reactor water purity without the need for any special inspection or testing.

Appendix I describes examination requirements for pipe welds and supports.

Subsection 7.3 describes the testing requirements for the primary containment isolation control system valves in the RWCUS.

TABLE 4.9.1

REACTOR WATER CLEANUP SYSTEM EQUIPMENT DATA
MAIN CLEANUP RECIRCULATION PUMPS

Number Installed - 2
Capacity (each) - 100%
Rated Discharge Flow
(per pump) at 533°F - 410 gpm
Design Temperature - 575°F
Design Pressure - 1,450 psig
Discharge Head at Rated Flow - 550 ft

HEAT EXCHANGERS

	Regenerative	Non-Regenerative
Reactor Coolant Flow per unit (lb/hr)	180,000 (design) 153,000 (nominal)	180,000 (design) 153,000 (nominal)
Shell Side Design Pressure (psig)	1,450	150
Shell Side Design Temperature (°F)	575	370
Tube Side Design Pressure (psig)	1,450	1,450
Tube Side Design Temperature (°F)	575	575

FILTER-DEMINERALIZERS

Number Required - 2
Capacity (each) - 50%
Flow Rate/Unit (lb/hr) - 90,000 (design), 76,500 (nominal)
Effluent Conductivity - < 0.1 $\frac{\text{micro-mho}}{\text{cm}}$ max.
Effluent pH - 6.5 to 7.5
Effluent insolubles (ppb maximum-measured as residue on 0.45 micron filter paper) - < 10
Design Temperature - 150°F
Design Pressure - 1,450 psig
Time to remove a unit from service, backwash, precoat,

and return to service - ≤ 60 360 min (nominal)

4.10 NUCLEAR SYSTEM LEAKAGE DETECTION AND LEAKAGE RATE LIMITS

4.10.1 Safety Objective

The safety objective is to provide a reliable means to detect abnormal leakage from the nuclear system process barrier so that appropriate action can be taken before the integrity of the nuclear system process barrier is unduly compromised. The system which detects and indicates abnormal leakage is a process safety system and is not required to be safety-related.

4.10.2 Safety Design Basis

1. A means is provided to detect a leakage rate less than that indicating an unacceptable level.

2. Limits on leakage are such that corrective action can be taken before unacceptable results occur. The following results are unacceptable:

   a. A threat of significant compromise to the nuclear system process barrier.

   b. A leakage rate in excess of the coolant makeup capability to the reactor vessel.

   c. Flooding of equipment required for safe operation or shutdown of the plant.

4.10.3 Description

The various conditions of leakage are defined as follows:

1. Identified Normal Design Leakage - A controlled quantity of fluid released from seals or sealing systems of piping components which are properly assembled and in good condition.

2. Abnormal Leakage - Fluid released from a small crack or damaged seal in the nuclear system process barrier at such a rate that the guideline limits of 10CFR20 could be violated.

3. Gross Leakage - Uncontrolled fluid released from a ruptured piping component at such a rate that the

guideline limits of 10CFR100 could be violated if isolation is not effected.

This subsection describes the leakage detection systems provided to detect abnormal leakage from the nuclear system process barrier, both inside and outside the primary containment. Also discussed in this subsection are nuclear system leakage rate limits and their bases.

The systems which detect gross leakage resulting from a pipe rupture and initiate automatic isolation are discussed in subsection 7.3, "Reactor Vessel and Primary Containment Isolation Control System." The controls available for manually initiating isolation are also discussed in subsection 7.3. In some cases, a leakage detection system which provides an automatic isolation signal also provides an indication or alarm signifying abnormal leakage. In such cases, the indication of alarm function provided is discussed in this subsection.

4.10.3.1 Identified Leakage Rate

The identified leakage rate is the sum of all component leakage rates which are measured.

Leakage from the reactor vessel head flange gasket is piped to a collection chamber (section of pipe) and then to the equipment drain sump. If leakage is indicated, the chamber filling time may be measured during plant operation and the leakage rate may be calculated. A more detailed discussion of this is in subsection 7.8, "Reactor Vessel Instrumentation."

