4.0 REACTOR COOLANT AND ASSOCIATED SYSTEMS

4.1 REACTOR COOLANT SYSTEM

4.1.1 DESIGN BASIS

The Reactor Coolant System (RCS) initially operated at a power level of 2570 MWt. Due to the conservative design of the RCS, it is capable of attaining higher power levels. Rated thermal power was upgraded first to 2700 MWt, and then increased once again to 2737 MWe as part of a measurement uncertainty recapture modification. The major systems and components that bear significantly on the acceptability of the site have been evaluated for operation at a core power level of 2737 MWt. The principal design parameters for the RCS after the approval for 2737 MWt operation are listed in Table 4-1. The design parameters for each of the major components are given in Section 4.1.3 for each individual component. The RCS is designated a Category 1 system for seismic design and is designed to the criteria for load combinations and stress, which are presented in Table 4-8.

The number of seismic loading cycles used for design was based upon the occurrence of 40 full cycles, with significant motion peaks, during one seismic event and the assumption that 5 seismic events may occur during the life of the plant. These design loads were calculated for longitudinal and circumferential breaks occurring at any point in the geometry of the RCS without restriction. The worst case was selected for each geometry and for each direction.

The system design temperature and pressure are conservatively established and exceed the combined normal operating value and the change due to anticipated operating transients. They include the effects of instrument error and the response characteristics of the control system. The change due to the anticipated transients also considers the effect of reactor core thermal lag, coolant transport time, system pressure drop and the characteristics of the safety and relief valves.

The following design cyclic transients, which include conservative estimates of the operational requirements for the components discussed in Section 4.1.3, were used in the fatigue analyses required by the applicable codes listed in Table 4-9. The design cyclic transients for the operational requirements of the steam generator (SG) and pressurizer are discussed in Sections 4.1.3.2 and 4.1.3.5, respectively:

a. 500 [300 for replacement reactor vessel closure head (RVCH)] heatup and cooldown cycles during the system 40-year design life at a heating and cooling rate of 100°F/hr between 70°F and 532°F;

b. 15,000 (11,000 for replacement RVCH) power change cycles over the range of 15% to 100% of full load with a ramp load change of 5% of full load per minute increasing and decreasing;

c. 2,000 cycles of 10% of full load step power changes, increasing from 10% to 90% of full power and decreasing from 100% to 20% of full load;

d. 10 cycles of hydrostatic testing the RCS at 3125 psia and a temperature at least 60°F above the Nil Ductility Transition Temperature (NDTT) of the component having the highest NDTT;

e. 320 cycles of leak testing at 2500 psia and at a temperature at least 60°F greater than the NDTT of the component having the highest NDTT;

f. 10^6 cycles of normal variations of ±100 psi and ±6°F at operating temperature and pressure; and

g. 400 reactor trips from 100% power.

In addition to the above list of normal design transients, the following abnormal transients were also considered when arriving at a satisfactory usage factor as defined in the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, Section III; however, these transients were not used to form the basis for the code design of the components.

a. 40 cycles of loss of turbine load from 100% power without a direct reactor trip [replacement steam generators (RSGs) used 50 cycles];
b. 40 cycles of total loss of reactor coolant flow when at 100% power; and
c. 5 cycles of loss of secondary system pressure.

4.1.1.1 Effect of Steam Generator Replacement on Structural Design Basis

The original steam generators (OSGs) provided by Combustion Engineering, Inc. (CE) have been replaced by SGs designed and fabricated by Babcock & Wilcox, Canada (BWC). The RSGs are designed to be "form, fit, and function" replacements of the OSGs, with only minor differences in design and functionality. The RSGs consist of a combination of retained OSG components and replacement components. The OSG steam drum shell and the upper half of the transition cone shell are retained while the rest of the pressure boundary and all of the internals are replaced. Reference 40 provides a comparison of the OSGs and RSGs for the factors that have the potential to affect the structural evaluation of the RCS loop. The differences are summarized below:

a. The weight of the RSG is 2% greater than that of the OSG.
b. The center of gravity of the RSG is 1% lower than that of the OSG.
c. The replacement secondary shell of the RSG is 35% thinner than that of the OSG.
d. The RSG replacement pressure boundary materials have 3.5% higher coefficients of thermal expansion than those of the OSG.
e. The RSG replacement pressure boundary materials have 3.5% lower Young's Modulii than those of the OSG.

Reference 40 has evaluated the effects of the differences between the RSGs and OSGs on the structural qualification of the RCS components, component supports, and piping for pressure, thermal, deadweight, seismic, and pipe rupture loading. It concluded the existing RCS design basis loading and analyses remain valid for all components.

4.1.2 SYSTEM DESCRIPTION

The function of the RCS is to remove heat from the reactor core and internals and transfer it to the secondary (steam generating) system. The RCS, which is entirely located within the containment, consists of two heat transfer loops connected in parallel across the reactor pressure vessel. Each loop contains one SG, two circulating pumps, connecting piping, and flow and temperature instrumentation. Coolant system pressure is maintained by a pressurizer connected to one of the loop hot legs.

The RCS is shown in Figure 4-1 (Unit 1) and Figure 4-17 (Unit 2). During operation, the four pumps circulate water through the reactor vessel where it serves as both coolant and moderator for the core. The heated water enters the two SGs, transferring heat to the secondary (steam) system, and then returns to the pumps to repeat the cycle.

System pressure is maintained by regulating the water temperature in the pressurizer where steam and water are held in thermal equilibrium. Steam is either formed by the pressurizer heaters or condensed by the pressurizer spray to limit the pressure variations caused by contraction or expansion of the reactor coolant. The pressurizer is located with its base at a higher elevation than the reactor coolant loop piping. This eliminates the need for a separate pressurizer drain, and ensures that the pressurizer is drained before maintenance operations. A vent is provided in the pressurizer vapor sample line for removal of non-condensable gases during natural circulation.

Overpressure protection is provided by two power-operated relief valves (PORVs) and two spring-loaded safety valves connected to the top of the pressurizer. Steam discharged from the valves is cooled and condensed by water in the quench tank. The reactor coolant vent lines from the reactor vessel and the pressurizer also discharge to the quench tank. In the unlikely event that the discharge exceeds the capacity of the quench tank, the tank is relieved to the containment via the quench tank rupture disc. The quench tank is located at a level lower than the pressurizer. This ensures that any PORV or pressurizer safety valve leakage from the pressurizer, or any discharge from these valves, drains to the quench tank.

The RCS and its associated controls are designed to accommodate plant step load changes of ±10% of full power and ramp changes of ±5% of full power per minute without reactor trip. The system will accept, without damage, a complete loss of load.

To maintain reactor coolant chemistry within the limits discussed in Section 4.1.4.2.3 and to control pressurizer level, a continuous but variable letdown flow from one loop upstream of the reactor coolant pump (RCP) is maintained. This bleed flow is controlled by pressurizer level. Constant coolant makeup is added by charging pumps in the Chemical and Volume Control System (CVCS).

An inlet nozzle on each of the four reactor vessel inlet pipes allows injection of borated water into the reactor vessel from the Safety Injection System in the event emergency core cooling is needed. During a normal plant shutdown, these nozzles are also used to supply shutdown cooling flow from the low pressure safety injection pumps. An outlet nozzle on one reactor vessel outlet pipe is used to remove shutdown cooling flow.

There are two normally closed, solenoid valves in each of the vent lines from the reactor vessel and the pressurizer. These valves fail closed.

Drains from the reactor coolant piping to the radioactive waste processing system are provided for draining the RCS for maintenance operations. A connection is also provided on the quench tank for draining it to the radioactive waste processing system following a relief-valve or safety-valve discharge.

The major RCS components are designed for a 40-year service life. To assure that this objective can be attained, strict quality assurance standards as outlined in Sections 4.1.5.5 and 4.1.5.6 were followed.

Protection provided the RCS against environmental factors such as fires, floods and missiles are described in Sections 1.0, 5.0, 9.0, and 11.0.

4.1.3 <u>COMPONENT DESCRIPTION</u>

4.1.3.1 <u>Reactor Vessel</u>

The reactor vessel (Figure 4-2) is supported vertically and horizontally by three pads welded to the underside of the reactor vessel nozzles. Each assembly consists of the following:

a. A support foot (SA-508 CL2) welded to a reactor coolant nozzle;

b. A socket [American Society for Testing and Materials (ASTM A283-67)] bolted to the support foot with Allenoy cap screws; and
c. A sliding bearing (ASTM B22, Alloy E) whose spherical crown fits into the socket and whose flat sliding surface rests on a base plate (AISI-4140).

Design parameters for the reactor vessel are given in Table 4-2.

The arrangement of the vessel supports allows radial growth of the reactor vessel due to thermal expansion while maintaining it centered and restrained from movement caused by seismic disturbances. Departure from levelness of not more than 0.005 inch per foot of flange diameter is maintained during construction to facilitate proper assembly of reactor internals. The supports were analyzed in accordance with ASME B&PV Code, Section III.

The vessel closure flange is a forged ring with a machined ledge on the inside surface to support the reactor internals and core. The flange is drilled and tapped to receive the closure studs and is machined to provide a mating surface for the reactor vessel closure seal. The vessel closure contains 54 studs, 7" in diameter, with 8 threads per inch. The stud material is ASTM A540, Grade B24, with a minimum yield strength of 130,000 psi. The tensile stress in each stud when elongated for operational conditions is approximately 40 ksi. Calculations show that 32 uniformly distributed studs can fail before the closure will separate at design pressure. However, 16 uniformly distributed broken studs or 4 adjacent studs is necessary before the closure will fail by "zippering" open.

Analysis has shown that two thirds of the reactor vessel head studs can be removed while in Mode 5 in preparation for refueling operations. Eighteen of the total 54 studs remain in place (every third stud), tensioned to a pre-load equal to a minimum of 75% of design pre-load, with RCS pressure equal to 500 psia. Two loading conditions are considered: (1) RCS temperature equal to the saturation temperature, 467°F, simulating the worst case (highest) temperature, which could occur as a result of a loss of shutdown cooling; and (2) RCS temperature equal to 200°F, which is the normal maximum temperature for Mode 5 operation. The analysis indicates that the reactor vessel O-rings remain in compression during both conditions (no leakage), and the stud stresses due to detensioning in Mode 5 are less than allowable. Calculated stresses within the flanges which arise from Mode 5 detensioning are bounded by the stresses from the original design. It was also determined that the minimum number of studs required to properly seat the vessel head in Mode 5 is 12, but 18 is used for conservatism. Also, the single LTOP shutdown cooling PORV setpoint will be used to ensure reactor vessel pressure is maintained ≤ 500 psia.

Six radial nozzles on a common plane are located just below the vessel closure flange. Extra thickness in this vessel-nozzle course provides most of the reinforcement required for the nozzles. Additional reinforcement is provided for the individual nozzle attachments. A boss located around each outlet nozzle on the inside diameter of the vessel wall provides a mating surface for the internal structure which guides the outlet coolant flow. This boss and the outlet sleeve on the core support barrel are machined to a common contour to reduce core bypass leakage. A fixed hemispherical head is attached to the lower end of the shell. There are no penetrations in the lower head.

The removable top closure head is hemispherical. The head flange is drilled to match the vessel flange stud bolt locations. The 54 stud bolts are fitted with spherical washers located between the closure nuts and head flange to maintain stud alignment during head flexing due to boltup. To ensure uniform loading of the closure seal the studs are hydraulically tensioned with a special tool.

Flange sealing is accomplished by a double-seal arrangement utilizing two silver-plated Ni-Cr-Fe alloy, self-energized O-rings. The space between the two rings is monitored to allow detection of any inner ring leakage. The control element drive mechanism (CEDM) nozzles (Ni-Cr-Fe alloy through the head, stainless steel flange) terminate with threaded and seal-welded flanges at the upper end. There are eight instrumentation nozzles that terminate with a threaded flange and swagelok type seal for the instruments. In addition to these nozzles there is a 3/4" vent connection.

The core is supported from an internally machined core support ledge. The CEDMs are supported by the nozzles in the reactor vessel head.

4.1.3.2 Steam Generator

Replacement Steam Generator

The nuclear steam supply system (NSSS) utilizes two SGs (Figures 4-3A and 4-3B) to transfer the heat generated in the RCS to the secondary system. The SG shell is constructed of carbon steel. Manways and handholes are provided for easy access to the SG internals. The design parameters for the SGs are given in Table 4-3.

The SG is a vertical U-tube heat exchanger. The SG operates with the reactor coolant in the tube side and the secondary fluid in the shell side.

Reactor coolant enters the SG through the inlet nozzle, flows through 3/4" OD U-tubes, and leaves through two outlet nozzles. A vertical divider plate in the lower head separates the inlet and outlet plenums. The plenums are stainless steel clad with a SB-168 Alloy 690 unclad divider plate, while the primary side of the tube sheet is UNS N06052 cladding. The vertical U-tubes are SB-163 Alloy 690. The tube-to-tube sheet joint is welded on the primary side. Tubes that have degraded may be repaired using NRC approved methods. Any gases that may be trapped in the SG U-tubes are swept out by the initial reactor coolant flow and released via the pressurizer, CEDM, or reactor vessel venting systems. The maximum number of plugged tubes is 847 (10 percent of total tubes) per SG. If more than 847 tubes per SG are plugged, RCS flow may decrease below the Technical Specification required design flow. Of the 847 tubes, up to 20 may be plugged as a result of an intentional removal of a piece of the tube for purposes such as material testing (Section 14.20.1). The maximum allowed difference between the number of tubes plugged in each SG is 750.

Feedwater enters the SG through the feedwater nozzle where it is distributed via a feedwater distribution ring. The SG feedwater distribution system incorporates an all welded SA-335 P22 schedule 80 piping design, a “gooseneck” feed system positioned between the thermal sleeve and the feedwater header, and top discharging hairpin bend J-tubes to minimize the risk of water hammer, thermal flow stratification, and erosion. The water exits through J-tubes on the top of the feedwater ring, then flows into the downcomer. The downcomer is an annular passage formed by the inner surface of the SG shell and the cylindrical shell wrapper, which encloses the U-tubes. At the bottom of the downcomer, the secondary water is directed upward past the vertical U-tubes where heat transfer from the primary side produces a water-steam mixture.

Upon exiting from the vertical U-tube heat transfer surface, the steam-water mixture enters the primary separators, which are centrifugal-type separators. The mixture enters the curved arms of the primary separator where a film of water develops on the inner wall of the return cylinder, and spirals downward for recirculation. Steam at greater than 90% quality exits the top of the primary separators into the inter-stage region, which distributes steam prior to the secondary separators. The secondary separators are also centrifugal-type separators. Once the mixture is through the secondary separators, the design maximum moisture carryover is 0.05% (by weight) for 2717 MWt Reactor Core Power plus reactor coolant pump heat. Field experience and laboratory testing has demonstrated the primary and secondary separators are insensitive to operational pressure fluctuations, steam flow imbalances, and wide range in water level fluctuations. The separators are designed to last for the life of the RSG without maintenance or periodic cleaning. The primary separators have large flow passages that preclude plugging even if deposition occurs. The secondary separator inlet body and outlet passages are also large. The skimmer and vent hole passages of the secondary separator are continuously swept by flow during operation. In the unlikely event of skimmer or vent hole plugging, cleaning by water lancing is possible.

The installation of the RSGs involved first cutting and removing the upper (steam drum) portion of each OSG. The steam drums were then refurbished and reinstalled atop the new RSG lower assemblies. The previous paragraph discusses design improvements in moisture separation equipment for the RSGs. Other design improvements include:

a. Use of improved SG tube materials and fabrication to increase tube reliability.
b. Use of improved tube support materials to minimize flow-assisted corrosion of the tube supports.
c. Design measures taken to optimize feedwater distribution within the SGs, minimize thermal stratification, and preclude water hammer.
d. Design improvements to the blowdown functions of the RSGs for enhanced SG water chemistry control.
e. Use of an integral steam flow restrictor to enhance the results of the main steam line break accident analysis.

In performing the “2-piece” replacement of the SGs, the code of record for the steam drums vessels is and will remain the ASME III 1965 Edition, with Winter 1967 Addenda. New items (RSG lower assembly and feedwater transition piece) were constructed to meet all requirements of ASME Section III, 1989, Edition, No Addenda.

Overpressure protection for the shell side of the SGs and the main steam line piping up to the inlet of the turbine stop valve is provided by 16 spring-loaded ASME code safety valves that discharge to atmosphere. The power-operated steam dump valves and steam bypass valves obviate opening of the main steam safety valves following turbine and reactor trip from full power. The Main Steam System, including the steam dump and bypass valves, and the main steam safety valves, is described in Section 10.1.

The SGs are mounted on bearing plates which allow lateral motion due to thermal expansion of the reactor coolant piping. Stops are provided to limit this motion in case of a coolant pipe rupture. The top of each unit is restrained from sudden lateral movement by suitable stops and hydraulic snubbers mounted rigidly to the concrete structure. Each SG is supported by a lower support assembly consisting of:

a. A support skirt (SA-533, Type B, Class 1) welded to the high pressure head;

b. A sliding base (ASTM A27, Gr 70-40) bolted to the skirt with ASTM A193-97 bolts and ASTM A194, Gr 7 nuts; and

c. Four sockets (ASTM A283-67) are bolted to the underside of the base with Allenoy cap screws and a sliding bearing (ASTM B22, Alloy E) whose spherical crown fits into each socket and whose flat sliding surface rests on a base plate (AISI 4140).

