5.2 INTEGRITY OF REACTOR COOLANT PRESSURE BOUNDARY

This section discusses the measures employed to provide and maintain the integrity of the Reactor Coolant Pressure Boundary (RCPB) throughout the facility design lifetime. The RCPB is defined in accordance with 10CFR50.2(v), to include all pressure containing components such as Pressure vessels, piping, pumps, and valves which are:

a) Part of the Reactor Coolant System (RCS), or

b) Connected to the RCS, up to and including any and all of the following:

   1) The outermost containment isolation valve in system piping that penetrates the primary containment

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

   3) The RCS primary safety valves

5.2.1 COMPLIANCE WITH CODES AND CODE CASES

5.2.1.1 Compliance With 10CFR50.55a

10CFR50.55a provides minimum dates for codes and standards applicable to the RCPB*. This regulation became effective on July 12, 1971.

The construction permit application for Waterford 3 was filed in December, 1970. Major components for the Nuclear Steam Supply System were ordered consistent with the application date and anticipated schedule. The reactor vessel, steam generators, pressurizer and piping were ordered in March, 1971 and the reactor coolant pumps, in March, 1972. The construction permit was received in November, 1974.

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*Components that are connected to the RCS and are part of the RCPB may be excluded from complying with 10CFR50.55a, provided:

a) For postulated failure of the component during normal reactor operation, the reactor can be shutdown and cooled in an orderly manner assuming makeup is provided by the Chemical and Volume Control System (CVCS).

b) The component is or can be isolated from the RCS by two valves (both closed, both open, or one closed and the other open). Each open valve must be capable of automatic actuation and its closure time must be such that for postulated failure of the component during normal reactor operation the reactor can be shutdown and cooled in an orderly manner assuming makeup is provided by the CVCS only.

The codes editions and addenda used are later than those listed in the PSAR and are listed in Table 5.2-1.

Waterford 3 meets or exceeds the in-service inspection requirements of 50.55a (g) (3) with respect to the design and access provisions as well as with respect to preservice examination requirements. The codes and addenda used for preservice examination requirements are also given in Table 5.2-1. The ability to conduct subsequent in-service inspections throughout the life of the unit, to the extent practical within the limitations of design, geometry and materials of construction of the components, is addressed in Subsection 5.2.4.

➔(DRN 00-1059, R11-A)

Section 50.55a(h) requires that protection systems meet the requirements of IEEE-279 in effect on the docket date of the application for a construction permit. Protection systems at Waterford 3 meet or exceed the requirements of 50.55a(h).

🡐(DRN 00-1059, R11-A)

➔(EC-8458, R307))

Fracture toughness standards for materials employed in pressure-retaining components of the reactor coolant pressure boundary existed prior to the order dates of all components used at Waterford. The reactor vessel, pressurizer, primary coolant piping and pumps all meet the fracture toughness requirements in effect through the Winter 1971 Addenda to the ASME Code. The replacement steam generators meet the fracture toughness requirements in effect in the 1998 Edition with 2000 Addenda of the ASME Code. Thus, longitudinal direction Charpy impact tests were performed satisfactorily for all of these components. Charpy V-notch results are described in Subsection 5.2.3.

🡐(EC-8458, R307)

The Summer 1972 Addenda and 10CFR50 Appendix G expanded testing requirements to include dropweight testing and transverse Charpy impact testing. Materials to perform such additional testing were not available.

To present the fracture toughness data as required by Appendix G to the maximum extent practical, the available test data for the reactor vessel, steam generators, pressurizer and pumps have been evaluated according to Branch Technical Position MTEB 5-2 "Fracture Toughness Requirements". This approach, which was recommended by the NRC Staff at a December, 1974 meeting, results in a downgrading of the material fracture toughness properties and provides more conservatism than if the testing actually had been performed in accordance with 10CFR50, Appendix G. The available fracture toughness data are reported in Subsection 5.2.3.

➔(DRN 00-1059, R11-A)

The methods of MTEB 5-2, which allow the development of an RT_{NDT} for materials exhibiting a fracture toughness of at least 30 ft-lbs absorbed energy, were applied.