Most valves and pumps in the nuclear system inside the drywell are equipped with double seals. Leakage from the primary recirculation pump seals is piped to the equipment drain sump and is instrumented as shown in the subsection 4.3, "Reactor Recirculation System." Main steam relief and safety valve leakages are identified by temperature sensors reading out in the main control room. Any temperature increase above drywell ambient could indicate leakage.

Automatic detection and isolation of a leak in the main steam line is by any one or a combination of: (1) low water level in the reactor vessel, (2) high flow rate at the flow limiter, and (3) high temperature in the main steam line tunnel. These functions are described in subsection 7.3, "Primary Containment and Reactor Vessel Isolation Control System." Manual detection of a leak in the main steam line is by high temperature in the main steam line tunnel. This signal is obtained by a temperature sensor located along the main steam line at the same location as the temperature sensors used for automatic isolation. The output of the temperature sensor is transmitted to the control room for remote temperature readout and alarm. In the event of loss of power or component failure, the temperature sensing system will annunciate in the main control room.

Figures 4.10.1 and 4.10.2 are diagrams of the drywell cooling system and drywell sumps, respectively. As shown in the figures, there are two drywell sumps. One sump, the drywell equipment drain sump, receives drainage from the pump seal leak-off, reactor vessel head flange vent drain, and other equipment drains. The second sump, the floor drain collector sump, receives control rod drive, valve stem, vent cooler drains, and flange leakages, floor drains, and closed cooling water system drains. Collection of leakage in excess of normal background amounts is indicative of a reactor coolant leak. The discharge lines from the equipment drain sump and floor drain sump to the radwaste system are provided with sample points outside the primary containment. For anticipated leakage rates of equipment and specific piping paths, see subsection 9.2, "Liquid Radwaste System."

4.10.3.2 Unidentified Abnormal Leakage Rate and Detection within the Primary Containment

The unidentified leakage rate is that portion of the total leakage rate received in the drywell sump which is not identified as described in paragraph 4.10.3.1, "Identified Leakage Rate."

A threat of significant compromise to the nuclear system process barrier exists if the barrier contains a crack which is large enough to propagate rapidly. The unidentified leakage rate is limited because of the possibility that most of the unidentified leakage rate might be emitted from a single crack in the nuclear system process barrier.

Unidentified reactor coolant leakage is limited to ≤5 gpm whenever fuel is in the reactor vessels and coolant temperature is above 212°F. For leakage of the order of 5 gpm, experimental and analytical data suggest a reasonable margin of safety that such leakage magnitude would not result from a crack approaching the critical size for rapid propagation. Leakage less than the magnitude specified can be detected in a reasonable matter of time utilizing the leakage detection schemes to allow time for corrective action to be taken before the process barrier is significantly compromised.

Unidentified leakage through the reactor coolant pressure boundary within the primary containment is detected by monitoring drywell temperatures, pressures, airborne radioactivity, and changes of volumetric discharge flow rates in the floor drain sumps. The following is a description of the systems used to monitor these variables.

4.10.3.2.1 Primary Containment Temperature Monitoring System

Temperatures within the drywell are monitored at various elevations. A drywell ambient temperature rise will indicate the presence of reactor coolant or steam leakage. This provides an indirect indication of leakage into the primary containment.

The system consists of 26 strategically placed temperature sensors. The output of twenty-five of these sensors are displayed on a digital indicator in the main control room. The output of the remaining sensor is recorded in the control room.

4.10.3.2.2 Primary Containment Pressure Monitoring System

The primary containment pressure monitoring system also provides an indirect method of detecting leakage from the RCPB. This pressure is monitored by two electronic transmitters (0 to 70 psig) located external to the containment in the reactor building and transmitting to two separate recorders located in the main control room.