The supports were analyzed in accordance with ASME B&PV Code, Section III. The upper SG supports consist of:

a. Eight clevices (SA-533, Gr. B, Class 1) welded to the upper shell;

b. A hydraulic snubber is attached to each clevice by means of a pin (ASTM A193, B7); and

c. Two keys (SA-533, Gr. B, Class 1) welded to the upper shell. The clevices and keys are analyzed to ASME B&PV Code, Section III. The pin and hydraulic snubbers are sized so that stresses will not exceed 90% of yield. These items are tested at rated load to confirm their adequacy.

The following design cyclic transients for the operational requirements of the SG were used in the fatigue analyses required by the applicable codes listed in Table 4-9 and in accordance with the load combinations noted in Table 4-8 (References 1 and 2):

a. 500 heatup and cooldown cycles during the vessels 40-year design life at a heating and cooling rate of 100°F/hr between 70°F and 532°F;

b. 15,000 power change cycles over the range of 15% to 100% of full power with ramp load change of 5% of full power per minute increasing and decreasing;

c. 2000 cycles of 10% of full load step power changes, increasing from 90% to 100% of full load and decreasing from 100% to 90% of full load;

d. 10,000 cycles of adding 600 gpm of 70°F feedwater with the plant in hot standby condition;

e. 10^6 cycles of normal primary side variations of ±100 psi and ±6°F at operating temperature and pressure;

f. 10^6 cycles of normal secondary side variations of ±40 psi at operating temperature and pressure;

g. 4,000 cycles of transient pressure differentials of 85 psi across the primary head divider head plate due to starting and stopping the primary coolant pumps;

h. 400 reactor trips from 100% power;

i. 40 cycles of total loss of reactor coolant flow when at 100% power;

j. 50 cycles of loss of turbine load from 100% power without a direct reactor trip;

k. 5 cycles of loss of secondary system pressure;

l. 8 cycles of loss of feedwater flow;

m. 320 cycles of leak testing at 2500 psia and 130°F;

n. 320 cycles of leak testing at 1015 psia and 150°F;

o. 10 cycles of hydrostatic testing the RCS at 3125 psia and 70°F for shop hydrostatic tests, and 150°F for field hydrostatic tests with the secondary side at atmospheric pressure. Any combination of shop or field hydrostatic tests can be used to obtain a total of 10 cycles.

p. 10 cycles of hydrostatic testing of the secondary side at 1269 psia and 70°F for shop hydrostatic tests and 175°F field hydrostatic tests with the primary side at atmospheric pressure. Any combination of shop or field hydrostatic test can be used to obtain a total of 10 cycles.

q. 40 cycles for testing the upper snubber lugs;

r. 100 cycles with SG level lower than top of feedwater header gooseneck for greater than 60 minutes with no feedwater flow; and

s. 200 pre-load cycles for primary manway, secondary manway, handhole, and inspection port covers.

In addition to the normal design transients listed above and those listed in Section 4.1.4.1, the following additional abnormal transient was also considered in arriving at a satisfactory usage factor as defined in ASME B&PV Code, Section III: 8 cycles of adding a maximum of 650 gpm of 32°F feedwater, with the SG secondary side dry and at 600°F. This transient, however, did not form a basis for the code design of the SGs.

The unit is capable of withstanding these conditions for the prescribed numbers of cycles, in addition to the prescribed operating conditions without exceeding the allowable cumulative usage factor as prescribed in ASME B&PV Code, Section III.

More specifically, the SG will be designed such that no damage to the equipment is caused by the frequency ranges of 14-15 cps and 70-75 cps. The lower frequency range is defined as a mechanical vibration induced by the RCP, and the upper frequency range is defined as a sinusoidal pressure variation of ±6 psi in the cold leg piping also induced by the RCP. It has been determined that there will be tube wall margin under the postulated condition of the largest design basis pipe break in the reactor coolant pressure boundary during reactor operation. Tubing design incorporates an acceptable wall thinning up to 44.5 percent. Therefore, providing excess material in the tube wall thickness has accommodated any degradation of tubes that may occur during the service life.

The SG has been designed to ensure that critical vibration frequencies will be well out of the range expected during normal operation and during abnormal conditions. The SG tubing and tubing supports are designed and fabricated with considerations having been given to both secondary-side flow-induced vibrations and RCP-induced vibrations. In addition, the heat transfer tubing and tube supports have been designed such that they shall not be structurally damaged under the loss of secondary pressure conditions that may produce a fluid velocity in the tube bundle four times design velocity.

It has been found that all tubes and tube sections that will experience forcing functions from cross flow and parallel flow have natural frequencies sufficiently different from the frequency of the forcing function that they will not experience damaging vibrations. The mechanical excitation frequency is sufficiently different from the lowest natural frequency for out-of-plane or lateral vibration in any tube span that critical vibration will not occur.

Even though some of the tube sections have a geometry such that they may vibrate near resonance with the 70 to 75 cps hydraulic pulse forcing function, the resulting stresses and deflections are negligible.

Tube denting has occurred on both units at Calvert Cliffs in the original CE SGs to a minor extent. Denting is a phenomenon characteristic to SGs with drilled carbon steel support plates. Volumetric expansion of the support plate material, when converted to magnetite, causes a squeezing action on the tubes. If the support plates are detached from the shroud, a large portion of the stresses are relieved. On original CE SGs 11 and 12, the attachment lugs for support plate numbers 9 and 10 were cut for this purpose. The RSGs do not have drilled plate tube supports, but instead use stainless steel lattice grids and fan bars. Therefore, the denting phenomena is not expected.

Replacement Steam Generator Structural Damping

Damping values are assigned to the RSG internals based on the recommendations of NRC Regulatory Guide (RG) 1.61 and ASME Code Case N-411.

With the exception of the U-tubes, the RSG internals are predominantly an assembly of welded and bolted steel structures. Per RG 1.61, they qualify for 2% to 4% damping for the operating basis earthquake (OBE) and 4% to 7% for the safe shutdown earthquake (SSE). Damping values of 2% for the OBE and 4% for the SSE are used for the internals.

The U-tubes are an assembly of small pipes. For OBE loading, per ASME Code Case N-411, 5% damping is used for frequencies below 10 Hz, 2% damping for frequencies above 20 Hz, and a linear relationship between these damping values is used for intermediate frequencies. The ASME Code Case N-411 damping is also used for the U-bend support assembly, since it is embedded within and supported by the U-tubes, and it is only significantly loaded when the U-tubes are in motion. Testing done at BWC has shown that higher structural damping is appropriate for the U-tubes and U-bend support assembly, as allowed by RG 1.61 when supported by documented tests data. Therefore, for SSE loading, 7% damping is used for the U-tubes and U-bend support assembly.

Steam Generator Tube Inspections

A program of inservice inspection of SG tubes using eddy current techniques is employed. The details regarding techniques, extent of inspections, frequency of inspections, and calibration and acceptance criteria are specified by the Technical Specifications and ASME B&PV Code, Section XI.

The baseline examination of the Unit 1 SG tubes was performed in June 1972. It was performed for the purpose of establishing typical eddy current signatures of new tubes in good condition. Approximately 8% of the tubes in each SG were examined. Thirty-eight tubes (in both generators) were examined over their entire lengths. For the remainder, examinations did not extend beyond the bends. No significant indications were noted.

Subsequently, new "baseline" examinations of the Unit 1 SG tubes were performed in November 1986, and Unit 2 in April 1987. They were performed to establish, based on current technology, eddy current signatures of tube conditions. All tubes in each SG were examined over their entire length. Relevant indications were noted and resolved by tube plugging or data analysis.

With the replacement of the original CE SGs with BWC RSGs, the preceding two paragraphs present historical information only. The baseline for the BWC RSGs was established by performing a 100% bobbin-coil preservice inspection for the SG tubes. In addition, rotating probe coil examinations were performed for the following:

a. dings, dents, manufacturing burnish marks, and anomalous signals over 5 Volts;
b. First three rows of U-bends above the upper support grid;
c. Two smallest radius rows of non-heat treated U-bends; and
d. 100% top of tubesheet (±3 inches of secondary side face of tubesheets).

Steam Generator Pressure/Temperature Limitation

The SG pressure and temperature are limited to ensure that the pressure induced stresses in the SGs do not exceed the maximum allowable fracture toughness stress limits. The temperatures of both the primary and secondary coolants in the SGs are maintained above 80°F for Unit 1 and Unit 2 (Reference 41) when the pressure of either coolant in the SG is greater than 200 psig. These limitations are based on SG secondary side limitations and are sufficient to prevent brittle fracture.

4.1.3.3 Reactor Coolant Pumps

The reactor coolant is circulated by four vertical, single-suction, centrifugal-type pumps (Figure 4-4). The suction nozzles are in the bottom vertical position. The pressure containing components are designed and fabricated in accordance with ASME B&PV Code, Section III, Class A.

The pump impeller is keyed and locked to the shaft. A close clearance thermal barrier assembly is mounted above the water-lubricated bearing to retard heat flow from the pump to the seal cavity which is located above the thermal barrier. The thermal barrier assembly also tends to isolate the hot fluid in the pump from the cooler fluid above, and, in the event of a seal failure, serves as an additional barrier to reduce leakage from the pump. Each pump is equipped with replaceable casing wear rings. A water-lubricated bearing is located in the fluid between the impeller and thermal barrier to provide shaft support. Additional shaft support is provided by bearings in the electric motor, which is directly connected to the pump shaft by a rigid coupling.

The shaft seal assembly, located above the thermal barrier, consists of four face-type mechanical seals, three full-pressure seals mounted in tandem and a fourth low-pressure backup vapor seal designed to withstand operating system pressure with the pumps operating or stopped (Figures 4-4, 4-6, and 4-6A). The performance of the shaft seal system is monitored by pressure and temperature sensing devices in the seal system. A controlled bleedoff flow through the pump seals is maintained to cool the seals and to equalize the pressure drop across each seal. The controlled bleedoff flow is collected and processed by the CVCS. Any leakage past the vapor seal (the last mechanical seal) is piped to containment trench at Elevation 10'0". The seals are cooled by circulating the controlled leakage through a heat exchanger mounted integrally within the pump cover assembly; no damage would result in the event of pump operation without cooling water for at least five minutes. To reduce plant downtime and personnel exposure to radiation during seal maintenance, the seal system is contained in a cartridge which can be removed and replaced as a unit. The face seals can be replaced without draining the pump casing. The seal detail is shown in Figure 4-6.

The original Byron Jackson SU seals have been replaced with a state-of-the-art design under FCR 87-0074. The new seals are designed in accordance with ASME B&PV Code, Section III, 1983 Edition with Summer 1983 Addenda, and manufactured by Sulzer Bingham Pumps. The model (RCR875B-3V) incorporates several design features in addition to those listed above. These features include:

Flexible Stator - Accommodates large shaft tilt without disturbing the sealing faces or fluid film.

Rotating Support Ring - Isolates the sealing ring from adjacent metallic parts through the use of a narrow support nose and three nonsealing O-rings. This feature protects the sealing faces and fluid film from the effects of large temperature and pressure transients.

Material Selection - All materials were carefully chosen based on their properties for strength, corrosion, temperature and radiation resistance, chemical content and differential thermal expansion.

A motor-mounted flywheel reduces the rate of flow decay upon loss of pump power. The combined inertia of the pump motor and flywheel is 100,000 lbm/ft^2. Flow coastdown characteristics are discussed in Section 14.9.

The pump motor assembly includes motor bearing oil coolers, seal chamber, controls and instruments. Cooling water is provided from the Component Cooling System. A mechanism is provided on each pump to prevent reverse rotation. This mechanism is a circular rack with sprags.

The design parameters for the RCPs are given in Table 4-5. The CE report of November 1973 on RCP flow measurement is found in Appendix A at the end of this chapter.

The RCP and motor are supported by four support lugs welded to the volute. The pump is supported on spring hangers employed between the support lugs and the floor below. The lugs are analyzed to ASME B&PV Code, Section III. Movement in the horizontal plane to compensate for pipe thermal growth and contraction is permitted. Vertical movement is not restrained.

The pump is constructed of high alloy cast stainless steel parts to minimize corrosion. The mechanical seals consist of a rotating tungsten carbide ring riding over a carbon graphite stationary face. The design life of this seal arrangement is at least four years. Each seal is designed to accept a pressure drop equal to full operating system pressure, but normally operates at one-third this pressure drop.

The expected pump performance curve is shown in Figure 4-7. The air-cooled, self-ventilated pump motor is sized for continuous operation at flows resulting from four-pump operation or partial pump operation with 0.74 specific gravity water. The motor service factor is sufficient to allow continuous operation with 1.0 specific gravity water. The motors are designed to start and accelerate to speed under full load when 80% or more of their normal voltage is applied. The motors are contained within standard drip-proof enclosures and are equipped with electrical insulation suitable for a 0% to 100% humidity and radiation environment of 30 R/hr.

4.1.3.3.1 Reactor Coolant Pump Flywheel

The design requirements of the RCPs include a minimum inertia for the rotating assembly of 100,000 lbm/ft^2; to achieve this total, a flywheel with an inertia of 70,000 lbm/ft^2 has been incorporated.

The flywheel assembly consists of two discs bolted together and keyed to the shaft above the motor. The dimensions of the discs are:

	Upper Disc	Lower Disc
Outside Diameter, in.	75	65
Thickness, in.	8	5
Weight, ea, lb	10,123	4,750

The selection of material, machining and manufacturing operations, quality control and the rigorous acceptance criteria established to assure the integrity of the flywheel and to minimize operating stresses include the following:

a) The A533 vacuum-degassed flywheel material normally shows a 15 ft/lb longitudinal V-notch transition temperature of 40°F; the NDTT of the flywheel material is at least 30°F below the minimum motor operating temperature.

b) In flame cutting of the bore, at least l/2" of stock was left on the radius for machining;

c) There are no stress concentrations such as stencil or punch marks, or drilled or tapped holes within 8" of the edge of the flywheel bore;

d) Each flywheel plate was ultrasonically inspected to ASME B&PV Code, Section III, Paragraph N-322;

e) After balancing, the flywheel and motor assembly were tested at 120% of design speed. The maximum allowable vibration for acceptance of the assembly was 1.5 mils; and

f) A keyway fillet radius not less than 1/8" minimizes stress concentration factors.

An analysis to determine the rotational velocity which would cause flywheel failure if a crack at the flywheel bore were present indicates the following:

a) No failure would occur at the design speed;

b) At 100% overspeed a crack 15" on each side of the bore would be required; and

c) At 150% overspeed a crack 6" on each side of the bore would be required.

To assure that no deleterious effects occur after the flywheels are placed in service, the flywheels will be inspected periodically as part of the inservice inspection program.

The NDTT of the flywheel material is less than +10°F. The data for the eight heats used in the fabrication of the RCP flywheels is tabulated below:

Heat No.		V-notch at +10°F		
C7689	T	53	59	55
	L	115	110	104
C7174	T	53	70	61
	L	110	105	97
A6678	T	52	41	57
	L	86	57	103

Heat No.		V-notch at +70°F		
B3616	T	128	116	112
	L	114	118	116
D8392	T	110	112	110
	L	140	146	146

Heat No.		V-notch at +10°F		
A6735	T	27	43	52
	L	53	51	50
C7434	T	65	65	50
	L	78	81	116
C9220	T	51	32	43
	L	92	88	94

The Charpy V-notch upper shelf energy level was not specified. The data contained above is acceptable.

The fracture toughness of the material at 75°F is 140 ksi/in$^{1/2}$.

Flywheel material has been subjected to a 100% volumetric ultrasonic inspection from the flat surface, per ASME B&PV Code, Section III, Paragraph N-322. The acceptance standard is as follows:

a. Indication greater than 4" diameter is cause for rejection per ASTM A435; and

b. Indications of complete loss of back reflection greater than one crystal diameter and less than 4" diameter shall be tested by angle beam technique. Indications greater than a 3% notch is cause for rejection.

The flywheel has been subjected to a magnetic-particle or liquid-penetrant examination in accordance with ASME B&PV Code, Section III, Paragraph N-322 before assembly of flywheel plates. This inspection was done on both plate surfaces to a distance of 8" minimum beyond the final bore diameter and after machining in the bore. The acceptance standard for the inspection is as follows:

a. Indications of cracks and linear defects are subject to rejection. A linear defect is one in which the length is three times the width. The minimum length of defects to be considered linear shall be 3/16"; and

b. The principal stress is to be not greater than 50% of the yield point (based on transverse test specimens taken at 1/4t) of the flywheel material at normal operating speed, not considering keyway stress concentration factors.

The pump assembly is designed to be capable of operating up to 125% of normal operating speed for periods not exceeding 10 seconds in accordance with NEMA standards. In addition, each motor and flywheel is tested at 120% overspeed.

The calculated combined primary stresses in the flywheel at the normal operating speed is 7320 psi at 900 rpm. The interference fit of the flywheel in this assembly is negligible.

During normal plant operation, the RCPs are powered by station-generated power. Since the motor speed is directly proportional to the frequency of the power source, the maximum anticipated pump speed is 110% of nominal, which is equal to the overspeed trip setpoint of the turbine generator.

The estimated rotational speed that the flywheel would attain during a rupture in the discharge piping of the RCP is about 152% of rated speed (888 rpm) or 1350 rpm. A break in the suction piping causes the reactor coolant to flow through the pump opposite to the normal direction of flow decelerating the rotating assembly until it is brought to rest against the anti-reverse rotation device. The model used is described in CENPD-2, Appendix I, a proprietary report entitled, "Combustion Engineering Analytical Techniques for Evaluating Loss of Coolant Accidents."