🡐(DRN 00-1059, R11-A)

➔(DRN 02-218, R11-A)

Conservatism in the evaluations of Waterford 3 primary system pressure boundary ferritic materials has been confirmed by testing performed in accordance with Appendix G of Part 50. In addition to the Charpy impact testing conducted with test specimens prepared from longitudinal (strong direction) material, Charpy impact testing on transverse (weak direction) material and drop weight tests on base metal, welds and HAZ materials for the most limiting reactor vessel beltline material have been performed. Materials for the most limiting materials in the area beltline region were set aside and retained for purposes of performing baseline testing as part of Waterford's reactor vessel material surveillance program. LP&L elected not to wait to perform baseline testing of the limiting plate in the beltline region as is customarily done. The results of this

🡐(DRN 02-218, R11-A)

→(DRN 02-218, R11-A)
testing, when contrasted with the results of the MTEB 5-2 evaluations, demonstrate the wide margin of conservatism in our evaluation technique.
←(DRN 02-218, R11-A)

In summary, the components subject to 50.55a meet or exceed the design and construction requirements of that section as further discussed in the letters LP&L 8254, dated February 24, 1978 and LP&L 9992, dated November 10, 1978 and in all respects other than certain documentation and analyses requirements for valves which were promulgated subsequent to procurement of Waterford 3 components. Fracture toughness requirements of Appendix G have been satisfied by alternate methods of evaluation (use MTEB 5-2 and early testing of baseline surveillance materials).

Based on the above evaluations, testing and analyses, Waterford components comply with Section 50.55(a)(2)(ii) and valves, with Section 50.55(a)(2)(i).

→(EC-1020, R307)
All of the pressure retaining materials used in the fabrication of the Replacement Reactor Vessel Closure Head (RRVCH) have been tested to demonstrate compliance with the fracture toughness requirements of 10 CFR 50 Appendix G as required by the Code. All aspects of the fracture toughness (impact testing) were performed in compliance with subarticle NB-2200 and subarticle NB-2300 of the ASME Code Section III, Division 1, 1998 Edition through 2000 Addenda.
←(EC-1020, R307)

5.2.1.2 Applicable Code Cases

The code cases applied to components within the RCPB are listed in Table 5.2-2.

5.2.1.2.1 Regulatory Guide 1.84

Code cases applied to Waterford 3 are on the approved list except for the differences noted below:

a) Code Cases 1604

This code case was applied prior to the effective data of Regulatory Guide 1.84 and has been incorporated into the ASME Section III Code, Subsection NB, Paragraph NB 6223, 1974 Edition, Winter 1974 Addenda.

b) Code Case 1361-1

This code case is acceptable because the affected component was ordered to this specific revision prior to the specific approved version in the guide per Paragraph D.2.

5.2.1.2.2 Regulatory Guide 1.85

Code Cases applied to Waterford 3 are on the approved list except as noted below:

a) Code Cases 1332-4, 1332-5, 1334-6, 1344-2, and 1557

   These cases are acceptable because the affected components were ordered to these specific revisions prior to the specific approved version in the guide per Paragraph D.2.

b) Code Case 1401-1

   This code case was previously approved by the guide and has since been annulled and is acceptable per Paragraph D-3 of the guide.

c) Code Cases 1459 and 1459-1

These code cases were applied prior to the effective data of the guide and have since been incorporated into ASME Code Section III.

5.2.2 OVERPRESSURIZATION PROTECTION

5.2.2.1 Design Bases

→(DRN 03-2059, R14)

The primary safety valves on the pressurizer and the secondary safety valves on the main steam lines are designed to protect the systems from overpressure, as required by ASME Code Section III. This is documented in the ASME code report on Overpressure Protection. See Appendix 5.2A.

←(DRN 03-2059, R14)

5.2.2.2 Design Evaluation

An evaluation of the functional design of the overpressurization protection system is given in Section 15.2. In this analysis, the ability of the overpressure protection system to maintain secondary and primary operating pressures within 110 percent of design is clearly demonstrated. The analytical model used in the analysis has been documented in Section 15.2.

→(DRN 03-2059, R14)

The assumptions used in the loss of load analysis are listed in Subsection 15.2.1.3 (Loss of Condenser Vacuum). These assumptions are chosen to maximize the required relieving capacity of the primary and secondary safety valves. The analysis demonstrates that sufficient relieving capacity has been provided so that, when acting in conjunction with the reactor Protective System, the safety valves will prevent the NSSS from exceeding 110 percent of the design pressure.

The pressurizer level remains below the primary safety valve inlet as demonstrated in the Feedwater Line Break analysis in Subsection 15.2.3, which produces the greatest increase in pressurizer level.

←(DRN 03-2059, R14)

5.2.2.3 Piping and Instrumentation Diagrams

The piping and instrumentation diagram showing the primary safety valves and the associated blowdown lines are given on Figure 5.1-3. The secondary safety valves are shown on Figure 10.2-4.