Additional drywell pressure instrumentation has been provided in the control room to allow pressure measurement from 5 psia to in excess of four times design pressure of the drywell. This involves four instrument channels. Two channels are connected to pressure transmitters with the range 0 to 225 psig. Two channels are connected to absolute pressure transmitters with the range 5 to 25 psia. One of each type of channel is assigned to one of two independent safeguard power electrical divisions to ensure control room indication in event of single failure. Containment isolation is initiated when the containment pressure monitors detect an increase in drywell pressure above 2 psig. Control room alarms are annunciated at positive drywell pressures of 0.25, 0.75, 1.5, and 2 psig.

4.10.3.2.3 Primary Containment Airborne Radioactivity Monitoring System

A. Description

The primary containment is continuously monitored for airborne radioactivity. A sample is drawn from the containment atmosphere through the pumping system of the containment hydrogen/oxygen analyzer. Any sudden increase in the monitor readings indicates the presence of steam or reactor water leakage. Monitoring is accomplished through the use of a radioactive noble gas monitor.

Readout and controls are located in the reactor and radwaste buildings. The output of the digital rate meter is recorded in the control room. A common trouble alarm is located in the control room.

The skid has provisions for taking grab samples, these samples are removed to the radiochemical lab for specific isotopic determination. Comparisons can be made with preceding samples to determine the nature of any changes in the drywell atmosphere radioactive content.

B. Sensitivity and Response Time

The objective of the drywell leak detection monitor as indicated in Regulatory Guide 1.45, Position 5, is to be able to detect less than 1 gpm of unidentified primary coolant pressure boundary leakage in one hour. The radiation monitor discussed above measures activity level as an indicator of coolant leakage based on the assumption that flashing coolant will result in radioactivity in the atmosphere.

The reliability, sensitivity and response time of a radiation monitor to detect 1 gpm in 1 hour of Reactor Coolant Pressure Boundary leakage will depend on many complex factors. The major factors are discussed below.

1. Source of Leakage

   a. Location of Leakage - The amount of activity which would become airborne following a leak from the RCPB will vary depending upon the leak location and the coolant temperature and pressure. For example, a feedwater pipe leak may have concentration factors of 100 to 1000 lower than a recirculation line leak. A steam line leak may be a factor of 50 to 100 lower in iodine and particulate concentrations than the recirculation line leak, but the noble gas concentrations may be comparable.

A RWCU leak upstream of the demineralizers and heat exchangers may be a factor of 10 to 100 higher than downstream, except for noble gases. Differing coolant temperatures and pressures will affect the flashing fraction and partition factor for iodines and particulates. Thus, an airborne concentration cannot be directly correlated to a quantity of leakage without knowing the source of the leakage.

b. Coolant Concentrations - Variations in coolant concentrations during operation can be as much as 2-to-3 orders of magnitude within a time frame of several hours. These effects are mainly due to spiking during power transients or changes in the use of the RWCU system. Examples of these transients for I-131 can be found in NEDO-10585 (8/72), "Behavior of Iodine in Reactor Water During Plant Shutdown and Startup." An increase in the coolant concentrations could give increased containment concentrations when no increase in unidentified leakage occurs.

c. Other Sources of Leakage - Since the unidentified leakage is not the sole source of activity in the containment, changes in other sources will result in changes in the containment airborne concentrations. For example, identified leakage is piped to the equipment drain sump in the drywell, but the sump is vented to the drywell atmosphere allowing the release of noble gases and some small quantities of iodines and particulates from the drain sump.

2. Drywell Conditions Affecting Monitor Performance

Note: The use of MPC units presented in Section 4.10.3.2.III.B.2.a is historical and describes the analysis for the original plant design. Current units used to describe activity levels are contained in 10CFR20.

a. Equilibrium Activity Levels - During normal operation the activity release from acceptable quantities of identified and unidentified leakage will build up to significant amounts in the drywell air. Levels as high as .1 to 10 times MPC are not uncommon for noble gases and iodines. (MPC refers to "maximum permissible concentration" as defined by 10CFR20; MPC is used here only as a convenient reference.) Due to these high equilibrium activity levels the small increases due to a 1 gpm increase in leakage may be difficult to detect within an hour. Typical ranges of drywell activity level are:

0.1 MPC to 10 MPC

Noble Gases 1×10^{-6} - 1×10^{-4} μCi/cc

b. Purge and Pressure Release Effects - Changes in the detected activity levels have occurred during containment venting operations. These changes are of the same order of magnitude as approximately a 1 gpm leak, and are sufficient to invalidate the results from iodine and particulate monitors.

c. Plateout, Mixing, Fan Cooler Depletion - Plateout effects on measured iodine and particulate levels will vary with the distance from the coolant release point to the detector. Larger travel distances would result in more plateout. In addition the pathway of the leakage will influence the plateout effects. For example, a leak from a pipe with insulation will have greater plateout than a leak from an uninsulated pipe. Although the drywell air will be mixed by the fan coolers, it may be possible for a leak to develop in the vicinity of the radiation detector sample lines. In addition, condensation in the coolers will remove iodines and particulates from the air. Variations in the flow, temperature and number of coolers will affect the plateout fractions. Plateout within the detector sample chamber will also add to the reduction of the iodine and particulate activity levels. The uncertainties in any estimate of plateout effects could be as much as one or two orders of magnitude.

3. Physical Properties and Capabilities of the Detectors

a. Detector Range - The detector was chosen to ensure that the operating range covered the noble gas concentrations expected in the drywell.

The Operating Range is: 1×10^{-6} to 1 μCi/cc

b. Sensitivity - In the absence of background radiation and equilibrium drywell activity levels, the Noble Gas detector has the following minimum sensitivity:

1×10^{-6} μCi/cc for gamma emissions of 80 keV and above.

c. The radioactivity monitor is statistically able to detect increases in activity. The uncertainities in calibrating sample flow and other instrument design parameters tend to make the uncertainity in activity closer to the 20% to 40% of the equilibrium drywell activity.

d. Monitor Setpoint - Due to the uncertainty and extreme variability of the concentrations to be measured in the containment, the use of an alarm setpointon the radioactivity monitor would not be practical or useful. As indicated below the setpoint which would be required to alarm at 1 gpm would be well within the bounds of uncertainty of the measurements. The use of such a setpoint would result in many unnecessary alarms and the frequent resetting of the setpoint. An indication of leakage is initially obtained from sump level monitoring; the radioactivity monitor is for supporting information to confirm that the leak is radioactive. The alarm setpoint for the radiation monitor is set significantly above normal readings to prevent nuisance alarms.

C. Conclusion

There is no direct correlation or known relationship between the detector count rate and the leakage rate, because the coolant activity levels, source of leakage, and background radiation levels (from leakage alone) are not known and cannot be cost-effectively determined in existing reactors. There are also several other sources of containment airborne activity (e.g. safety relief valve leakage) which further complicate the correlation.

4.10.3.2.4 Drywell Floor Drain Sump Monitoring System

The drywell floor drain sump collects unidentified leakage from vent coolers, control rod drive flanges, chilled water drains, cooling water drains, valve stems and flanges, leaks and any overflow from the equipment drain sump.

The plant Technical Specifications limit the amount of coolant leakage from unidentified sources to less than or equal to 5 gpm and total leakage from both identified and unidentified sources to less than or equal to 25 gpm. Both the equipment drain sump and the floor drain sump have capacities of 500 gallons, and each is equipped with two 50 gpm pumps. The capacity of these pumps is in excess of the maximum allowable leakage rates because of the possibility that most of the leakage could flow into one sump.

Each sump has an alarm system and automatic pump starting sequence as follows: At the first high water level setting, the preferred pump is automatically started and pump operation is indicated. If the water level continues to rise, a higher water level setting starts the standby pump and actuates an alarm. The pumps are alternately selected for operation by an automatic pump selector switch. Therefore, an alarm would suggest that leakage into that sump is exceeding the capacity of one sump pump. If only one pump is operational, the Hi-Hi sump level alarm will occur every other pump start.