Previous studies of the rotating limitations of RCPs have indicated that the first characteristic which will prevent excessive operating speeds is binding of the motor against the stator.

This limitation does not become effective for the Calvert Cliffs RCP motors during any mode of operation, including the worst-case pipe rupture, since it is estimated that approximately twice the speed calculated following a loss-of-coolant accident (LOCA) is required to produce rotor binding.

The RCPs are typical centrifugal volumetric flow machines. The pump response following a LOCA is predicted using generally accepted methods as described in CENPD-26. A spectrum of breaks in the RCP discharge line have been analyzed and the results follow a predictable pattern. Assuming loss of electrical power to the pump at the start of the LOCA, it is seen that the pumps initially lose speed because the volumetric flow through the pump is not sufficient to sustain the nominal speed of rotation. The volumetric flow increases during the transient, accelerating the pump to its maximum speed. The extent of the initial loss of speed varies with the break size. The larger the break size, the less the initial deceleration and the higher the maximum speed attained. As previously stated, for the double-ended discharge break, the maximum speed the pump attains is 152% of normal operating speed. The calculated torque imposed on the impeller follows the same trend as the speed, with the maximum value occurring following the double-ended discharge break.

The need for a disengaging device to prevent motor overspeed has been evaluated. In view of the fact that the maximum anticipated pump speed is well within the safe operating limits for all rotating parts, a means to disengage the motor from the pump is not necessary.

Analysis of the LOCA leading to pump overspeed incidents indicates no probability of significant structural damage to the pumps.

The calculational model used to obtain the bursting speeds is described as follows:

Outer radius of flywheel, r = 37.5 in.

Density of A 533 steel $\rho = 0.283/386$ lb sec^2/in^4

Angular velocity $w = 2\ \pi N/60$ rad/sec

Poisson ratio $\mu = 0.3$

The maximum stress, S, in a solid disk of uniform thickness (Timoshenko, Theory of Elasticity, p. 71)

$$S = \frac{3 + \mu}{8} \rho w 2_r 2$$

S = 0.00465N2

If the entire plate was in a uniform stress field equal to this maximum value, the stress would be related to the toughness, K_{IC}, by the approximate formula,

$$K_{IC} = S\sqrt{\pi C}$$

where C is the distance from the crack tip to the center of the disk. If a conservative value of 100,000 psi $\sqrt{in}$ is assumed for the room temperature fracture toughness for this steel, a relationship between C and N can be determined.

$$N = \left(\frac{K_{IC}}{\sqrt{\pi C}\ (0.00465)}\right)^{1/2}$$

$N = 3500\ C^{-1/4}$

4.1.3.3.2 Reactor Coolant Pump Oil Collection System

Each Calvert Cliffs RCS has four RCPs, with an oil collection system which drains to two vented closed collection tanks (two RCPs per tank). The capacity of each oil collection tank is 275 gallons. The RCP motors have an upper bearing oil pot capacity of 200 gallons and a lower guide bearing oil pot capacity of 25 gallons. The total lube oil inventory of 225 gallons for one RCP is accommodated by the capacity of the oil collection tank.

The RCPs are separated by a horizontal distance of 25 feet and are seismically supported. It is very unlikely that a seismic event would result in the simultaneous failure of both of the RCP lube oil systems being served by the same oil collection tank.

An oil spillage protection system has been provided for each RCP motor. The system consists of encapsulating devices installed around all potential leakage points. Typical motor components encapsulated are:

lube oil lift pump, lube oil cooler, connecting flanges, and oil reservoir drainage points. Piping from the encapsulations accommodate a major oil leak and are interconnected to a common drain leading to an oil collection tank located at Containment elevation 10'-0". Drain lines are sized and arranged to accommodate the maximum leak.

The components of the oil collection system, except for the collection tanks, have been designed so that they are capable of withstanding an SSE. The oil collection tanks, located on the containment floor, are not ASME Code qualified, and are not seismically qualified. They are, however, supported and restrained to prevent movement during an SSE.

The consequences of a simultaneous release of all the oil from two RCP motors would not be significant from a fire standpoint. Ignition of the oil is very unlikely, particularly with the encapsulation system in place, since the oil would remain within the collection system and would not come in contact with hot coolant piping, hangers or equipment. If the oil collection tank were to overflow, the oil would flow from the tank vent line down to the floor, then eventually flow to the containment sump. The lubricating oil used in the RCP motors has a flash point greater than 400°F and there are no ignition sources at the floor level of the lower Containment.

All lube oil collection tank vents are equipped with flame arrestors.

A procedure (OP-6) contains steps to demonstrate that the oil collection systems remain functional, as follows:

a. At each refueling outage a visual examination of the encapsulation devices, drain piping, and oil collection tanks is performed prior to startup.
b. Prior to startup, the oil collection tank level is checked routinely in accordance with the procedure.

The oil collection system is required by National Fire Protection Association (NFPA)-805, Section 3.3.12, Reactor Coolant Pumps.

Compliance with NFPA-805 is addressed as an approved engineering equivalency evaluation under the NFPA- 805 fire protection program (see Section 9.9).

4.1.3.4 Reactor Coolant Piping

The RCS piping consists of two loops which connect the SGs to the reactor vessel. Each loop can be considered to consist of a 42" ID "hot leg" pipe connecting the reactor vessel outlet to the SG inlet, and 30" ID piping connecting the SG outlets to the RCPs and the coolant pumps to the reactor vessel inlets. A 12" schedule 160 surge line connects one loop hot leg to the pressurizer. Design parameters for the reactor coolant piping are given in Table 4-6.

The reactor coolant piping was designed and fabricated in accordance with the rules and procedures of ANSI B31.7, Class I. The anticipated transients listed in Section 4.1.1 form the basis for the required fatigue analysis to ensure an adequate usage factor.

The reactor coolant piping is fabricated from SA516, GR70 carbon steel mill clad internally with type 304L stainless steel. A minimum clad thickness of 1/8" is maintained. The 12" surge line is fabricated from ASTM A351, Gr CF8M alloy steel.

Thermal sleeves are installed in the surge nozzle, charging nozzles and shutdown cooling inlet nozzle to reduce thermal shock effects from auxiliary systems. Clad sections of piping were fitted, where necessary, with safe ends for field welding to stainless steel components.

The piping was shop fabricated and shop welded into subassemblies to the greatest extent practicable to minimize the amount of field welding. Fabrication of piping and subassemblies was done by shop personnel experienced in making large heavy wall welds. Welding procedures and operations met the requirements of ASME B&PV Code, Section IX. All welds were 100% radiographed and liquid-penetrant tested and all reactor coolant piping penetrations were attached in accordance with the requirements of ANSI B31.7. Cleanliness standards consistent with nuclear service were maintained during fabrication and erection.

4.1.3.5 Pressurizer

The pressurizer maintains RCS operating pressure and compensates for changes in coolant volume during load changes. Table 4-7 gives design parameters for the pressurizer. The pressurizer is shown in Figure 4-8.

Pressure is maintained by controlling the temperature of the saturated liquid volume in the pressurizer. At full load nominal conditions, slightly more than one-half the pressurizer volume is occupied by saturated water, and the remainder by saturated steam. A number of pressurizer heaters are operated continuously to offset the heat losses and the continuous minimum spray, thereby maintaining the steam and water in thermal equilibrium at the saturation temperature corresponding to the desired system pressure.

During load changes, the pressurizer limits pressure variations caused by expansion or contraction of the reactor coolant. The average reactor coolant temperature is programmed to vary as a function of load as shown in Figure 4-9. A reduction in load is followed by a decrease in the average reactor coolant temperature to the programmed value for the lower power level. The resulting contraction of the coolant lowers the pressurizer water level causing the reactor system pressure to decrease. This pressure reduction is partially compensated by flashing of pressurizer water into steam. All pressurizer heaters are automatically energized on low system pressure, generating steam and further limiting pressure decrease. Should the water level in the pressurizer drop sufficiently below its setpoint, the letdown control valves close to a minimum value and additional charging pumps in the CVCS are automatically started to add coolant to the system and restore pressurizer level.

When steam demand is increased, the average reactor coolant temperature is raised in accordance with the coolant temperature program (Figure 4-9). The expanding coolant from the reactor coolant piping hot leg enters the bottom of the pressurizer (in-surge), compressing the steam and raising system pressure. The increase in pressure is moderated by the condensation of steam during compression and by the decrease in bulk temperature in the liquid phase. Should the pressure increase be large enough, the pressurizer spray valves open, spraying coolant from the RCP discharge (cold leg) into the pressurizer steam space. The relatively cold spray water condenses some of the steam in the steam space, limiting the system pressure increase. The programmed pressurizer water level is a power-dependent function. A high level error signal produced by an in-surge causes the letdown control valves to open, releasing coolant to the CVCS and restoring the pressurizer to the prescribed level.

Small pressure and coolant volume variations are accommodated by the steam volume which absorbs flow into the pressurizer, and by the water volume which allows flow out of the pressurizer. The total volume of the pressurizer is determined by consideration of the following factors:

a. Sufficient water volume is necessary to prevent draining the pressurizer as the result of a reactor trip or a loss-of-load incident. In order to preclude the initiation of safety injection and of automatic injection of concentrated boric acid by the charging pumps, the pressurizer is designed so that the minimum pressure observed during such transients is above the setpoint of the safety injection actuation signal;
b. The heaters should not be uncovered by the out-surge following load decreases; 10% step decrease, and 5% per minute ramp decrease;
c. The steam volume should be sufficient to yield acceptable pressure response to normal system volume changes during load change transients;
d. The water volume should be minimized to reduce the energy release and resultant containment pressure during a LOCA;
e. The steam volume should be sufficient to accept the reactor coolant in-surge resulting from loss of load or loss of feedwater without the water level reaching the safety and PORV nozzles; and
f. During load following transients, the total coolant volume change and associated charging and letdown flows should be kept as small as practical and be compatible with the capacities of the volume control tank, charging pumps, and letdown control valves in the CVCS.

The following design cyclic transients, which include conservative estimates of the operational requirements for the components discussed in Section 4.1.3, were used in the fatigue analyses required by the applicable codes listed in Table 4-9, and in accordance with the load combinations noted in Table 4-8:

a. 500 heatup and 500 cooldown[a] cycles during the system's 40-year design life at a heating rate of 100°F/hr and a cooling rate of 200°F/hr between the temperature limits of 70°F and 653°F;
b. 15,000 power change cycles (increasing load) over the range of 15% to 100% of full load with a ramp load change of 5% of full load per minute and 15,000 power change cycles (decreasing load) over the range of 100% to 15% of full load with a ramp load change of 5% of full load per minute;
c. 2,000 cycles of 10% of full load step power changes (increasing load) from 10% to 90% of full power, and 2,000 cycles of 10% of full load step power changes (decreasing load) from 100% to 20% of full load;
d. 10^6 cycles of normal variations[b] of ± 100 psi and ± 7°F;
e. 400 reactor trips from 100% power;
f. 320 cycles of leak testing[c] at 2500 psia and a temperature between 100°F and 400°F;
g. 10 cycles of hydrostatic testing the RCS at 3125 psia and a temperature between 100°F and 400°F;
h. 40 cycles of total loss of reactor coolant flow when at 100% power[d];
i. 40 cycles of loss of turbine load from 100% power with a direct reactor trip[d]; and
j. 5 cycles of loss of secondary system pressure[d].

NOTES:

[a] Cooldowns may include the introduction of 40°F spray water at the rate of 132 gpm with the temperature of the pressurizer and spray nozzle at 443°F and the pressure at 395 psia.

[b] During normal pressure and temperature variations, as the pressure increases above 2300 psia, the spray nozzle will be subjected to an instantaneous increase in spray flow from 1.5 gpm to 375 gpm, with the fluid temperature increasing from 500°F to 550°F in 7 seconds. Fluid temperature remains at 550°F and fluid flow at 375 gpm for 5 minutes. Then the flow instantaneously drops to 1.5 gpm and the fluid temperature decreases to 500°F at the rate of 2°F/minute.

[c] Performed in conjunction with heatup.

[d] These transients specify abnormal conditions. Their effect should be evaluated in conjunction with the other transients specified. The abnormal conditions do not form the basis for Code Design of the Vessel.

To account for these factors and to provide adequate margin at all power levels, the water level in the pressurizer is programmed as a function of average coolant temperature. A representation of this relationship is shown in Figure 4-10. Actual response is determined by the calibration of the reactor regulating system. High or low water level error signals result in the control actions shown in Figure 4-11 and described above.

The pressurizer heaters are single-unit, direct-immersion heaters which protrude vertically into the pressurizer through sleeves welded in the lower head. Each heater is internally restrained from high amplitude vibrations and can be individually removed for maintenance during plant shutdown. Approximately 20% of the heaters are connected to proportional controllers which adjust the heat input as required to account for steady state losses and to maintain the desired steam pressure in the pressurizer. If pressure falls below the proportional band, all of the backup heaters are energized.

The remaining backup heaters are connected to on-off controllers. These heaters are normally deenergized but are turned on by a low pressurizer pressure signal or high level error signal. This latter feature is provided since load increases result in an insurge of relatively cold coolant into the pressurizer, decreasing the temperature of the water volume. The action of the CVCS in restoring the level results in a pressure undershoot below the desired operating pressure. To minimize the pressure undershoot, the backup heaters are energized earlier in the transient, contributing more heat to the water before the low pressure setting is reached. A low-low pressurizer level signal deenergizes all heaters to prevent heater burnout.

The pressurizer spray is supplied from each of the RCP discharges on one loop to the pressurizer spray nozzle. Automatic spray control valves control the amount of spray as a function of pressurizer pressure; both of the spray control valves function in response to the signal from the controller. These components are sized to use the differential pressure between the pump discharge and the pressurizer to pass the amount of spray required to prevent the pressurizer steam pressure from opening the PORVs during normal load following transients. A small continuous flow is maintained through the spray lines at all times to keep the spray lines and the surge line warm, reducing thermal shock during plant transients. This continuous flow also aids in keeping the chemistry and boric acid concentration of the pressurizer water equal to that of the coolant in the heat transfer loops.

The pressurizer spray system is designed for the transients listed for the pressurizer. In addition, operating experience has discovered the existence of thermal stratification in the pressurizer spray piping under various plant conditions. In particular, during plant cooldowns with the initiation of auxiliary spray, the pressurizer spray piping can see a large number of thermal shock transient cycles. See Reference 38 for further details on the conditions and allowed number of occurrences of these and other transients that affect the pressurizer spray piping.

An auxiliary spray line is provided from the charging pumps to permit pressurizer spray during plant heatup, or to allow cooling if the RCPs are shut down. The pressurizer specification requires the design to be suitable for 500 cycles of auxiliary spray at 40°F and 395 psia. These conditions represent a conservative maximum for analytical purposes. The results of the design analysis indicate an overall usage factor of less than 0.5 for the spray connections. The number of auxiliary spray occurrences during the lifetime of the plant is expected to be much less than 500. However, it is planned to record each operation of auxiliary spray during plant cooldown and review the total periodically. If the total approaches 500 during the plant lifetime, a detailed review of the fatigue history of the spray connections will be conducted and appropriate action taken.

The auxiliary spray system is designed for the transients listed for the pressurizer. In addition, the auxiliary spray system fatigue analysis considers other transients, including actuations of auxiliary spray at various RCS temperatures and plant operating modes. See Reference 39 for further details, including the allowed number of auxiliary spray actuations and the assumed operating conditions of these transients.

In the event of an abnormal transient which causes a sustained increase in pressurizer pressure, at a rate exceeding the control capacity of the spray, a high pressure trip level will be reached. This signal trips the reactor and opens the two PORVs. The steam discharged by the relief valves is piped to the quench tank where it is condensed. In accordance with ASME B&PV Code, Section III, the RCS is protected from overpressure by two spring-loaded safety valves. The discharge from the safety valves is also piped to the quench tank.

The pressurizer is supported by a cylindrical skirt (SA 516, Gr70) welded to the lower head. The skirt is analyzed to ASME B&PV Code, Section III. Since the pressurizer surge line has sufficient flexibility, no provisions are made for horizontal movement and the skirt is bolted rigidly to the floor.

The pressurizer was designed and fabricated in accordance with ASME B&PV Code, Section III, Class A. The pressurizer is constructed of A-533, Grade B, Class 1 steel plate. The interior surface of the cylindrical shell and upper head is clad with weld deposited stainless steel. The lower head is clad with a Ni-Cr-Fe alloy to facilitate welding of the Ni-Cr-Fe alloy heater sleeves to the shell. Safe ends were provided on the pressurizer nozzles as required to facilitate field welds to the connecting piping.

In May 1989, leakage caused by primary water stress corrosion cracking (PWSCC) was discovered in the Ni-Cr-Fe Alloy 600 heater sleeves of the Unit 2 pressurizer. In 1990, modifications were completed on all of the 120 Calvert Cliffs Unit 2 pressurizer heater sleeves. One sleeve location was bored out for testing and plugged with a Ni-Cr-Fe Alloy 690 plug in a Ni-Cr-Fe Alloy 690 outer sleeve. Location N-3 was plugged in 2011. The remaining 118 heater sleeve locations were replaced with a dual sleeve (inner and outer) design also using Alloy 690 material. The replacement outer sleeves were welded to the original Ni-Cr-Fe Alloy clad and welded to Ni-Cr-Fe Alloy weld pad buildups made on the outside of the pressurizer lower head. Inner sleeves were installed through the outer sleeves and welded to the outer sleeve to return the heater geometry to its original configuration.