5.2.2.4 Equipment and Component Description

The primary safety valves are discussed in Subsection 5.4-13. A schematic drawing of the primary safety valve is given on Figure 5.4-11. The safety valve parameters are given in Table 5.4-9. The primary safety valves are designed to withstand the following transients:

→(DRN 02-524, R12)

a) 650°F to 375°F in 50 seconds and return to 650°F in 2000 seconds for five cycles (loss of secondary pressure).

←(DRN 02-524, R12)

→(DRN 00-1059, R11-A)

b) Temperature changes of 100°F to 400°F and a return to 100°F at a rate of 100°F/hr; and simultaneous pressure changes from 400 psig to 2250 psig and returning to 400 psig in step changes. 200 cycles of this combined transient are allowed (plant leak test).

←(DRN 00-1059, R11-A)

c) ± 10°F step change from 653°F,1,030,000 cycles. (Plant loading, unloading, ± 10 percent step load, normal plant variation.)

→(DRN 06-1002, R15)

d) 75°F to 653°F and return to 75°F at a rate of 200°F/hr with pressures at saturated levels for 500* cycles. (Plant heatup and cooldown.) Heatup and cooldown are separate transients, each beginning at steady state conditions.

→(DRN 06-1002, R15)

e) Pressurize to 1.5 times set pressures at 100°F-200°F for 10 cycles plus number of hydros conducted prior to valve shipment (Hydrostatic test).

A description of overpressurization equipment and components for the Main Steam System is included in Section 10.3.

5.2.2.5 Mounting of Pressure-Relief Devices

Figure 5.2-1 provides design and installation details for the pressure relief devices mounted in the secondary side of the steam generator.

Design and installation details for the primary safety valves are provided in Subsection 3.9.3.3.

5.2.2.6 Applicable Codes and Classifications

The applicable codes and classification for the overpressurization protection system are contained in Table 3.2-1. Also see Subsections 5.4.11, 5.4.13, 10.3.1 and 10.3.6.

5.2.2.7 Material Specification

Material specifications for the overpressure protection system are given in Subsections 5.4.13 and 10.3.6.

5.2.2.8 Process Instrumentation

Figures 5.1-3 and 10.2-4 show process instrumentation for the overpressurization protection system.

5.2.2.9 System Reliability

Reliability of the main steam (secondary) safety valves is discussed in Section 10.3. The primary safety valves are passive spring-actuated mechanisms which do not fail-close if setpoint pressure is exceeded. The operational reliability of the primary safety valves is assured by:

- Compliance with ASME Code Sections III and XI for safety valves
- Conservative design criteria
- Selection of a vendor with proven experience and expertise
- Accounting for thermal cycling during valve operation
- Technical Specifications

→(DRN 06-1002, R15)

* The pressurizer is analyzed for 200 Plant heatup and cooldown cycles.

←(DRN 06-1002, R15)

5.2.2.10 Testing & Inspection

Testing and inspection of the primary safety valve is governed by ASME Section XI, Subsection IWV. Testing and inspection of the main steam safety valves is discussed in Subsection 10.3.4.

5.2.3 REACTOR COOLANT PRESSURE BOUNDARY MATERIALS

5.2.3.1 Material Specifications

A list of specifications for the principal ferritic materials, austenitic stainless steels, bolting and weld materials, which are a part of the reactor coolant pressure boundary is given in Tables 5.2-3 and 4.

To reduce sensitivity to neutron-induced changes in service, low residual requirements for copper, phosphorus, and vanadium were imposed on plate and weld materials in the reactor vessel beltline. The core beltline region, as defined by Appendix G of 10CFR50, includes the intermediate and lower shell courses and their longitudinal weld seams. Also included is the girth seam joining these two shell courses.

The chemical content of the reactor vessel beltline material as determined by chemical analysis is given in Table 5.2-5.

5.2.3.2 Compatibility with Reactor Coolant

5.2.3.2.1 Reactor Coolant Chemistry

Controlled water chemistry is maintained within the RCS. Control of the reactor coolant chemistry is the function of the Chemical and Volume Control System which is described in Subsection 9.3.4. Water chemistry limits applicable to the RCS are given in Subsection 9.3.4.

5.2.3.2.2 Materials of Construction Compatibility to Reactor Coolant

→(DRN 00-1059)

The materials of construction used in the RCPB which are in contact with reactor coolant are designated by an "a" in Table 5.2-3. These materials have been selected to minimize corrosion and have previously demonstrated satisfactory performance in other existing operating reactor plants.

5.2.3.2.3 Compatibility with External Insulation and Environmental Atmosphere

←(DRN 00-1059)

The possibility of leakage of reactor coolant onto the reactor vessel head or other part of the reactor coolant pressure boundary causing corrosion of the pressure boundary has been investigated by C-E.