As the water which has been collected in the sump is pumped out, the discharge flow from each sump is individually metered by flow integrators. Total leakage rate is periodically calculated from these flow integrators. A flow recorder continually plots time versus discharge flow rate from each sump; an increase in leakage rate is detectable by an increase in sump discharge flow time and an increased frequency in discharge flow cycles.

The discharge from each of the sumps is monitored every four hours. The operator verifies at each interval that (1) floor drain leakage has not exceeded an average of 5 gpm, (2) floor drain leakage plus equipment drain leakage has not exceeded an average of 25 gpm, and (3) that floor drain leakage has not increased by 2 gpm per 24-hour surveillance period when the reactor is operated in the "RUN" mode. If any of these limits are exceeded, a plant shutdown is initiated in accordance with the Technical Specifications.

There is no exact quantitative relationship between the sump discharge and the leakage rate from any source. The quantity is dependent upon the temperature and pressure of the containment and the leak, and the location of the leak. Part of the leak will flash to steam; it may also be partially trapped between insulation layers. Presumably the leakage will get to an equilibrium level where most of it ends up in the sump, unless the drywell is being vented. Small variations in the ability to relate the sump quantity to the actual leaked quantity are ignored, and it is assumed that all leakage reaches the sump. The errors introduced will not impair the ability to detect larger leaks which could rapidly result in severe accidents. Some leakage will no doubt be trapped in insulation etc., but no large reservoirs for leakage are present within the primary containment.

4.10.3.2.5 Conclusion

The recommended procedure for the control room operator is to monitor the sump volumetric discharge as is discussed in 4.10.3.2.4 above, (by measuring water collected in the sump which may not precisely correspond to water leaking from an unidentified source). If an alarm is received indicating high level in either of the drywell sumps, the operator will review all other monitors (e.g., noble gas, temperature, pressure, etc.) to determine if the leakage is from the primary coolant pressure boundary and not from an SRV or cooling water system, etc. Appropriate actions will then be taken in accordance with Technical Specifications. The review of other monitors will consist of comparisons of the increases and rates of increase in the values previously recorded on the strip chart recorders. Increases in all parameters except sump discharge will not be correlated to a RCPB leakage rate. Instead, the increases will be compared to normal operating limits and limitations (e.g., 2 psi maximum pressure for ECCS initiation) and abnormal increases will be investigated.

Since the Technical Specification limit for leakage is allowed to be averaged over 24 hours, quick and accurate responses are not necessary unless the leakage is very large and indicative of a pipe break. In this case, the containment pressure and reactor vessel water level monitors will alarm within seconds, and the sump high level monitor would alarm within minutes or tens of minutes.

Radiation monitor alarms are not set to levels that are intended to correspond the RCPB leakage levels since the correlations can't be made. Since the containment airborne activity levels vary by orders of magnitude during operation due to power transients, spiking, steam leaks, and outgassing from sumps, etc., an appropriate alarm setpoint is determined by the operator based on experience with the specific plant. A set point level of 10 times the normal level during full power steady state operation may be useful for alarming large leaks and pipe breaks, but it would not always alarm for 1 gpm in 1 hour and therefore could not be considered as any more than a qualitative indication of the presence of abnormal leakage.

Due to the sum total of the uncertainties identified in the previous paragraphs the iodine and particulate monitors are not relied upon for immediate leak detection purposes but only as supporting instrumentation by the means of sampling. The noble gas monitor is used to give supporting information to that supplied by the sump discharge monitoring and it would be able to give an early warning of a major leak especially if equilibrium containment activity levels are low. However, the uncertainties and variations in noble gas leaks and concentrations would preclude the setting of a meaningful alarm set point.