The PWSCC-susceptible portions of the heater sleeves of Unit 1 were nickel-plated in 1994. The decision to do this was based on the PWSCC damage which occurred on the Calvert Cliffs Unit 2 pressurizer heater sleeves and other PWSCC-related Ni-Cr-Fe Alloy 600 failures at other plants. Plating was performed to eliminate exposure of the Ni-Cr-Fe heater sleeve Alloy 600 material to the primary reactor coolant, thus eliminating PWSCC occurrence. The nickel-plating process has been used successfully to reduce PWSCC failures on SG tubes in nuclear power plants in Europe.

Also in 1994, leakage due to PWSCC damage occurred on Unit 1 pressurizer heater sleeves at penetrations B-3 and FF-1. The lower portions of the sleeves were removed from the penetrations and Alloy 690 plugs were installed in their place. The plugs were attached by welding them to the pressurizer lower head outside surface using the SMAW half-bead technique.

Based on leakage which occurred in Unit 1 pressurizer heater at location CC-1, the heater was removed from service and a stainless steel plug was seal-welded at the bottom of the existing heater sleeve. Location B-1 was plugged in 1998.

In 2012 117 out of 120 Unit 1 pressurizer heater penetrations were modified. Three heater sleeve penetrations, FF-1, B-3, and B-1, are plugged and not modified. Location CC-1 was also modified and a new plug installed. Location V-1 was modified and a plug has been installed in place of a heater. The two lower level instrument nozzles were also modified. The heater modification utilizes the half-nozzle approach to modify the pressurizer heater sleeves. A portion of the existing heater sleeve is removed, and a new stainless steel lower sleeve inserted into the penetration. The ambient temperature temper bead (IDTB) welding technique, using the gas tungsten arc welding process with stainless steel filler metal attached the new sleeve to the bore ID, establishing a new pressure boundary weld. The IDTB weld is disassociated from the original heater sleeve. A heater stabilizer insert (or plug at location CC-1 and V-1) was then welded to the lower end of each sleeve.

For the two instrument nozzles located in the bottom head of the pressurizer, the IDTB welding process was used to install the replacement stainless steel nozzles. A socket adaptor was then welded to the lower end of each replacement nozzle.

In 2014 Unit 1 pressurizer heaters L3 and BB3 were electrically disconnected due to a grounding issue.

In 2016 Unit 1 pressurizer heater N4 was electrically disconnected due to a grounding issue.

4.1.3.6 Reactor Coolant Vent System

Vents were added to the reactor vessel and to the pressurizer head in response to the TMI Lessons Learned Report, NUREG 0737, Item II.B.1. These vents are intended to provide a means of releasing non-condensable gases from the RCS during natural circulation. The pressurizer vent line valves are used as a backup to main and auxiliary spray to depressurize the RCS during a SG tube rupture. The original design of the Calvert Cliffs plant allowed venting of the RCS only during cold shutdown. The vent modifications provide electrically-operated solenoid valves, powered from emergency busses, that are operated from the Control Room. The reactor vessel and the pressurizer each have two of these valves in series, which fail closed (power-to-open). The reactor vessel vent line valves are installed in previously existing lines; the pressurizer vent line valves are installed in a line that was added as an additional branch off the pressurizer vapor sample line. The two vent lines join to a common line that leads to the quench tank. The common line contains a temperature element and alarm which is used for valve seat leak detection and flow indication. Detailed information on the solenoid vent valves is provided in Table 4-9A.

4.1.4 DESIGN EVALUATION

4.1.4.1 Codes

The codes adhered to and component classifications are listed in Table 4-9.

The impact properties of all materials which form a part of the pressure boundary meet the requirements of ASME B&PV Code, Section III, Paragraph N-330, at a temperature of 40°F or less. The replacement RVCH complies with 10 CFR Part 50, Appendix G, Fracture Toughness Requirements, and ASME B&PV Code, Section III, Paragraph NB-2300.

All cast valves and components which form the RCS boundary (Class I) were measured in the factory or at the field prior to plant operation to assure that their wall thickness was equal to or greater than the manufacturer’s calculated minimum wall thickness.

4.1.4.2 Materials Compatibility

4.1.4.2.1 Materials Exposed To Coolant

Materials exposed to reactor coolant have shown satisfactory performance in operating reactor plants. A listing of materials is given in Table 4-10.

4.1.4.2.2 Insulation

Piping and equipment are insulated with a mass-type and reflective-type material compatible with the temperature and functions involved.

A removable metal reflective-type thermal insulation is provided on the flange stud area of the reactor vessel closure head to permit access to the head studs for removal and reinstallation of the head. Removable insulation is provided on weld areas of the RCS subject to inservice inspection. Nonremovable (reflective type) metal thermal insulation is provided on the reactor cavity wall.

The thickness of insulation is such that the exterior surface temperature is not higher than approximately 55°F above the maximum containment ambient (120°F). All insulation support attachments are attached prior to final stress relief.

4.1.4.2.3 Coolant Chemistry

Reactor coolant chemistry is maintained within the limits of the reactor coolant chemistry program which is based on the guidance provided by the reactor vendor and the Electric Power Research Institute. Control is accomplished by limiting the input of potentially deleterious materials to the reactor coolant and by operation of the CVCS. High purity demineralized water is used for makeup and strict controls are placed on the purity of chemical additives and ion exchange resins. The CVCS can be used to further purify the reactor coolant. Periodic sampling and analysis of the reactor coolant is performed to ensure that positive control of the reactor coolant is maintained. Approved surveillance and specification procedures require routine monitoring and control of parameters that affect RCS chemistry (i.e., chloride, fluoride, sulfate, lithium, and dissolved hydrogen).

Chloride:

Chloride, in combination with oxygen and high temperature, induces stress corrosion cracking (SCC) of austenitic stainless steels. Therefore, it is important to maintain RCS chloride concentrations below the threshold values of SCC.

Fluoride:

The fluoride concentration in the reactor coolant system is limited to prevent fluoride attack of Zircaloy-4 in non-boiling systems and prevent SCC of austenitic stainless steels.

Sulfate:

The sulfate concentration in the RCS is monitored as an indicator of water purity. Additionally, reduced-sulfur bearing species have been implicated in SCC of austenitic stainless steels and in primary side cracking of Alloy 600 during cold shutdown conditions. Sulfate is the fully oxidized form of the aggressive species. Therefore, the sulfate concentration in the RCS is limited.

Lithium:

The lithium concentration in the RCS is monitored and controlled. Lithium directly effects primary coolant pH. Primary coolant pH is controlled to maintain the integrity of the RCS and minimize out-of-core radiation levels.

Dissolved Hydrogen:

Dissolved hydrogen is added to the RCS to scavenge oxygen and suppress the radiolysis of water during power operations. Hydrazine may be used to scavenge oxygen when there is insufficient gamma flux and the RCS temperature is below 250°F.

4.1.4.3 Welding Procedures

Sensitization of stainless steel occurs when unstabilized 300 Series stainless material is held in the temperature range of 900-1400°F for sufficient time to form a continuous network of chromium carbide precipitates. Sensitization occurs after approximately 100 hours at 900°F, as compared to one hour at 1400°F. Stabilized

300 Series stainless material avoids continuity of chromium carbide precipitates in the grain boundaries by careful control of metal chemistry.

No furnace sensitized stainless steels are employed in the RCS pressure boundary. Sensitization is precluded from NSSSs through materials selection and control of all welding and heat treating procedures.

Major portions of the RCS boundary in CEs nuclear plants are formed by carbon steels and a high nickel base alloy. None of these materials is susceptible to furnace sensitization (a continuous network of iron-chromium grain boundary carbides) in the sense of unstabilized 300 Series stainless steels. All internal carbon steel surfaces are weld-deposit or roll-on clad with Inconel or stainless steel, to preclude excessive corrosion product release.

Internal surfaces of the reactor vessel pressurizer and SG primary head are overlaid with 308 weld deposited metal (309L/308L cladding for replacement RVCH). Weld metal composition is carefully controlled to overcome interface dilution and promote an austeno-ferritic duplex structure. Therefore, during the stress relief heat treatment (1150°F ± 25°F) required by the ASME code for the pressure vessel, a continuous network of chromium carbide precipitates is not formed in the 308 weld overlay even though this material has been subjected to a furnace heat treatment. The delta ferrite acts as a carbon sink and prevents continuity of carbide precipitates.

The primary head of the RSGs is clad with Stainless Steel Type 308L and 309L weld-deposited metal. Use of these materials is in compliance with NRC RG 1.44, "Control of the Use of Stainless Steels." Since the ferrite content of Stainless Steel Type 308L and 309L weld-deposited metal is 5% or more (RSG specification is 5-15%), these materials are exempt from corrosion testing (ASTM A262 Pr. A/E) for verification of freedom from intergranular attack after exposure to sensitizing temperatures in the range of 800-1500°F. Further, this level of ferrite content allows for preferential carbide precipitation at the ferrite-austenite interfaces, which in turn prevents carbide precipitation leading to sensitization of the austenite grain boundaries. In addition to the benefit of ferrite in preventing intergranular attack, the stainless steel cladding utilized is low carbon "L" grade (0.03% max [C]), which will reduce carbide precipitation, and therefore susceptibility to intergranular corrosion.

Extensive testing has confirmed that, properly formulated (a duplex structure), 308 weld deposited metal does not form a continuous carbide network within grain boundaries even following a typical vessel post weld heat treatment (viz, 1150°F for 20 hours). Hence, the material is immune to intergranular corrosion.

All other type 300 Series stainless steel used either is not subjected to a furnace sensitization heat treatment or, as is the case of cladding on the primary piping, is of type 304L (low carbon) composition and is not susceptible to the formation of continuous chromium carbide grain boundary networks.

The primary system coolant pump casing is CF8M (Cast 316) which is a duplex material. The casting is solution annealed after welding; hence, this component will not have a sensitized structure.

Because carbon steel piping is used in the RCS, carbon steel safe ends are required on the reactor vessel and SG large nozzles. Where small diameter solid stainless pipes are employed (or in the instance of welding the coolant pump casing to carbon steel), an Inconel-18 weld deposit is built up on the nozzle prior to vessel post-weld heat treatment. Thereafter, an annealed stainless steel safe end is shop-welded to the Inconel-182 buildup using 182 filler metal.

In joining small diameter annealed solid stainless steel piping, as is used in the pressurizer surge line, charging pump lines and safety injection systems, some carbide precipitation will occur as a result of welding. However, the precipitation that occurs in the weld heat affected zone (HAZ) does not sensitize the material in the context of forming continuous grain boundary carbide precipitates. Typical samples from such welds pass the industry accepted standard for intergranular corrosion susceptibility (i.e., Strauss Test - ASTM A393). Metallographic examination of such welds reveals that only discontinuous grain boundary precipitates are present.

The following four welding processes are used to weld stainless steel in CEs NSSSs. Welding processes are performed in accordance with written procedures, as provided in the Quality Assurance Topical Report. Nitrogen will not be used in lieu of argon or helium gas as a purge gas in the welding process.

- Shielded Metal Arc (SMA)
- Gas Tungsten Arc (GTA)
- Gas Metal Arc (GMA)
- Submerged Arc (SA)

a. Shielded metal arc (SMA) is a process wherein coalescence is produced by heating with an electric arc drawn between a flux covered metal electrode and the work.

b. In the GTA, coalescence is produced by heating with an electric arc drawn between a tungsten electrode and the work. Filler metal, if required, is added by feeding a bare metal rod or wire into the weld pool. Shielding of the weld is obtained from an inert gas mixture.

c. With GMA, coalescence is effected by heating with an arc drawn between a continuous feed wire electrode and the work. Shielding of the weld is obtained from an externally supplied inert mixture.

d. Submerged arc (SA) produces coalescence by heating with an arc or arcs drawn between a bare metal (filler) electrode or electrodes and the work. The arc and weld are shielded by a blanket of granular fusible flux.

The procedures used in welding nozzles with CE manufacturing facilities are generally as follows. For nozzles with stainless steel safe ends, the safe ends are not attached until after final stress relief. The stainless steel safe end is welded to Inconel buttering on the alloy steel and the weld made using Inconel weld wire. With the above procedure, furnace sensitizing of stainless steel is precluded.

During manufacture of the core structures, various parts of the core structure are tested for sensitization using the Strauss Test (ASTM A393). Test specimens consist of: (1) mock ups of various welded joints, and (2) monitoring specimens included in any heat treatment of various components. None of the specimens tested in conjunction with fabrication of reactor vessel internals for previous CE plants have failed the Strauss Test.

The typical weld heat input with the above processes as used by CE to join 300 Series stainless steel varies from 6000 joules per inch (GTA) to 96,000 joules per inch (SA). To avoid weld HAZ sensitization, CE limits the interpass temperature on multipass welds in stainless steels to 350°F maximum. The combination of normal heat input using the above welding procedures and control of interpass temperature assures minimum carbide precipitation in the weld HAZ. Samples from large welds have been examined in the laboratory and none have failed the Strauss Test.

In field welding operations, Bechtel uses welding procedures that limit heat input to the weld areas, and thus precludes the possibility of sensitization of austenitic stainless steels. Most of the welding employed is of the manual SMA process; a minor amount of GMA welding is also used. Neither one of these processes would be classified as an excessively high heat input welding procedure.

Further precautions employed to preclude field sensitization of austenitic stainless steels consist of:

- Preheat and interpass temperatures are limited to 350°F maximum.
- Controlled welding sequence is used to minimize heat input.
- The practice of block welding is prohibited.
- Post-weld heat treatment is prohibited on equipment and/or parts that are completely or partially fabricated of austenitic stainless steel.
- Application of heat to correct weld distortions resulting in dimensional deviations in equipment and/or parts fabricated of austenitic stainless steel is prohibited.

In preparing for and engineering the field welding requirements, close liaison was maintained between Bechtel and CE. Detailed welding parameters prepared by CE were submitted to Bechtel. Based on this information and its own welding practices, Bechtel prepared detailed welding procedures which were then submitted to CE for review and mutual concurrence and approval before they were adopted for use. Bechtel quality assurance procedures for field welding are discussed in Appendix 5B.

Nitrogen-enhanced stainless steel, type 304 or 316, was not used in the fabrication of pressure containing stainless steel parts of the reactor coolant pressure boundary or those load-bearing stainless steel members which are vital to the structural integrity of the reactor vessel core. The process of electroslag welding was not used in the fabrication of any components within the reactor coolant boundary. All B31.7 Class I and Class II piping welds in the system have been inspected using radiographic techniques as required by applicable codes. In addition, welds that fall within the scope of ASME B&PV Code, Section XI will be inspected as shown in Technical Specifications. The selection of welds to be inspected on the pre-service and subsequently on inservice are based on ASME B&PV Code, Section XI.

4.1.4.4 Seismic Design

The NSSS is designed to withstand the loads imposed by the maximum hypothetical accident and the maximum seismic disturbance without loss of functions required for reactor shutdown and emergency core cooling. The method of combining stresses produced by these simultaneous conditions is described in Section 4.1.1.

4.1.4.5 Prevention Of Brittle Fracture

Ferritic materials, such as carbon steels and low-alloy steels, undergo a ductile-to-brittle transition. The temperature at which this transition occurs is a function of many variables including material composition, processing, and neutron irradiation.

To ensure that brittle failure will not occur special attention must be given to materials selection, fabrication procedure, and operating procedures and the effect of irradiation upon the material properties. Gamma heating in the vessel wall accounts for only 20°F temperature rise at the outer vessel wall at the core midplane. A complete and thorough stress analysis of RCS components which are part of the pressure boundary establishes the stress distributions. These analyses include the effects of geometric shapes and surface stress concentrations. Following fabrication of the components, a post-weld heat treatment reduces the effect of residual stresses, the magnitudes of which would otherwise be unknown.

4.1.4.5.1 Material NDTT

The original materials that form the pressure boundary of the RCS have impact properties which meet the requirements of ASME B&PV Code, 1965, N67, Section III, Paragraph N-330, at 40°F or less, except the CEDMs which meet these requirements at 75°F or less. Materials for several replacement components were designed and fabricated to later Code editions. Refer to Table 4-9 for specific code requirements.

For the RSGs and the replacement RVCH, the RT_{NDT} for each pressure boundary plate, forging, and weld is equal to or less than 0°F.

4.1.4.5.2 NDTT Shift During Irradiation

Figure 4-12 presents the CE design curve for NDTT increase as a function of integrated neutron exposure (E>l Mev) for A-302-B and A 533 B steel irradiated at 550°F. This curve forms an outer envelope with respect to all known irradiation data available in April 1970. The data was compiled from the reference documents listed in Table 4-11. This design curve provides a conservative prediction for the change in NDTT with irradiation.

4.1.4.5.3 NDTT Determination

The reactor vessel is designed and fabricated in such a manner that significant operational limitations will not be imposed on the RCS resulting from shifts in reactor vessel NDTT. The vessel material monitoring program will be conducted within the guidelines of ASTM E185, "Recommended Practice for Surveillance Tests on Structural Materials in Nuclear Reactors." The pre-irradiated NDTT of the baseplate material has been established using drop weight tests in accordance with ASTM E208 and correlations made with Charpy impact specimen tests conducted in accordance with ASTM A370. This correlation, along with the Charpy impact specimens irradiated in the surveillance program, was used to monitor vessel material NDTTs.

For the pre-irradiated Charpy tests, a minimum of three specimens of each material were tested at any one temperature. Tests were performed at a sufficient number of different temperatures to establish the energy-temperature curve.

The test material used in establishing the unirradiated NDTT of the base metal was obtained from (1/4) T (where T is plate thickness) locations of sections of the plate used in the intermediate or lower shell courses. The thermal history of the plate from which the specimens were taken was representative of that of the shell plating. The impact properties at this location were considered to be representative of the material through the plate. The properties at the (1/4) T location were used to establish the initial minimum operating temperature and form the basis for the predicted minimum operating temperature after irradiation.

The material toughness test requirements were as follows:

a. For the Reactor Vessel:

   Carbon and low-alloy steel materials which form a part of the pressure boundary shall meet the requirements of ASME B&PV Code, 1965, N67, Section III, Paragraph N-330 at a temperature of +40°F. Refer to Table 4-9 for specific code requirements. It shall be an objective that the materials meet this requirement at +10°F. Charpy tests shall be performed and the results used to plot a transition curve of impact values vs. temperature extending from fully brittle to fully ductile behavior. The actual NDTT of inlet and outlet nozzles, vessel and head flanges, and shell and head materials shall be determined by drop weight tests per ASTM E208. NDTT will be established by Charpy test. Drop weight tests will be conducted and the results used for information only. The replacement RVCH is made of two forgings - a flange and dome portion. Both forgings meet the ASME B&PV Code (1995 Edition, 1996 Addenda) Section III, NB 2300. The maximum RT_{NDT} is confirmed to be 0°F or less. This has been confirmed by drop weight testing per ASTM E208-91.

b. For the RSG and Pressurizer:

   It shall be an objective that impact properties of all ferritic steel materials forming a part of the pressure boundary shall meet the requirements of ASME B&PV Code, Section III, at a temperature of 0°F for the SG lower assembly, +10°F for the SG steam drum, and +10°F for the pressurizer shell. For the SG steam drum and pressurizer, alternate higher temperature levels up to 40°F may be used only if the material fails at +10°F. Such higher temperature levels, if applicable, shall be determined and documented.

c. For the Reactor Coolant Piping:

   Materials used to fabricate the pipe and fittings shall be specified, examined and tested to satisfy, as a minimum, the requirements of Chapter I-III of ANSI Code for Pressure Piping B31.7, Class 1. Impact properties of carbon steel materials, including welds, shall have a minimum V-notch value of 20 ft/lb (average of three specimens) or 15 ft/lb (any individual specimen) at 40°F. It shall be a design objective that the materials meet this requirement at 10°F. Weld procedure qualifications and weld metal certifications records may serve to demonstrate impact properties of welds.

The maximum NDTT for the reactor vessel as obtained from drop weight tests is: Unit 1 = +10°F, Unit 2 = +30°F. Drop weight tests were conducted only on material used in the reactor vessel. The maximum RT_{NDT} for the replacement RVCH is 0°F or less.

The minimum upper-shelf C_v energy value for the strong direction of the material used in the reactor vessel is: Unit 1 = 101 ft/lb, Unit 2 = 108 ft/lb.

The upper-shelf C_v energy was not determined for the materials used in fabricating the SG, pressurizers, or reactor coolant piping. The data was not obtained for the weak direction in the material used to fabricate the reactor vessels.

The identification and location of the material relating to limiting values listed above is as follows:

a) For Maximum NDTT:

Unit 1 -	reactor vessel outlet nozzles	(forging)
	reactor vessel outlet nozzle extensions	(forging)
Unit 2 -	reactor vessel flange	(forging)

b) For Minimum upper-shelf C_v energy value:

Unit 1 -	reactor vessel outlet nozzle	(forging)
Unit 2 -	reactor vessel outlet nozzle	(forging)

4.1.4.5.4 Radiation Embrittlement of Reactor Pressure Vessel Materials and Fracture Toughness Requirements for Protection Against PTS Events

Neutron irradiation of reactor pressure vessel materials reduces the fracture toughness of these materials over time. The reactor pressure vessel fracture toughness is an important material property that must meet minimum requirements to ensure the reactor pressure vessels are able to withstand pressurized thermal shock (PTS) events.

10 CFR Part 50 Appendices G and H require monitoring changes in fracture toughness of reactor pressure vessel materials induced by neutron irradiation. Regulatory Guide 1.99 Revision 2 provides a method acceptable to the Nuclear Regulatory Commission for calculating the changes in the fracture toughness of the reactor pressure vessel materials. The current PTS rule contained in 10 CFR 50.61, Fracture Toughness Requirements for Protection against Pressurized Thermal Shock Events, incorporates the requirements of RG 1.99, Revision 2.

Both the PTS rule and RG 1.99 Revision 2 require the computation of a value for each reactor pressure vessel material representing the effect of neutron embrittlement on that material. This value is given in terms of the reference temperature (RT_{NDT}). The value from RG 1.99, Revision 2 is called the ART and is determined by the following equation:

$$ART = \text{Initial } RT_{NDT} + \Delta RT_{NDT} + \text{Margin}$$

The Initial RT_{NDT} is the initial reference temperature for the unirradiated material as defined by ASME B&PV Code, Section III, Paragraph NB-2331. Tables 4-11A and 4-11B provide the initial RT_{NDT} and other material properties for the reactor vessel beltline.

The shift in the reference temperature caused by irradiation (ΔRT_{NDT}) is the product of a chemistry factor and a fluence factor. Regulatory Guide 1.99, Revision 2 provides two ways to determine the chemistry factor.

a. By using tabular values given in RG 1.99, Revision 2; or
b. By a calculation using a "Least Squares" fit to a minimum of two sets of credible surveillance data.

Margin is a quantity given in RG 1.99, Revision 2 that is to be added to yield conservative upper-bound values of the ART.

The PTS Rule provides the same methodology as RG 1.99, Revision 2, but uses slightly different terminology. The ART is equivalent to RT_{PTS} which is given by:

$$RT_{PTS} = \text{Initial } RT_{NDT} + \Delta RT_{PTS} + \text{Margin}$$

where ΔRT_{PTS} is computed identically to ΔRT_{NDT}.

Calvert Cliffs Nuclear Power Plant (CCNPP) has submitted projected values of RT_{PTS} for Calvert Cliffs Units 1 and 2 reactor vessel beltline materials, calculated in accordance with the procedures given in 10 CFR 50.61(b)(2) (References 1, 2, and 3).

Fluence values used in the final RT_{PTS} calculations are based on analyses performed by Babcock and Wilcox Nuclear Technologies (BWNT) in 1993 and 1994 as part of analyzing the 97° capsule removed from each reactor vessel in those years. Those analyses established the actual neutron fluence in the Unit 1 reactor vessel through the end of Cycle 10, and in the Unit 2 reactor vessel through the end of Cycle 9.

The Unit 1 fluence calculations were performed by ABB/Combustion Engineering under subcontract to BWNT, using the two-dimensional neutron transport code DOT-4 (Reference 4), with the 22-group energy structure of the CASK (Reference 5) transport cross-section library. The reaction cross sections were calculated using the SAND (Reference 6) computer code with the DOSDAM (Reference 7) cross-section library, which is based on ENDF/B-V data. The Unit 2 fluence calculations were performed by BWNT also using DOT-4 with the 47-group energy structure of the BUGLE (Reference 8) transport cross-section library. The reaction cross-sections were taken directly from ENDF/B-V.

The final neutron fluence projections for both Units 1 and 2 reflected the 24 month fuel cycle core pattern similar to that used in Unit 1 Cycle 11, and for Unit 1 incorporated the extensive flux reduction measures which were taken during Cycles 10, 11, and 12.

For each beltline material, CCNPP calculated RT_{PTS} values for the projected fluence at the end of the current 40-year license (2014 for Unit 1 and 2016 for Unit 2), and for the end of a renewed operating license (20 years beyond the current operating license).

The projected values of RT_{PTS} do not exceed the screening criteria for any of the Unit 1 or Unit 2 beltline materials. Note, however, that the projections for Unit 1 weld 2-203 A/B/C are based on surveillance data which was obtained from Duke Power Company's William B. McGuire Nuclear Generating Station, Unit 1. The Baltimore Gas and Electric Company November 29, 1993 submittal demonstrated the equivalence of the conditions between Calvert Cliffs Unit 1 and McGuire Unit 1. This now permits CCNPP to use the surveillance results obtained from McGuire Unit 1 to calculate a RT_{PTS} using the "least squares" method described above. Similar weld material has also been placed in a supplemental surveillance capsule which was placed in the Unit 1 reactor vessel in a location left vacant when the first surveillance capsule was removed (263° position). Calvert Cliffs Nuclear Power Plant will continue to update the RT_{PTS} projections for Calvert Cliffs as new data becomes available.

4.1.4.6 Reactor Vessel Thermal Shock

An analysis of the thermal stresses produced in the reactor vessel wall due to the operation of the safety injection system has been performed. The analysis has been reported in a CE report "Thermal Shock Analysis on Reactor Vessels Due to Emergency Core Cooling System Operation," A-68-9-1, and was submitted for the record on Docket No. 50-309, Maine Yankee Atomic Power Station.

Experiments on the heat transfer coefficient involved instrumenting a two-foot square steel block six-inches thick, quenching it from a representative reactor vessel temperature and calculating a suitable average heat transfer coefficient from the test data. This work was reported in a CE report, "Experimented Determination of Limiting Heat Transfer Coefficients during the Quenching of Thick Steel Plates in Water," A-68-10-1, which was submitted to the AEC and is on file in the Public Document Room.

The stress near the tip of axial and circumferential vessel cracks of various depths has been determined by the finite element method. This work was reported by a CE report, "Finite Element Analysis of Structural Integrity of a Reactor Pressure Vessel during Emergency Core Cooling," A-70-19-2, January 1970, and is part of the public record.

These reports substantiate the analytical conclusion that a vessel failure will not occur due to Emergency Core Cooling System (ECCS) operation. An acute crack, even if formed, will not propagate.

4.1.5 TESTS AND INSPECTIONS

4.1.5.1 General

Shop inspection and tests of all major components was performed at the vendor's plant prior to shipment. An inspection at the site was performed to assure that no damage has occurred in transit. Testing of the RCS was performed at the site upon completion of plant construction. These tests included hydrostatic tests of all fluid systems. A complete visual inspection of all welds and joints was performed prior to the installation of the insulation. All field welds were volumetrically and inspected in accordance with the requirements of the Codes applicable to the construction of the component.

A hot flow test of the reactor coolant loop up to full power operating pressure and temperature without the core installed was made. The system was checked for vibration and cleanliness. Auxiliary systems were checked for performance (Chapter 13).

4.1.5.2 Surveillance Program

The surveillance program was implemented to monitor the radiation-induced changes in the mechanical and impact properties of the pressure vessel materials. This surveillance program complies with 10 CFR Part 50, Appendix H, and ASTM E185-70. Changes in the impact properties of the material will be evaluated by the comparison of pre-irradiation and post-irradiation Charpy impact test specimens. Changes in mechanical properties will be evaluated by the comparison of pre-irradiation and post-irradiation data from tensile test specimens.

Three metallurgically different materials representative of the pressure vessel will be investigated. These are base metal, weld metal and weld HAZ material. In addition to the materials from the reactor vessel, material from a standard heat of A533, which has been made available through the Heavy Section Steel Technology (HSST) Program, will also be used. This reference material has been fully processed and heat treated and will be used for Charpy impact specimens so that a comparison may be made between the irradiations in various operating power reactors and in experimental reactors. A complete record of the chemical analysis, fabrication history and mechanical properties of all surveillance test materials will be maintained.

The exposure locations and a summary of the specimens at each location is presented in Table 4-12. The pre-irradiation NDTT of each plate in the intermediate and lower vessel shell courses was determined from the drop weight tests and correlated with Charpy impact tests.

Base metal test specimens were fabricated from sections of the shell plate in either the intermediate or the lower shell course which exhibits the highest unirradiated NDTT. All material for base test specimens was cut from the same shell plate.

The material used for the base metal test specimens was adjacent to the test material used for ASME B&PV Code, Section III tests and was at least one plate thickness away from any quenched edge. This material was heat treated to a condition which is representative of the final heat treated condition of the base metal in the completed reactor vessel.

Weld metal and HAZ material was produced by welding together two plate sections from the intermediate or lower shell course of the reactor vessel. All HAZ test material was also fabricated from the plate which exhibits the highest unirradiated NDTT.

The material used for weld metal and HAZ test specimens was adjacent to the test material used for ASME B&PV Code, Section III tests, and was at least one plate thickness from any water-quenched edge. The procedures used for making the shell girth welds in the reactor vessel was followed in the preparation of the weld metal and HAZ test material. The procedures for inspection of the reactor vessel welds will be followed for inspection of the welds in the test materials. The welded plate was heat treated to a condition which is representative of the final heat treated condition of the completed reactor vessel.

The test specimens were contained in six irradiation capsule assemblies. The axial position of the capsules was bisected by the midplane of the core. The circumferential locations include the peak flux regions.

The location of the surveillance capsule assemblies is shown in Figure 4-13. A typical surveillance capsule assembly is shown in Figure 4-14. A typical Charpy impact compartment assembly is shown in Figure 4-15. A typical tensile monitor compartment assembly is shown in Figure 4-16.

Fission threshold detectors (U-238) were inserted into each surveillance capsule to measure the fast neutron flux. Threshold detectors of Ni, Ti, Fe, S, and Cu with known Co content were selected for this application to monitor the fast neutron exposure. Cobalt was included to monitor the thermal neutron exposure.

The selection of threshold detectors was based on the recommendations of ASTM E261, "Method for Measuring Neutron Flux by Radioactive Techniques." Activation of the specimen material was also analyzed to determine the amount of exposure.

The maximum temperature of the encapsulated specimens is monitored by including in the surveillance capsules small pieces of low-melting-point eutectic alloys or pure metals individually sealed in quartz tubes.

The temperature monitors provide an indication of the highest temperature to which the surveillance specimens were exposed but not the time-temperature history or the variance between the time-temperature history of different specimens. These factors, however, will affect the accuracy of the estimated vessel material NDTT to only a small extent.

The periodic analysis of the surveillance samples will permit the monitoring of the neutron radiation effects upon the vessel materials.

Test specimens removed from the surveillance capsules will be tested in accordance with ASTM Standard Test Methods for Tension and Impact Testing. The data obtained from testing the irradiated specimens will be compared with the unirradiated data and an assessment of the neutron embrittlement of the pressure vessel material will then be made. This assessment of the NDTT shift is based on the temperature shift in the average Charpy curves, the average curves being considered representative of the material.

The integrated fast neutron dose (fluence) to the reactor vessel has been calculated using the methods described in Section 4.1.4.5.2. The predicted change in NDTT as a function of vessel fluence is shown in Figure 4-12.

All surveillance capsules were inserted into their designated holders during the final reactor assembly operation. Capsule withdrawal schedules are listed in Table 4-13A (Unit 1) and Table 4-13B (Unit 2). For capsule withdrawal history, refer to References 31, 32, and 33 in Section 4.1.6.

4.1.5.2.1 Supplemental Surveillance and Dosimetry

The 263° location left vacant by the withdrawal of the first surveillance capsule in Unit 1 was employed in a supplemental dosimetry program. The program consisted of installing one set of in-vessel and ex-vessel neutron dosimetry at azimuthally identical locations. This set of dosimetry capsules was installed at the beginning, and removed at the end, of Cycle 9.

During fabrication of the replacement in-vessel supplemental dosimetry capsule, installed in the 263° location in Unit 1 at the start of Cycle 10, an archival weld block comprised of Calvert Cliffs limiting material was located in Duke Power Company's McGuire Unit 1 Surveillance Program. Charpy impact specimens were fabricated from the McGuire archival weld block and from another reactor vessel weld with a similar weld chemistry and were inserted into the replacement capsule. Sufficient Charpy specimens were installed to provide two sets of data. The removal schedule for the in-vessel capsule with the Calvert Cliffs controlling material will be determined based on flux reduction plans and as a need dictates.

An ex-vessel dosimetry capsule was installed at the 263° location at the beginning of Cycle 10. This capsule was removed at the end of Cycle 10 in conjunction with removal of the azimuthally equivalent (not identical) 97° in-vessel surveillance capsule. At that time, it was determined that sufficient data had been obtained from this capsule, and replacement ex-vessel dosimetry was not installed at the end of Cycle 10. Further information can be found in Reference 33.

4.1.5.3 Non-destructive Tests

Prior to and during fabrication of the reactor vessel, non-destructive tests based upon ASME B&PV Code, Section III were performed on all welds, forgings and plates as follows.

All full penetration pressure-retaining welds were 100% radiographed to the standards of ASME B&PV Code, Section III, Paragraph N-624. Other pressure-retaining welds such as used for the attachment of mechanism housings, vents and instrument housings to the reactor vessel head were inspected by liquid-penetrant tests of the root passes, each one-half inch of weld material or one-third of weld thickness (whichever is less), and the final surface. The full penetration welds on the replacement RVCH were 100% RT inspected per ASME B&PV Code, Section III, Division 1, NB-5000 and ASME B&PV Code, Section V, Article 2. The other pressure retaining welds were PT inspected progressively at 1/2” increments and the final weld surfaces were inspected.

All forgings were inspected by ultrasonic testing, using longitudinal beam techniques. In addition, ring forgings were tested using shear wave techniques. Rejection under longitudinal beam inspection, with calibration so that the first back reflection is at least 75% of screen height, was based on interpretation of indications causing complete loss of back reflection. Rejection under shear wave inspection was based on indications exceeding in amplitude the indication from a calibration notch whose depth is 3% of the forging thickness, not exceeding 3/8" with a length of 1". All forgings for the replacement RVCH were UT inspected per the requirements of the ASME B&PV Code, Section III, NB-2542.