4.10.3.3 Total Leakage Rate within the Primary Containment

Limits on RCS operational leakage are required to ensure appropriate action is taken before the integrity of the reactor coolant system is impaired. A limited amount of leakage inside primary containment is expected from auxiliary systems that cannot be made 100% leak tight. Based on experimental evidence,Total leakage rate consists of all leakage, identified and unidentified, which flows to the drywell floor drain and equipment drain sumps. The criterion for establishing the total leakage rate limit is based on the makeup capability of the CRDS and the RCICS, and independent of the feedwater system, normal ac power, and the CSCS's. The CRDS supplies 63 gpm into the bottom of the reactor vessel; the RCICS can supply 600 gpm through the feedwater sparger to the reactor vessel. The the total leakage rate limit is established at 25 gpm. **This ensures that the probability is small that an imperfection or crack will not grow rapidly. The total leakage rate consists of both identified and unidentified leakage. No known reactor pressure boundary leakage is allowed (leakage past seals and gaskets is not considered pressure boundary leakage).**

The drywell contains two sumps that collect all leakage. The equipment drain sump and the floor drain sump each have a capacity of 500 gallons. Each is drained by two 50 gpm pumps. The pumps are sized assuming all leakage goes to one sump and

one pump in that sump fails. The total leakage rate is also set low enough to prevent overflow of the drywell sumps. The equipment drain sump (capacity 500 gal) and the floor drain sump (capacity 500 gal), which collect all leakage, are each drained by two 50-gpm pumps. The total leakage rate limit is set below the removal capacity of the two pumps in each sump because of the possibility that most of the total leakage could flow into one sump.

Each sump has an alarm system and automatic pump starting sequence. as follows: At the first high water level setting (hi setpoint), the preferred pump is automatically started and pump operation is indicated. If the water level continues to rise, a higher water level setting (hi-hi setpoint) starts the standby pump and actuates an alarm. The pumps are alternately selected for operation by an automatic pump selector switch. Therefore, an alarm would suggestThe alarm indicates that leakage into that sump is exceeding the capacity of one sump pump. If only one pump is operational, the hi-hi- sump level alarm will occur every other pump start. **To provide operational flexibility in the event of a failed pump, the remaining pump (in the equipment drain sump) can be set to operate at the hi-hi setpoint with a time delay for the control room alarm.**

As the water which has been collected in the sumps is pumped out, the discharge flow from each sump is individually metered by flow integrators. Total leakage rate is periodically calculated from these flow integrators. A flow recorder continually plots time versus discharge flow rate from each sump; an increase in leakage rate is detectable by an increase in sump discharge flow time and an increased frequency in discharge flow cycles. Increases in total leakage rate are also detectable from records kept of once a day checks of flow integrator readings.

4.10.3.4 Detection of Abnormal Leakage Outside the Primary Containment

Outside the primary containment, the piping within each system monitored for leakage is in compartments or rooms, separate from other systems where feasible, so that leakage may be detected by area temperature indications. Each leakage detection system discussed below is designed to detect leak rates that are less than the established leakage limits.

a. Room Ventilation Temperature

A ventilation air temperature sensing system is installed in each room containing equipment which is part of nuclear system process barrier. This includes the HPCIS, RCICS, RHRS, and RWCUS equipment rooms, as well as the main steam line tunnel.

b. Room Ambient Temperature

A temperature sensor is provided in the HPCIS, RCICS, RHRS, and RWCUS equipment rooms (except for the RWCU pump rooms), as well as in the main steam tunnel. The temperature recorder associated with these sensors initiate an alarm at that ambient temperature rise for their particular area calculated to be indicative of the limiting leakage. Remote readouts from temperature sensors located in the rooms are indicated in the control room.

c. Reactor Building Sump Measurement

Level switches and alarms monitor abnormal leakage into the sumps. The normal design leakage collected in the reactor building sumps consists of leakage from the RWCUS, fuel pool cooling and cleanup system, RCICS, HPCIS, CRDS and RHRS, and from other miscellaneous vents and drains. The HPCI, RCIC, RHR, and Core Spray floor drains are plugged to prevent flooding between the rooms as discussed in Appendix J. Administrative limits and controls have been established to identify small leaks from the RWCUS in the RWCU pump rooms prior to anticipated catastrophic failure of the high energy piping.

d. Flooding Detection in the Standby Core Cooling Pump Rooms and Torus Compartment

Instrumentation for detection of the torus cavity flooding condition consists of level switches in the torus cavity and in each of the pump rooms for the core spray system, RHRS, RCICS, and HPCIS. A float type liquid level indicator of the volume within the torus is also provided. High levels in the various cavities are annunciated in the main control room. Instrumentation is provided to detect any significant leakage in the torus cavity and all ECCS rooms.