All forgings were also subjected to magnetic-particle examination or liquid-penetrant testing depending upon the material. Rejection was based on ASME B&PV Code, Section III, Paragraph N626.3 (NB-2545 for the replacement RVCH) for magnetic-particle and Paragraph N627.3 (NB-2546 for the replacement RVCH) for dye-penetrant testing.

Plates were ultrasonically tested using longitudinal ultrasonic testing techniques. Rejection under longitudinal beam testing performed in accordance with ASME B&PV Code, with calibration so that the first back reflection is at least 50% of screen height, was based on defects causing complete loss of back reflection which could not be contained within a circle whose diameter is the greater of three inches or one-half the plate thickness. Two or more defects smaller than described above, which caused a complete loss of back reflection were unacceptable unless separated by a minimum distance equal to the greatest diameter of the larger defect unless the defects were contained within the area described above.

Non-destructive testing of the vessel was performed during several stages of fabrication with strict quality control in critical areas such as frequent calibration of test instruments, metallurgical inspection of all weld rod and wire, and strict adherence to the non-destructive testing requirements of ASME p87 Code, Section III.

The detection of flaws in irregular geometries was facilitated because most non-destructive testing of the materials was completed while the material was in its simplest form. Non-destructive inspection during fabrication was scheduled so that full penetration welds were capable of being radiographed to the extent required by ASME B&PV Code, Section III.

Each of the vessel studs received one ultrasonic test and one magnetic particle inspection during the manufacturing process.

The ultrasonic test was a radial longitudinal beam inspection, and a discontinuity which caused an indication with a height which exceeded 20% of the height of the adjusted first back reflection was cause for rejection. Any discontinuity which prevented the production of a first back reflection of 50% of the screen height was also cause for rejection.

The magnetic-particle inspection was performed on the finished studs. Linear axially-aligned defects whose lengths are greater than one-inch long and linear nonaxial defects were unacceptable.

Prior to and during fabrication of the components of the RCS, non-destructive testing based upon the requirements of ASME B&PV Code, Section III is used to determine the acceptance criteria for various size flaws. The requirements for the Class A vessels are the same as the reactor vessel. Vessels designated as Class C will be fabricated to the standards of ASME B&PV Code, Section III, Article 21.

Table 4-14 summarizes the component inspection program during fabrication and construction. Periodic tests and examinations of the RCS have been conducted after startup on a regular basis. For pre-operational and inservice structural surveillance of the RCS, refer to Section 4.0 of the Technical Specifications.

4.1.5.4 Additional Tests

During design and fabrication of the reactor vessel, additional operations beyond the requirements of ASME B&PV Code, Section III were performed by the vendor. Table 4-15 summarizes the additional tests by component.

During the design of the reactor vessel, detailed calculations were performed to assure that the final product would have adequate design margins. A detailed fatigue analysis of the vessel for all design conditions has been performed. In those areas which are not amenable to calculation, stress concentrations have been obtained through the use of photo-elastic models. In addition, CE has performed test programs for the determination and verification of analytical solutions to thermal stress problems. Also, fracture mechanics and brittle fracture evaluations have been performed.

All material used in the reactor vessel was carefully selected and precautions were taken by the vessel fabricator to insure that all material specifications were adhered to. To assure compliance, the quality control staff of CE reviewed the mill test reports and the fabricator's testing procedures.

All welding methods, materials, techniques, and inspections comply with ASME B&PV Code, Sections III and IX. Before fabrication was begun, detailed qualified welding procedures, including methods of joint preparation, together with certified procedure qualification test reports, were prepared. Also, prior to fabrication, certified performance qualification tests were obtained for each welder and welding operator. Quality control was exercised for all welder and wire by subjection to a complete and thorough testing program in order to insure maximum quality of welded joints.

During the manufacture of the reactor vessel, in addition to the areas covered by ASME B&PV Code, Section III, quality control by the vendor included:

a. preparation of detailed purchase specifications which included cooling rates for test samples;
b. requiring vacuum degassing for all ferritic plates and forgings;
c. specification of fabrication instructions for plates and forgings to provide control of material prior to receipt and during fabrication;
d. use of written instructions and manufacturing procedures which enabled continual review based on past and current manufacturing experiences;
e. performance of chemical analysis of welding electrodes, welding wire, and materials for automatic welding, thereby providing continuous control over welding materials;
f. the determination of NDTT through use of drop weight testing methods as well as Charpy impact tests; and
g. test programs on fabrication of plates up to 15" thick to provide information about material properties as thickness increases.

Longitudinal wave ultrasonic testing was performed on 100% of all plate material.

Cladding for the reactor vessel is a continuous integral surface of corrosion-resistant material, 5/16" nominal thickness. The detailed procedure used, i.e., type of weld rod, welding position, speed of welding, non-destructive testing requirements, etc., was in compliance with ASME B&PV Code. One hundred percent ultrasonic testing of the reactor vessel cladding has been performed.

Upon completion of all post-weld heat treatments, the reactor vessel was hydrostatically tested, after which all weld surfaces, including those of welds used to repair material, were magnetic-particle inspected in accordance with ASME B&PV Code, Section III, Paragraph N-618.

Surveillance of the quality control program was also carried out during the manufacture of the vessel by the Windsor quality control Section of CE and by the applicant with an independent consultant. This work included independent review of radiographs, magnetic-particle tests, ultrasonic tests, and dye-penetrant tests conducted during the manufacture of the vessel. A review of material certifications, and vendor manufacturing and testing procedures was also conducted. Manufacturers' records such as heat-treat logs, personnel qualification files and deviation files were also included in this review.

4.1.5.5 In-Service Inspection

Consideration for probable inservice inspection of the reactor coolant pressure boundary was made in the early stages of plant design. During design development of the containment internal structures, equipment arrangement and system piping design, provisions were made to facilitate in-service inspection in accordance with ASME B&PV Code, Section XI.

Design improvements have been made to the reactor vessel to facilitate inservice inspection. Additional room has been provided between the nozzle piping surrounding concrete to allow inspection of the piping.

Provisions have been made for access to perform the inspections. The general scheme of access is as follows:

a. Closure Head

   Head to Flange Weld, Cladding, and Nozzle Welds - Accessible for inspection with head in laydown position.

b. Reactor Vessel

   Vessel to Flange Weld - Accessible for inspection with head removed.

   Vessel to Nozzle Welds, Longitudinal Welds, Circumferential Welds and Cladding. Available for inspection from the inside with the core barrel removed. Partial inspection of these welds is also possible from the outside.

   Bottom Head Welds - Available for inspection from the inside with core barrel removed. Sufficient room for outside inspection using remote equipment.

c. Reactor Coolant Piping - Removable insulation allows access to butt welds and the required adjacent sections longitudinal welds.

d. Steam Generators and Pressurizer - Removable insulation provides access to welds from the outside. A remote inspection device is used to inspect the cavity which extends upward from the bottom of each SG. Cladding is accessible by removing man ways.

e. Reactor Coolant Pump Casings - Alternate examinations to those required by Section XI have been approved by the NRC as follows:

   1. The pump interior will be inspected to the extent practical should the pump be disassembled for any other reason.
   2. The RCPs shall be hydrostatically tested per the requirements of ASME Code Section XI.
   3. A surface examination of one RCP in each unit shall be performed on the exterior casing weld surface areas by the liquid penetrant method once per interval.
   4. A visual examination of one RCP in each unit shall be performed on the exterior pump case surfaces once per interval.

   Other components are generally arranged with sufficient surrounding space and removable insulation to allow inspection.

Calvert Cliffs Nuclear Power Plant retained CE and Southwest Research Institute to perform the pre-service baseline inspection of the Calvert Cliffs systems in accordance with ASME B&PV Code, Section XI.

It is planned to inspect the bulk of the reactor vessels and nozzles from the inside using a remote inspection device. This device is capable of volumetrically inspecting the longitudinal welds, circumferential welds, vessel to flange weld, nozzle to vessel welds, nozzle to transition piece welds and transition piece to pipe welds. The closure head to flange weld will be inspected.

In some areas, alternatives to the above inspection methods may be employed if they are feasible and considered desirable.

4.1.5.6 Pre-operational Vibration Test Program

An analysis has been performed and the results show that the natural frequency of the Reactor Coolant Loops is between 4 and 5 cps. The RCPs will impose frequencies of 14-15 cps from the shaft rotation and 70-75 cps from the impeller. Therefore, no vibrational problem is anticipated.

Piping and components within the reactor coolant boundary were observed by start-up engineers during testing and startup. The RCPs are equipped with vibration switches to alarm when vibration exceeds 0.0015" double amplitude. The remainder of the systems will be visually inspected. Any visible vibration will be evaluated to determine the resultant stress levels. Should any vibration which results in unacceptable stress levels occur, the condition causing them will be corrected.

The transient conditions under which the systems were inspected for vibrations are as follows:

a. Starting and stopping each RCP under full operating pressure. Running all combinations of RCPs.

b. Starting and stopping pressurizer spray flow.

c. Starting and stopping charging flow. Starting and stopping letdown flow.

d. Starting and stopping safety injection flow with the RCS vented to the quench tank. Open and close the motor operated valves in the path from the safety injection pumps to the coolant loops with the pumps operating. Open and close the motor operated valves isolating the safety injection tanks from the coolant loops.

e. Starting and stopping feedwater flow from the main feed pumps and the auxiliary feed pumps.

f. Starting and stopping steam flow through the turbine stop valves, through the turbine bypass valves and through the atmospheric steam dumps.

4.1.5.7 Boric Acid Corrosion Monitoring Program

A program has been implemented at Calvert Cliffs to monitor for boric acid leakage onto carbon steel components. The program consists of systematic measures to ensure that boric acid corrosion does not lead to an increase in the probability of abnormal leakage, rapidly propagating failure, or gross rupture of the reactor coolant pressure boundary.

Carbon steel components that could be subject to a boric acid environment are examined on a refueling outage basis.

4.1.6 REFERENCES

1. Letter from G. C. Creel (BGE) to NRC Document Control Desk, dated December 13, 1991, Response to 1991 Pressurized Thermal Shock Rule

2. Letter from R. E. Denton (BGE) to NRC Document Control Desk, dated November 29, 1993, Calvert Cliffs Nuclear Power Plant, Unit No. 1, Docket No. 50-317, Request for Approval to Use Plant-Specific Data for Reactor Vessel Fracture Toughness Analysis

3. Letter from R. E. Denton (BGE) to NRC Document Control Desk, dated July 21, 1995, Request for Approval of Updated Values of Pressurized Thermal Shock (PTS) Reference Temperatures (RT_{PTS}) Values (10 CFR 50.61)

4. Rhoades, W. A., and Childs, R. L., "An Updated Version of the DOT-4 One- and Two-Dimensional Neutron/Photon Transport Code," Oak Ridge National Laboratory, ORNL-5851, April 1982

5. "CASK Cross Section Library," Radiation Shielding Information Center, DLC-23F, Oak Ridge National Laboratory, Oak Ridge, TN 37830

6. "SAND Neutron Flux Spectra Determination by Multiple Foil Activation Iterative Method," Oak Ridge National Laboratory, CCC-112, Oak Ridge, TN 37830

7. "DOSDAM 81-82 Multigroup Cross Sections in SAND II Format for Spectral, Integral, and Damage Analysis," Oak Ridge National Laboratory, DLC-97, Oak Ridge, TN 37830

8. "BUGLE-80 Coupled 47 Neutron, 20 Gamma-Ray, P3, Cross Section Library for LWR Shielding Calculations," Radiation Shielding Information Center, DLC-75, Oak Ridge National Laboratory, Oak Ridge, TN 37830

9. J.R. Hawthorne & U. Potapovs, "Initial Assessments of Notch Ductility Behavior of A533 Pressure Vessel with Neutron Irradiation" NRL Report 6772, November 22, 1968

10. M.S. Hersh, F.A. Brandt, B.C. Beaudreau: "Dresden Nuclear Power Station Reactor Vessel Surveillance Program," GECR 5165, March 1966

11. L.E. Steele, et al, "Irradiation Effects Reactor Structural Materials," NRL Memorandum Report 1593, February 15, 1965.

12. L.E. Steele J. and Hawthorne, "New Information on Neutron Embrittlement and Embrittlement Relief of Reactor Pressure Vessel Steels," NRL Report 6160, October 1964

13. L.E. Steele, et al, "Irradiation Effects on Reactor Structural Materials," NRL Memorandum Report 1534, May 15, 1964.

14. L.E. Steele, et al, "Irradiation Effects on Reactor Structural Materials," NRL Memorandum Report 1676, February 15, 1966

15. J.R. Hawthorne, et al., "Irradiation Effects on Reactor Structural Materials," NRL Memorandum Report 1753, February 15, 1967

16. J.R. Hawthorne, et al, "Irradiation Effects on Reactor Structural Materials," NRL Memorandum Report 1833, November 15, 1967

17. J.R. Hawthorne and U. Potapovs, "Initial Assessments of Notch Ductility Behavior of A533 Pressure Vessel Steel with Neutron Irradiation," NRL Report 6722, November 2, 1968

18. L.E. Steele, et al, "Irradiation Effects on Reactor Structural Materials," NRL Memorandum Report 1937, November 15, 1968

19. L.E. Steele, et al, "Irradiation Effects on Reactor Structural Materials," NRL Memorandum Reports 2027, August 15, 1969

20. F.J. Witt, Program Director, Heavy-Section Steel Technology Program, Semiannual Progress Report for Period Ending August 31, 1968, ORNL-4377 April 1969

21. J.R. Hawthorne and U. Potapovs, "Initial Assessments of Notch Ductility Behavior of A533 Pressure Vessel Steel with Neutron Irradiation," NRL Report 6722 November 22, 1968

22. L.E. Steele, et al, "Irradiation Effects on Reactor Structural Materials," NRL Memorandum Report 1937, November 15, 1968

23. L.E. Steele, et al, "Irradiation Effects on Reactor Structural Materials," NRL Memorandum Report 2027, August 15, 1969

24. F.J. Witt, Program Director, Heavy-Section Steel Technology Program, Semiannual Progress Report for Period Ending August 31, 1968, ORNL-4377, April 1969

25. J.R. Hawthorne and U. Potapovs, "Initial Assessments of Notch Ductility Behavior of A533 Pressure Vessel Steel with Neutron Irradiation," NRL Report 6772, November 22, 1968

26. F.J. Witt, Program Director, Heavy-Section Steel Technology Program, Semiannual Progress Report for Period Ending August 31, 1968, ORNL-4377, April 1969

27. R.G. Berggren, et al "Radiation Studies on HSST Plate and Welds," Paper given at ORNL for NSST Program Information Meeting, April 1, 1970

28. T.R. Mager and F.O. Thomas, "Heavy Section Steel Technical Report No. 5 (Nov. 1969) Evaluation by Linear Elastic Fracture Mechanics of Radiation Damage to Pressure Vessel Steels," October 1969

29. J.R. Hawthorne, "Improved Radiation Embrittlement Resistance in Commercially Produced A533B Plate and Weld Metal," Paper given at ORNL for HSST Program Information Meeting, April 1, 1970

30. F.J. Witt, Program Director, Heavy-Section Steel Technology Program, Semiannual Progress Report for Period Ending February 28 1969, ORNL-4463, January 1970.

31. Letter from R. A. Clark (NRC) to A. E. Lundvall, Jr. (BGE), dated February 2, 1982, Revised Unit 1 Capsule Removal Schedule

32. Letter from A. W. Dromerick (NRC) to C. H. Cruse (BGE), dated December 15, 1997, Revision to the Reactor Vessel Surveillance Capsule Withdrawal Schedule for CCNPP

33. Letter from D. G. McDonald, Jr. (NRC) to R. E. Denton (BGE), December 23, 1992, Clarification of Previous Reactor Vessel Surveillance Program Actions and Withdrawal Schedule Change for the Unit 2 Reactor Vessel Material Specimens

34. Perrin, J.S. et al., "Calvert Cliffs Unit No. 1 Nuclear Plant Reactor Pressure Vessel Surveillance Program: Capsule 263," Battelle Columbus Laboratories, December 1980

35. Lowe, A.L., Jr., et al., "Analysis of Capsule 97° Baltimore Gas and Electric Company Calvert Cliffs Nuclear Power Plant Unit No. 1," B&W Nuclear Service Company, BAW-2160, June 1993

36. Norris, E.B., "Reactor Vessel Material Surveillance Program for Calvert Cliffs Unit 2 Analysis of 263° Capsule, " Southwest Research Institute, SwRI-7524, September 1985

37. Lowe, A.L., Jr., et al., "Analysis of Capsule 97° Baltimore Gas and Electric Company Calvert Cliffs Nuclear Power Plant Unit No. 2," B&W Nuclear Service Company, BAW-2199, February 1994

38. Bechtel Specification 6750-M-0310F, "Design Specification for Piping, Valves, and Associated Equipment of the Pressurizer Spray System"

39. Bechtel Specification 6750-M-0310D, "Design Specification for Piping, Valves, and Associated Equipment of the Charging System and Auxiliary Spray System"

40. Babcock & Wilcox, Canada Letter Report BWC-LR-2000-0006, Revision 3, “Reactor Coolant System Structural Evaluation”

41. Westinghouse Letter No. LTR-SGDA-02-76, Revision 0, March 15, 2002, "Review of Calvert Cliffs Steam Generator Manufacturing Records for RT of Secondary Side Head, Shell and Cone Plates"

TABLE 4-1

PRINCIPAL DESIGN PARAMETERS OF REACTOR COOLANT SYSTEM

Design Thermal Power,	
MWt	2737
Btu/hr	9.34×10^9
Design Pressure, psia	2500
Design Temperature (Except Pressurizer), °F	650
Coolant Flow Rate through core, Minimum, gpm	370,000
Cold Leg Temperature, Operating, °F	548
Average Temperature, Maximum, °F	577
Hot Leg Temperature, Maximum, °F	604
Normal Operating Pressure, psia	2250
System Volume, ft^3 (Without Pressurizer)	9576[(a)]
Pressurizer Water Volume, ft^3	800
Pressurizer Steam Volume, ft^3	700

(a) With no plugged SG tubes.

TABLE 4-2

REACTOR VESSEL PARAMETERS

Design Pressure, psia	2500
Design Temperature, °F	650
Nozzles	
Inlet (4 ea), ID, in.	30
Outlet (2 ea), ID, in.	42
CEDM (61), ID, nominal in.	Inlet 2 8/10, Mid 2 8/10, Out 2 4/10
Instrumentation (8), nominal, in.	Inlet 4 3/4, Out 4 6/10
Dimensions	
Inside Diameter, nominal, in.	172
Overall Height, Including CEDM Nozzles, in.	503 3/4
Height, Vessel Without Head, in.	408 9/16
Wall Thickness, minimum, in.	8 5/8
Upper Head Thickness, minimum, in.	7 1/2 min
Lower Head Thickness, minimum, in.	4 3/8
Cladding Thickness, nominal, in.	5/l6 (0.2 min for replacement RVCH)
Material	
Shell	SA-533 Grade B, Class I Steel
Forgings	A-508 Class 2 (SA-508 Gr3 Cl 1 for the replacement RVCH)
Cladding	Stainless Steel[a]
CEDM Nozzles	Ni-Cr-Fe Alloy
Instrumentation Nozzles	Ni-Cr-Fe Alloy
Dry Weights	
Head, lb	145,900
Vessel, without flow skirt, lb	671,400
Studs, Nuts, and Washers, lb	38,500
TOTAL	855,800

[a] Weld deposited type 308-309 stainless steel composition with type 308 in contact with coolant, 308L/309L for the replacement RVCH.

TABLE 4-3

STEAM GENERATOR PARAMETERS

Number	2
Type	Vertical U-Tube
Number of Tubes	8471
Tube Outside Diameter, in.	0.750
Nozzles, Ports, and Manways	
Primary Inlet Nozzle (1 ea), ID, in.	42
Primary Outlet Nozzle (2 ea), ID, in.	30
Steam Nozzle (1 ea), ID, in.	34
Feedwater Nozzle (1 ea), nominal, in.	16
Instrument Nozzles (14 ea), nominal, in.	1
Secondary Inspection Ports (12 ea), nominal, in.	2.5
Primary Manways (2 ea), ID, in.	18
Secondary Manways (2 ea), ID, in.	16
Secondary Handhole (6 ea), ID, in.	8
Bottom Blowdown (2 ea), nominal, in.	2
Surface Blowdown (1 ea), nominal, in.	2
Auxiliary Feedwater (1 ea) nominal, in.	4
Water Level Nozzle (12 ea), nominal, in.	1
Recirculation Nozzle (1 ea), nominal, in.	3
Thermowell Nozzle (1 ea), nominal, in.	1 NPT
Primary Side Design	
Design Pressure, psia/psig	2500/2485
Design Temperature, °F	650
Design Thermal Power (NSSS), MWt	2737
Coolant Flow (Each), lb/hr	$61x10^6$
Normal Operating Pressure, psia/psig	2250/2235
Coolant Volume, each, ft^3, cold conditions	1641.3
Coolant Volume, each, ft^3, hot conditions	1662.4
Secondary Side Design	
Design Pressure, psia/psig	1015/1000
Design Temperature, °F	550
Normal Operating Steam Pressure, Full Load, psia/psig	888/873[a]
Normal Operating Steam Temperature, Full Load, °F	530.4
Blowdown Flow, Full Power Maximum, Both SGs, lb/hr	150,000
Steam Flow (Each), lb/hr	$6.005x10^6/6.015x10^6$
Feedwater Temperature, °F	433.55[c]
Dimensions	
Overall Height, Including Support Skirt, in.	749
Upper Shell Outside Diameter, in.	239 - 3/4
Lower Shell Outside Diameter, in.	164 – 5/16
Dry Weight, tons	521.6
Flooded Weight, tons	824.5
Operating Weight, tons	630.9

[a] With no plugged SG tubes.

[c] Feedwater temperature will decrease if feedwater heaters are bypassed in accordance with operating instructions.

TABLE 4-5

REACTOR COOLANT PUMP PARAMETERS

Number	4
Type	Vertical, Limited Leakage Centrifugal
Shaft Seals	Mechanical (4)
Stationary Face	Carbon Granite
Rotating Support Ring	Tungsten Carbide
Rotating Seal Ring	Tungsten Carbide
Design Pressure, psia	2,500
Design Temperature, °F	650
Normal Operating Pressure, psia	2,250
Normal Operating Temperature, °F	548
Design Flow, gpm	81,200
Total Dynamic Head, ft	243
Maximum Flow (one-pump Operating), gpm	120,000
Dry Weight, lb	141,000
Flooded Weight, lb	148,000
Reactor Coolant Volume, ft^3	112
Motor	
Voltage, volts	13,200
Frequency, hz	60
Phases	3
Horsepower/Speed, Hot, hp/rpm	4500/900
Horsepower/Speed, Cold, hp/rpm	6000/900
Instrumentation	
Seal Temperature Detectors	1
Pump Casing Pressure Taps	2
Seal Pressure Taps	3
Seal Pressure Detectors[a]	2
Controlled Bleedoff Flow Detectors	1
Controlled Bleedoff Temperature Detectors	1
Motor Oil Level Detectors	2
Motor Bearing Temperature Detectors	4
Motor Stator Temperature Detectors	2
Vibration Detector	8
Oil Lift Pressure Detector	2
Lubrication Oil Temperature	2
Total Seal Assembly Leakage (Normal and Standby Operation)	
Three Pressure Seals Operating, gpm	1.50
Two Pressure Seals Operating, gpm	1.84
One Pressure Seal Operating, gpm	2.60

[a] Lower seal pressure transmitter and sensing line were removed and capped under FCR 82-190.

TABLE 4-6

REACTOR COOLANT PIPING PARAMETERS

Number of loops	2
Flow per loop, lb/hr	$61x10^6$
Pipe Size	
Reactor outlet, ID, in.	42
Reactor inlet, ID, in.	30
Surge line, nominal, in.	12
Design Pressure, psia	2500
Design Temperature, °F	650
Velocity Hot leg, ft/sec	42
Velocity Cold leg, ft/sec	37

TABLE 4-7

PRESSURIZER PARAMETERS

Design Pressure, psia	2,500
Design Temperature, °F	700
Normal Operating Pressure, psia	2250
Normal Operating Temperature, °F	653
Internal Free Volume, ft^3	1500
Normal Operating Water Volume, ft^3	600-800
Normal Steam Volume, Full Power, ft^3	700-900
Installed Heater Capacity, kW	1400-Unit 1,[c] 1475-Unit 2[a]
Spray Flow, Maximum, gpm	375
Spray Flow, Continuous, gpm	1.5
Nozzles	
Surge Line (1 ea) nominal, in.	12
Safety and Relief Valves (2) ID, in.	4
Spray (1 ea) nominal, in.	4
Heaters (112-Unit 1[c], 118-Unit 2[a]) OD, in.	0.855 (Unit 1)[d][e] 0.875 (Unit 2)[e]
Instrument, Level (4 ea) nominal, in.	1
Temperature (1 ea) nominal, in.	1
Pressure (2 ea) nominal, in.	1
Materials	
Vessel	A-533, Gr B, Class 1
Cladding	Stainless Steel[b] and Ni-Cr-Fe Alloy
Dimensions	
Overall Length, in.	441 3/8
Outside Diameter, in.	106 1/2
Inside Diameter, in.	95 9/16
Cladding Thickness, in. (minimum)	1/8
Dry Weight, Including Heaters, lb	206,000
Flooded Weight, Including Heaters, lb	300,000

(a) Penetration H-3 was plugged during Unit 2 pressurizer heater sleeves replacement in 1989-90; Location N-3 was plugged in 2011. Any further reduction in Unit 2 pressurizer heater capacity shall be compared against the original (120) heater capacity.

(b) Weld-deposited type 308-309 stainless steel composition with type 308 in contact with coolant.

(c) Unit 1 pressurizer heater sleeve at B-3, FF-1 and CC-1 were plugged in 1994; Location B-1 was plugged in 1998 and location V-1 was plugged in 2012. Locations L3 and BB3 were electrically disconnected in 2014. Location N4 was electrically disconnected in 2016. Any further reduction in Unit pressurizer heater capacity shall be compared against the original (120) heater capacity.

(d) Smaller diameter heaters for Unit 1 to allow for nickel plating of heater sleeves.

(e) Tolerances for these dimensions are proprietary.

TABLE 4-8

TABLE OF LOADING COMBINATIONS AND PRIMARY STRESS LIMITS

LOADING COMBINATIONS	**PRIMARY STRESS LIMITS** Vessels[d]	Piping	Supports
1. Design Loading + Operating Basis Earthquake	$P_M \leq S_M$	$P_M \leq 1.2\ S_h$	Working Stress
	$P_B + P_L \leq 1.5\ S_M$	$P_B + P_M \leq 1.2\ S_h$	
2. Normal Operating Loadings + Safe Shutdown Earthquake	$P_M \leq S_D$	$P_M \leq S_D$	Within Yield
	$P_B \leq 1.5\left[1 - \frac{(P_M)^2}{(S_D)^2}\right] S_D$ (b)	$P_B \leq \frac{4}{\pi} S_D Cos\left(\frac{\pi}{s} \bullet \frac{P_M}{S_D}\right)$ (c)	
3. Normal Operating Loadings + Pipe Rupture + Safe Shutdown Earthquake	$P_M \leq S_L$	$P_M \leq S_L$	Deflection of supports limited to maintain supported equipment within limits shown in columns (1) and (2)
	$P_B \leq 1.5\left[1 - \frac{(P_M)^2}{(S_L)^2}\right] S_L$ (b)	$P_B \leq \frac{4}{\pi} S_L Cos\left(\frac{\pi}{s} \bullet \frac{P_M}{S_L}\right)$ (a),(c)	

NOTES:

(a) These stress criteria are not applied to the piping run within which a pipe break is considered to have occurred.

(b) For loading combinations 2 and 3, stress limits for vessel, with the symbol P_M changed to P_L, should also be used in evaluating the effects of local loads imposed on vessels and/or piping.

(c) The tabulated limits for piping are based on a minimum "shape factor." These limits may be modified to incorporate the shape factor of the particular piping being analyzed.

(d) The above criteria does not apply to the replacement RVCH. Refer to Table 4-9 for the Design Code for the replacement RVCH and CEDMs.

TABLE 4-8

TABLE OF LOADING COMBINATIONS AND PRIMARY STRESS LIMITS

Legend:

P_M	=	Calculated Primary Membrane Stress
P_B	=	Calculated Primary Bending Stress
P_L	=	Calculated Primary Local Membrane Stress
S_M	=	Tabulated Allowable Stress Limit at Temperature from ASME B&PV Code, Section III or ANSI B31.7
S_Y	=	Tabulated Yield at Temperature, ASME B&PV Code, Section III
S_D	=	Design Stress
	=	S_Y (for ferritic steels)
	=	$1.2S_M$ (for austenitic steels)
S_L	=	$S_Y + 1/3 (S_u - S_Y)$
S_u	=	Tensile Strength of Material at Temperature

The following typical values are selected to illustrate the conservatism of this approach for establishing stress limits. Units are 10^3 lb/in^2.

Material	S_Y [(a)]	S_u	S_D	S_L
A 106B	25.4	60.0 [(b)]	25.4	36.9
SA 533B	41.4	80.0 [(b)]	41.4	54.3
304 SS	17.0	54.0 [(c)]	18.35	29.3
316 SS	18.5	58.2 [(c)]	22.2	31.7

a. From ASME B&PV Code, Section III, at 650°F.
b. Minimum value at room temperature which is approximately the same at 650°F for ferritic materials.
c. Estimated.

TABLE 4-9

REACTOR COOLANT SYSTEM CODE REQUIREMENTS

COMPONENT	CODE
Reactor Vessel	ASME III, Class A[(a)(h)]
RSG	ASME III, Class A[(a)(g)] Code Cases 1401, N-20-4, N-411-1, N-474-1, 2142-1, 2143-I, 1332-2, 1332-4, 1359-1
Pressurizer	ASME III, Class A[(a)]
Coolant Pumps	ASME III, Class A[(a)]
Seal Assemblies	ASME III, Class 1[(f)]
Quench Tank	ASME III, Class C[(c)]
Pressurizer Safety and Power Operated Relief Valves	ASME III[(b)]
Piping	ASME III[(a)(d)] USAS B 31.7 Class I[(e)]

NOTES:

(a) The latest editions, addenda and rulings in effect through the winter of 1967 were used for ASME B&PV Code.

(b) The latest editions, addenda and rulings in effect through winter of 1968 were used for ASME B&PV Code.

(c) Code effective date is May 29, 1969. The requirements of Paragraph UW-2(a) of Section VIII apply.

(d) The summer 1969 addenda was added. Code Case N-1401 is included.

(e) Code Cases 83 and 1477 are included.

(f) Replacement Sulzer Bingham RCP seals are designed in accordance with ASME B&PV Code, Section III, 1983 Edition with Summer 1983 Addenda.

(g) ASME B&PV Code 1989 Edition, no addenda is applicable to the RSG lower assembly part.

(h) The replacement RVCH was designed and fabricated to the 1995 Edition, 1996 Addenda of the ASME B&PV Code Section III. The CEDMs were designed and fabricated to the 1998 Edition through 2000 Addenda of Section III.

TABLE 4-9A

REACTOR COOLANT VENT VALVES

Valve Type:	Solenoid operated globe valve, energize to open
Size:	3/4" socket weld ends
Code:	ASME B&PV Code Section III, 1977 Edition, Winter 1977 Addenda, Class 1
Rating:	2173 ANSI B16.34
Seismic:	Seismic Category I
Material:	SA182F, 316L
Design pressure:	2485 psig
Design temperature:	700°F

TABLE 4-10

MATERIALS EXPOSED TO COOLANT

Reactor	
Vessel Cladding	Weld Deposited Type 308 SS (308L/309L for the replacement RVCH)
Vessel Internals	304 SS and Ni-Cr-Fe Alloy
Fuel Cladding	Zircaloy-4, Zircaloy-2P, M5®
Control Element Drive Mechanisms	Ni-Cr-Fe
Piping (excluding surge line)	Austenitic Stainless Steel Cladding Type 304L
Piping (at Hot Leg instrument nozzles)	SA-516, Gr 70 Carbon Steel
Surge Line	ASTM 351, Gr CF8M Alloy Steel
SG	
Bottom Head Cladding	Weld Deposited ER309L ER308L
Tube Sheet Cladding	Weld Deposited UNS N06052 (Code Case 2142-1)
Tubes	SB-163 Alloy 690 (Code Case N-20-4)
Divider Plate	SB-168 UNS N06690
Pumps	
Casing	Austenitic Stainless Steel, Type 3l6
Internals	Austenitic Stainless Steel, Type 316 & 304
Pressurizer	
Cladding	Weld Deposited Stainless Steel Type 308 and Ni-Cr-Fe Alloy
Upper Portion Heater	Electrodeposited Pure Nickel
Sleeve Bores [(a)(c)]	Plating
Upper Level Instrument Nozzle Base	SA-533, Gr B, Cl 1[(b)]
Heater Sleeves [(c)]	SA-213, Grade TP316/TP316L
Lower Level Instrument Nozzle [(c)]	SA-479, Grade TP316/TP316L

(a) Lower portions of the heater sleeves at B-3 and FF-1 are removed from their penetrations leaving a portion of the SA-533, Gr B, Cl 1 base material exposed to reactor coolant.

(b) Repairs to instrument nozzle for 2LT110X resulted in a small portion of the penetration bore base material being exposed to reactor coolant.

(c) The half-nozzle modification of the Unit 1 heater sleeves and lower level instrument nozzle results in a small portion of each penetration bore base material being exposed to reactor coolant.