Leakage in any one core cooling pump room, even without the addition of external water into the torus, does not impair the pumping capability of other core cooling pumps due to the single suction line to each pump from the torus and watertight room design.

e. Visual and Audible Inspection

Accessible areas are inspected periodically. The temperature and flow indicators discussed above are monitored per plant procedures. Any instrument indication of abnormal leakage can be investigated.

4.10.4 Safety Evaluation

There are methods of detecting abnormal leakage from each system within the nuclear system process barrier. The instrumentation is so designed that it may be set to provide alarms at levels based on established leakage rate limits. The alarm points are determined analytically, based on design data and on measurements of appropriate parameters determined during startup and pre-operational tests. This satisfies safety design basis 1.

Allowable leakage rates of coolant from the reactor coolant system have been based on the predicted and experimentally observed behavior of cracks in pipes. The normally expected background leakage due to equipment design and the detection capability for determining coolant system leakage were also considered in establishing the limits. The behavior of cracks in piping systems has been experimentally and analytically investigated as part of the AEC/NRC-sponsored Reactor Primary Coolant System Rupture Study (the Pipe Rupture Study). Work utilizing the data obtained in this study indicates that leakage from a crack can be detected before the crack grows to a dangerous or critical size by mechanically or thermally induced cyclic loading, or stress corrosion cracking or some other mechanism characterized by gradual crack growth. This evidence suggests that for leakage somewhat greater than the limit specified for unidentified leakage, the probability is small that imperfections or cracks associated with such leakage would grow rapidly. The established limit is sufficiently low that even if the entire unidentified leakage rate were coming from a single crack in the nuclear system process barrier, corrective action could be taken before the integrity of the barrier would be threatened with significant compromise. This satisfies safety design basis 2a.

The limit on total leakage also allows a reasonable margin below the discharge capability of the sump pumps. The capacity of each drywell floor sump pump is 50 gpm and the capacity of each drywell equipment sump pump is also 50 gpm. Removal of 25 gpm from either of these sumps can be accomplished with margin. Thus, the established total leakage rate limit allows sufficient time for corrective action to be taken, before either the nuclear system coolant makeup or the sump removal capabilities are exceeded. Therefore, safety design bases 2b and 2c are satisfied.

4.10.5 Inspection and Testing

Because the sump pumps are automatically started and their operation is verified by the alarms and discharge flow instrumentation, no special inspection or testing during power operation of the plant is necessary. From and after the date that one of these systems is made or found to be inoperable for any reason, reactor power operation is permissible in accordance with the Required Actions of Technical Specification 3.4.5. The pumps and controls are inspected and tested during scheduled shutdowns.

4.11 MAIN STEAM LINES, FEEDWATER PIPING, AND DRAINS

4.11.1 Power Generation Objective

The power generation objective of the main steam lines is to conduct steam from the reactor vessel through the primary containment to the steam turbine. The power generation objective of the feedwater lines is to provide the piping path for delivery of water back to the reactor vessel.

4.11.2 Power Generation Design Basis

1. The main steam and feedwater lines are designed with suitable accesses to allow in-service testing and inspections.

2. The main steam lines are designed to conduct steam from the reactor vessel over the full range of reactor power operation.

3. The feedwater lines are designed to conduct water to the reactor vessel over the full range of reactor power operation.

4.11.3 Safety Design Basis

The main steam and feedwater lines are designed to accommodate operational stresses, such as internal pressures, without a failure which could lead to a release of radioactivity in excess of the guideline values in published regulations.