TABLE 4-11

REFERENCES FOR IRRADIATED MATERIAL TEST DATA USED AS BASIS FOR CE DESIGN CURVE OF FIGURE 4-12

DATA POINT	MATERIAL	FLUENCE ($n/cm^2 x10^{19}$)	NDTT INCREASE (°F)	SELECTED CHEMISTRY (%) P	S	Cu	REFERENCE
1	A302B Plate	0.2	50	0.015	0.021	0.22	For Data Points 1-12 (Reference 9)
2	A302B Plate	1.1	140	0.015	0.021	0.22	
3	A302B Plate	1.5	155	0.015	0.021	0.22	
4	A302B Plate	1.7	140	0.015	0.021	0.22	
5	A302B Plate	2.1	150	0.015	0.021	0.22	
6	A302B Plate	2.3	160	0.015	0.021	0.22	
7	A302B Plate	3.0	155	0.015	0.021	0.22	
8	A302B Plate	3.1	160	0.015	0.021	0.22	
9	A302B Plate	3.1	170	0.015	0.021	0.22	
10	A302B Plate	3.1	155	0.015	0.021	0.22	
11	A302B Plate	3.4	180	0.015	0.021	0.22	
12	A302B Plate	4.8	195	0.015	0.021	0.22	
13	A302B Plate	0.5	75	0.018	0.018	NA[(a)]	For Data Points 13-18 (Reference 10)
14	A302B Plate	1.0	120	0.018	0.018	NA[(a)]	
15	A302B Plate	0.7	100	0.018	0.018	NA[(a)]	
16	A302B Plate	1.5	140	0.018	0.018	NA[(a)]	
17	A302B Plate	4.8	155	0.018	0.018	NA[(a)]	
18	A302B Plate	17.5	225	0.018	0.018	NA[(a)]	
19	A302B Plate	3.0	205	0.012	0.025	0.20	For Data Points 19-22 (Reference 11)
20	A302B Plate	3.0	135	0.009	0.024	NA	
21	A302B Plate	3.0	140	0.009	0.024	NA	
22	A302B Plate	3.0	120	0.009	0.024	NA	
23	A302B Plate	0.5	65	0.009	0.024	NA	For Data Points 23-24 (Reference 12)
24	A302B Plate	3.1	165	0.009	0.024	NA	
25	A302B Plate	3.4	170	Nominal A302B Composition[(b)]			For Data Point 25 (Reference 13)
26	A302B Plate	3.8	160	Nominal A302B Composition[(b)]			For Data Point 26 (Reference 14)
27	A302B Plate	2.9	195	Nominal A302B Composition[(b)]			For Data Points 27-28 (Reference 15)
28	A302B Plate	1.1	130				
29	A302B Plate	2.1	85	0.007	0.018	0.11	For Data Points 29-31 (Reference 16)
30	A302B Plate	2.1	30	0.007	0.018	0.11	
31	A302B Plate	2.1	70	0.007	0.018	0.11	

TABLE 4-11

REFERENCES FOR IRRADIATED MATERIAL TEST DATA USED AS BASIS FOR CE DESIGN CURVE OF FIGURE 4-12

DATA POINT	MATERIAL	FLUENCE ($n/cm^2 x10^{19}$)	NDTT INCREASE (°F)	SELECTED CHEMISTRY (%)			REFERENCE
				P	S	Cu	
32	A533B Plate	2.3	120	0.009	0.022	0.14	For Data Points 32-42 (Reference 17)
33	A533B Plate	2.3	95	0.010	0.023	0.14	
34	A533B Plate	1.7	190	0.010	0.017	0.19	
35	A533B Plate	0.2	0	0.008	0.015	0.09	
36	A533B Plate	2.0	80	0.008	0.015	0.09	
37[(c)]	A533B Plate	2.0	90	0.008	0.015	0.09	
38	A533B Plate	0.5	35	0.008	0.015	0.09	
39	A533B Plate	2.0	75	0.008	0.015	0.09	
40	A533B Plate	1.7	70	0.008	0.015	0.12	
41	A533B Plate	1.7	85	0.008	0.019	0.11	
42	A533B Plate	1.8	50	0.008	0.018	0.12	
43	A533B Plate	0.5	0	0.008	0.014	0.09	For Data Points 43-44 (Reference 18)
44	A533B Plate	2.4	85	0.008	0.014	0.09	
45	A533B Plate	2.5	60	0.003	0.014	0.09	For Data Point 45 (Reference 19)
46[(d)]	A533B Plate	3-4	215	0.012	0.016	0.25	For Data Points 46-47 (Reference 20)
47[(d)]	A533B Plate	3-4	255	0.012	0.016	0.25	
48	A533B Submerged Arc Weld	1.7	200	0.015	0.011	0.22	For Data Points 48-49 (Reference 21)
49	A533 Electroslag Weld	1.8	165	0.008	0.014	0.19	
50	A533B Weld	0.5	0	NA	NA	0.09	For Data Points 50-53 (Reference 22)
51	A533B Weld	2.4	90	NA	NA	0.09	
52	A533B Weld	0.5	105	0.010	0.014	0.14	
53	A533B Weld	2.4	210	0.010	0.014	0.14	
54	A533B Electroslag Weld	2.5	100	0.002	0.012	0.09	For Data Point 54 (Reference 23)
55	A533B Weld	3-4	260	NA	NA	NA	For Data Point 55 (Reference 24)
56	A533B Submerged Arc HAZ	1.7	145	From Data Points 40 or 41			For Data Point 56 (Reference 25)
57	A533B Submerged Arc HAZ	3-4	115	From Data Point 55			For Data Point 57 (Reference 26)
58	A533B Plate	1.0	101	0.012	0.018	NA	For Data Points 58-60 (Reference 27)
59	A533B Plate	1.0	126	0.012	0.018	NA	
60	A533B Plate	1.0	70	0.012	0.018	NA	

TABLE 4-11

REFERENCES FOR IRRADIATED MATERIAL TEST DATA USED AS BASIS FOR CE DESIGN CURVE OF FIGURE 4-12

DATA POINT	MATERIAL	FLUENCE ($n/cm^2 x10^{19}$)	NDTT INCREASE (°F)	SELECTED CHEMISTRY (%)			REFERENCE
				P	S	Cu	
61	A533B Plate	5.0	80	0.012	0.25	NA	For Data Points 61-64 (Reference 28)
62	A533B Plate	4.0	135	0.012	0.25	NA	
63	A533B Plate	1.0	85	0.012	0.25	NA	
64	A533B Submerged Arc Weld	3.5	256	0.019	0.13	0.22	
65	A533B Submerged Arc Weld	2.8	65	0.009	NA	0.03	For Data Points 65-66 (Reference 29)
66	A533B Submerged Arc Weld	2.8	40	0.009	NA	0.03	
67	A533B Submerged Arc Weld	.47	70	0.012	0.018	NA	For Data Points 67-69 (Reference 30)
68	A533B Submerged Arc Weld	.94	95	0.012	0.018	NA	
69	A533B Submerged Arc Weld	1.05	130	0.012	0.018	NA	

NOTES:

(a) Analysis not available.

(b) Specific analysis of material not available.

(c) Transverse specimens.

(d) Exact fluence not reported.

TABLE 4-11A

CALVERT CLIFFS UNIT 1 REACTOR VESSEL BELTLINE MATERIAL PROPERTIES

ID	LOCATION	WELD WIRE SPEC. (Heat No.)	WELD FLUX TYPE. (Lot No.)	PLATE HEAT NO.	Cu (%)	Ni (%)	INITIAL RT_{NDT} (°F)	INITIAL UPPER SHELF ENERGY (ft/lb)
2-203-A,B,C	Intermediate Shell Axial Welds	MIL B-4 Mod. (20291, 12008)	Linde 1092 (3833)	---	0.22	0.83	-50.0	110.0
3-203-A,B,C	Lower Shell Axial Welds	MIL B-4 Mod. (21935)	Linde 1092 (3869)	---	0.17	0.72	-56.0	109.0
9-203	Lower to Intermediate Girth Weld	MIL B-4 (33A277)	Linde 0091 (3922)	---	0.23	0.16	-80.0	160.0
D-7206-1	Intermediate Shell Plate			C4351-2	0.11	0.55	20.0[a]	90.0[a]
D-7206-2	Intermediate Shell Plate	---	---	C4441-2	0.12	0.64	-30.0[a]	81.0[a]
D-7206-3	Intermediate Shell Plate			C4441-1	0.12	0.64	10.0	112.0
D-7207-1	Lower Shell Plate			C4420-1	0.13	0.54	10.0[a]	77.0[a]
D-7207-2	Lower Shell Plate	---	---	B8489-2	0.11	0.56	-10.0[a]	90.0[a]
D-7207-3	Lower Shell Plate			B8489-1	0.11	0.53	-20.0[a]	81.0[a]

All values, except those for Cu and Ni, are from the Comprehensive Reactor Vessel Surveillance Program, Revision 2. The Cu and Ni values can be found in the 1995 PTS submittal.

[a] These values have been corrected for the transverse charpy direction in accordance with NRC Branch Technical Position MTEB 5-2.

TABLE 4-11B

CALVERT CLIFFS UNIT 2 REACTOR VESSEL BELTLINE MATERIAL PROPERTIES

ID	LOCATION	WELD WIRE SPEC. (Heat No.)	WELD FLUX TYPE. (Lot No.)	PLATE HEAT NO.	Cu (%)	Ni (%)	INITIAL RT_{NDT} (°F)	INITIAL UPPER SHELF ENERGY (ft/lb)
2-203-A,B,C	Intermediate Shell Axial Welds	MIL B-4 (A8746)	Linde 124 (3878)	---	0.16	0.10	-56.0	83.5
3-203-A,B,C	Lower Shell Axial Welds	MIL B-4 (33A277)	Linde 0091 (3922)	---	0.23	0.16	-80.0	160.0
9-203	Lower to Intermediate Girth Weld	MIL B-4 (10137)	Linde 0091 (3999)	---	0.21	0.06	-60.0	140.0
D-8906-1	Intermediate Shell Plate			A4463-1	0.15	0.56	10.0[a]	77.0[a]
D-8906-2	Intermediate Shell Plate	---	---	B9427-2	0.11	0.56	10.0[a]	74.0[a]
D-8906-3	Intermediate Shell Plate			A4463-2	0.14	0.55	5.0[a]	75.0[a]
D-8907-1	Lower Shell Plate			C5804-1	0.15	0.60	-8.0[a]	83.0[a]
D-8907-2	Lower Shell Plate	---	---	C5286-1	0.14	0.66	20.0	115.0
D-8907-3	Lower Shell Plate			C5803-3	0.11	0.74	-16.0[a]	84.5[a]

All values, except those for Cu and Ni, are from the Comprehensive Reactor Vessel Surveillance Program, Revision 2. The Cu and Ni values can be found in the 1995 PTS submittal.

[a] These values have been corrected for the transverse charpy direction in accordance with NRC Branch Technical Position MTEB 5-2.

TABLE 4-12

SUMMARY OF SPECIMENS PROVIDED FOR EACH EXPOSURE LOCATION

CAPSULE LOCATION ON VESSEL WALL	BASE METAL			WELD METAL		HAZ		REFERENCE IMPACT[c]	TOTAL SPECIMENS	
	IMPACT		TENSILE	IMPACT	TENSILE	IMPACT	TENSILE		IMPACT	TENSILE
	L[a]	T[b]								
83°	12	12	3	12	3	12	3	-	48	9
97°	12	12	3	12	3	12	3	-	48	9
104°	12	-	3	12	3	12	3	12	48	9
263°	12	-	3	12	3	12	3	12	48	9
277°	12	12	3	12	3	12	3	-	48	9
284°	12	12	3	12	3	12	3	-	48	9
	72	48	18	72	18	72	18	24	288	54

---

[a] L = Longitudinal

[b] T = Transverse

[c] Reference material correlation monitors

TABLE 4-13A

UNIT 1 REACTOR VESSEL SURVEILLANCE CAPSULE REMOVAL SCHEDULE

Capsule Azimuthal Position	Target Fast Neutron Fluence (x 10^{19} n/cm^2)	Projected End-of-Cycle Data
263°	0.505[a]	Withdrawn, 1979
97°	1.94[a]	Withdrawn, 1992
284°	2.33[a]	Withdrawn, 2010
83°	4.01[b]	2020
277°	5.09[c]	2032
104°	STANBY	

Notes:

[a] Actual capsule fluence, E. J. Long and J. I. Duo, "Analysis of Capsule 284° from the Calvert Cliffs Unit No. 1 Reactor Vessel Radiation Surveillance Program," WCAP-17365-NP, Revision 0, March 2011.

[b] Withdrawal criteria - Capsule fluence that corresponds to the projected fluence at the vessel inner wall location at end of extended life.

[c] Withdrawal criteria - Not less than once or greater than twice the peak end of extended life vessel fluence at the vessel inner wall ($3.86 \times 10^{19} <$ fluence in n/cm^2 $< 7.72 \times 10^{19}$). Note: This capsule also satisfies the requirement in the Nuclear Regulatory Commission safety evaluation report for Calvert Cliffs license renewal, that one capsule containing dosimetry is to be removed during the final 5 years of the extended license.

TABLE 4-13B

UNIT 2 REACTOR VESSEL SURVEILLANCE CAPSULE REMOVAL SCHEDULE

Capsule Azimuthal Position	Target Fast Neutron Fluence (x 10^{19} n/cm^2)	Projected End-of-Cycle Data
263°	0.825[a]	Withdrawn, 1982
97°	1.95[a]	Withdrawn, 1993
104°	2.44[a]	Withdrawn, 2011
83°	4.32[b]	2023
277°	5.17[c]	2033
284°	STANBY	

Notes:

a Actual capsule fluence, E. J. Long and J. I. Duo, "Analysis of Capsule 104° from the Calvert Cliffs Unit No. 2 Reactor Vessel Radiation Surveillance Program," WCAP-17501-NP Revision 0, February 2012.

b Withdrawal criteria - Capsule fluence that corresponds to the projected fluence at the vessel inner wall location at end of extended life.

c Withdrawal criteria - Not less than once or greater than twice the peak end of extended life vessel fluence at the vessel inner wall (4.28×10^{19} < fluence in n/cm^2 <8.54×10^{19}). Note: This capsule also satisfies the requirement in the Nuclear Regulatory Commission safety evaluation report for Calvert Cliffs license renewal, that one capsule containing dosimetry is to be removed during the final 5 years of the extended license.

TABLE 4-14

RCS QUALITY ASSURANCE PROGRAM

a.	Reactor Vessel		
	Forgings		
		Flanges	UT,MT
		Studs	UT,MT
		Cladding	UT,PT
		Nozzles	UT,MT
	Plates		UT,MT
		Cladding	UT,PT
	Welds		
		Main Seams	RT,MT
		CEDM Head Nozzle Connection	PT,RT[(d)],UT[(d)]
		Instrumentation Nozzles	PT,RT[(d)],UT[(d)]
		Main Nozzles to Shell	RT,MT
		Cladding	UT,PT
		Nozzle Safe Ends	RT,PT
		Vessel Support Buildup	UT,MT
		All Welds - After Hydrostatic Test	MT,PT & UT of all J-welds[(d)]
		Replacement RVCH Vent	PT,UT[(d)]
b.	RSG		
	Tube Sheet		
		Forging	UT,MT
		Cladding	UT,PT
		Weld Buildup	UT,PT
	Primary Head		
		Forging	UT,MT
		Cladding	UT,PT
	Secondary Shell and Head		
		Plates and Forgings	UT,MT
	Tubes		UT,ET
	Nozzles (Forgings)		UT,MT
	Studs (>2”)		UT,MT
	Studs (≤2”)		MT
	Welds		
		Shell, Longitudinal[(a)]	RT,MT
		Shell, Circumferential[(c)]	RT,MT,UT
		Cladding	UT,PT
		Nozzles to Shell[(a)]	RT,MT
		Tube-to-Tube Sheet	PT
		Instrument Connections	MT,RT,PT
		Temporary Attachments After Removal	MT
		All Welds – After Hydrostatic Test	MT or PT
		Nozzle Safe Ends[(a)]	RT,(MT or PT)
		Level Nozzles[(b)]	MT,RT,PT
		Vessel Support Buildup	UT,MT
		Girth Weld (Transition Weld)	RT (PT or MT)

TABLE 4-14

RCS QUALITY ASSURANCE PROGRAM

c.	Pressurizer		
	Heads		
		Plates	UT,MT
		Cladding	UT,PT
	Shell		
		Plates	UT,MT
		Cladding	UT,PT
	Heaters		
		Tubing	UT,PT
		Centering of Elements	RT
	Nozzles		UT,MT
	Studs		UT,MT
	Welds		
		Shell, Longitudinal	RT,MT
		Shell, circumferential	RT,MT
		Cladding	UT,PT
		Nozzles	RT,MT
		Nozzle Safe Ends	RT,PT
		Instrument Connections	PT
		Support Skirt	RT,MT
		Temporary Attachments After Removal	MT
		All Welds After Hydrostatic Test	MT
		Heater Assembly	RT,PT
d.	Pumps		
	Castings		RT,PT
	Forgings		UT,PT
	Welds		
		Circumferential	RT,PT
		Instrument Connections	PT
		All Welds After Hydrostatic Test	PT
e.	Piping		
	Fittings		RT,PT
	Pipe		RT,PT
	Nozzles		RT,PT
	Welds		
		Circumferential	RT,PT
		Nozzle to Run Pipe	RT,PT
		Instrument Connections	PT
		Cladding	UT,PT

RT - Radiographic; PT - Dye Penetrant; ET - Eddy Current
UT - Ultrasonic; MT - Magnetic Particle; GT - Gas Leak Test

(a) These examinations only apply to the steam drum.

(b) RT and PT are not performed for steam drum level nozzles.

(c) UT is not performed for steam drum circumferential welds.

(d) Applicable to replacement RVCH.

TABLE 4-15

RCS INSPECTION CE REQUIREMENTS[a]

REACTOR VESSEL	**CE REQUIREMENTS**	**CODE REQUIREMENT**
Ultrasonic Testing (UT)	1. UT of Weld Clad for bond	1. None
Dye-Penetrant	1. PT Test Root each 1/2 in. and Final Layer of Welds for Partial Penetration Welds to Control Element Driver Mechanism Head Adapters and Instrument Tube Connections	1. PT Test of each 1/3 weld throat or 1/2" which-ever is lesser N-462.4 (d)(l)
Replacement Steam Generator		
Ultrasonic Test	1. UT for Defects in Tube Sheet Clad	1. None
	2. UT of Weld Clad for bond	2. None
Pressurizer		
Ultrasonic Testing (UT)	1. UT clad for bond	1. None
Radiography (RT)	1. Radiograph Heaters to Check Heater Wire Positioning	1. None

[a] Replacement steam generators are manufactured by BWC.