4.11.4 Description

The feedwater piping is designed to conduct water from sources outside the primary containment to the reactor vessel. The general requirements of the feedwater system are covered in subsection 7.10, "Feedwater Control and Instrumentation," and subsection 11.8, "Condensate and Feedwater Systems." All main steam and feedwater piping are classified according to service and locations. The materials used in piping are in accordance with the applicable design code and supplementary requirements. A diagram of the main steam and feedwater piping is shown in Drawing M-351.

The main steam piping is designed to conduct steam from the reactor vessel through the primary containment to the steam turbine. Four steam lines are utilized between the reactor and the turbine. The use of these multiple lines permits turbine stop valve and main steam line isolation valve tests during plant operation with a minimum amount of load reduction. To fully achieve this objective, the four steam lines are manifolded upstream of the turbine stop valves. This manifold also ensures that the turbine bypass system is connected to the operative steam lines and not to idle lines.

As indicated in paragraph C.1.2 and Appendix A, each main steam line up to and including the main steam line isolation valve external to the primary containment is seismic Class I. The main steam line anchor is in the seismic Class I portion of the main steam line and, therefore, does not separate the seismic Class I part of the main steam line from the seismic Class II part. The main steam lines from outer main steam isolation valves up to but not including the turbine stop valves, meet the stress requirements of seismic Class I piping as defined in Appendix A. Additional restraints are provided for the seismic Class II portions of the main steam lines to protect the adjacent seismic Class I piping in the pipe tunnel. The design methods and design stress criteria are similar to those provided for Monticello Unit 1.

The feedwater piping from the reactor out to and including the reactor feed pump discharge gate valves and branch connections up to and including the first valve are designed to meet the stress requirements of seismic Class I piping as defined in Appendix A.

Outside the drywell the main steam and feedwater lines were analyzed for transient conditions that induce dynamic loadings in the pipe, such as sudden turbine stop valve closure resulting from a main steam turbine trip or operation of feedwater minimum flow valves. Restraints are placed as necessary to limit the motion of the piping during these transients.

A drain line is connected to the low points of each main steam line, both inside and outside the drywell. Both sets of drains are manifolded and connected by valving to permit drainage to the main condenser hotwell. An orifice is provided around the final valve to the condenser hotwell to permit continuous draining of the steam line low points. The inside steam line drains slope downward from the steam line low point to the orifice outside the drywell. The drain line from the orifice to the condenser hotwell slopes down to the main condenser. An additional drain is provided from the low point of the drains to clean radwaste to permit purging the lines for maintenance.

The inside and outside steam line drains are capable of being utilized to equalize pressure across the main steam line isolation valves prior to restart following a steam line isolation. Assuming all steam line isolation valves have closed and the steam lines outside the drywell have been depressurized, the isolation valves outside the drywell are opened first; the drain lines are then used to warm up and pressurize the outside steam lines. Finally, the main steam line isolation valves inside the drywell are opened.

Quality control methods are used during the fabrication and assembly of main steam and feedwater piping to ensure that the design specifications are met.

4.11.5 Safety Evaluation

Differential pressures on reactor internals under the assumed accident conditions of a ruptured steam line are limited by the utilization of flow restrictors and the utilization of four main steam lines. All main steam and feedwater piping is designed as described in Appendix A. Design of piping in accordance with these requirements ensures the meeting of the safety design basis.

4.11.6 Inspection and Testing

In-service inspection is considered in the design of the main steam and feedwater piping. This consideration assures adequate working space and access for inspection of selected components.

Main steam piping received non-destructive testing as described in Appendix A for Group II, Schedule II. The turbine stop valves received the following non-destructive testing:

1. 100 percent magnetic particle inspection
2. Radiographic inspection
3. Dimensional check
4. Hydrostatic test of valve body
5. Hydrostatic test of seat leakage
6. Stem clearance check
7. Closing time test

8. Valve lift test.

In-service inspection is in accordance with Appendix I and the Peach Bottom Units 2 and 3 Technical Specifications.