Tests have shown that RCS leakage onto surfaces of the reactor coolant pressure boundary will not affect the integrity of the pressure boundary.

→(DRN 00-1059; 02-88)

The reactor vessel and closure head are insulated with stainless steel reflective insulation or Owens-Corning Fiberglas nuclear blanket type thermal insulation qualified per Reference 1, to minimize insulation contamination in the event of active solution spillage. The reactor vessel supports are not insulated. Removable panels of insulation are provided on the closure head, on the vessel lower head, and around the reactor vessel nozzles as required to allow access for in-service inspection of weld areas.

←(DRN 00-1059; 02-88)

→(DRN 00-1059, R11-A; 02-88, R11-A)

←(DRN 00-1059, R11-A; 02-88, R11-A)

In the local areas around stainless steel and nickel-based alloy nozzles in the reactor vessel head, some small plugs of mineral wool insulation encapsulated in fiber glass cloth may be used. The C-E specification for the mineral wool/fiber glass insulation limits the amount of leachable halides in accordance with Regulatory Guide 1.36 (2/23/73). The amount of mineral wool contained in these small plugs would not be sufficient to restrict the openings in the Safety Injection System sump screens.

5.2.3.3 Fabrication and Processing of Ferritic Materials

5.2.3.3.1 Fracture Toughness

Wherever possible, the tests and acceptance requirements of 10CFR50, Appendix G, were applied to the primary system pressure boundary ferritic materials, bolting and weld materials used for fabrication of the reactor vessel, steam generators (primary side), pressurizer, and 42 in. and 30 in. reactor coolant piping.

→(EC-1020, R307; EC-2800, R307)
These materials, except for the Replacement Reactor Vessel Closure Head (RRVCH) material, were ordered to earlier code requirements (see Table 5.2-1) and, therefore, some of the additional tests required by 10CFR50, Appendix G, were not performed. The CEDM fracture toughness requirements comply with 10CFR50, Appendix G with no application of BTP MTEB 5-2.
←(EC-1020, R307; EC-2800, R307)

Testing and measuring equipment for fracture toughness tests for the reactor vessel, steam generators, pressurizer, and reactor coolant pumps were calibrated in accordance with Paragraph NA-4600 of the 1971 ASME Code Section III, through Summer 1971 Addenda. Testing and measurement equipment for piping fracture toughness tests were calibrated in accordance with Paragraph NB-2360 of the 1971 ASME Code Section III through Summer 1972 Addenda.

→(DRN 00-1059, R11-A; EC-1020, R307)
To present the fracture toughness data as required by 10CFR50, Appendix G, the available test data for the reactor coolant pressure boundary materials were evaluated according to Branch Technical Position MTEB 5-2, "Fracture Toughness Requirements." The available fracture toughness data are reported in Tables 5.2-6 through 5.2-9. This approach results in a downgrading of the material fracture toughness properties and provides more conservatism than if the testing were performed in accordance with 10CFR50, Appendix G. Charpy V-notch results are shown in Figures 5.2-2 through 5.2-30, except for the RRVCH forging for which results are shown in Table 5.2-6. Footnotes in Table 5.2-6 through 5.2-9 indicate the sections of MTEB 5-2 that were used in evaluating the data. The various tests were performed in accordance with the applicable ASME Code and applicable Addenda, as noted in Table 5.2-1.
←(EC-1020, R307)

→(DRN 02-218, R11-A)
The SA 516 Gr. 70 plate material for the RCS piping meets the fracture toughness requirement of Table I-1.1, Appendix I of the Summer 1971 Addenda of the ASME Code (20 ft-lbs avg.). The methods of MTEB 5-2, which allow the development of an RT_{NDT} for materials exhibiting a fracture toughness of at least 30 ft-lbs absorbed energy were applied. Footnotes in Table 5.2-7 indicate which MTEB 5-2 sections were utilized
←(DRN 00-1059, R11-A; 02-218, R11-A)

→(DRN 00-1059, R11-A; 02-218, R11-A)
in evaluating the piping. For those plates which exhibited fracture toughness energies between 20 and 30 ft-lbs a generic basis to establish a conservative RT_{NDT} was utilized. These plates are denoted by a footnote D in Table 5.2-7. The Charpy energy absorbed vs. temperature and the mils lateral expansion vs. temperature for the Waterford 3 material and for the 25 additional heats of 516 Gr- 70 plate from Southern California Edison Co.'s San Onofre Units 2 and 3, were plotted to establish the highest temperature necessary to achieve 50 ft-lbs absorbed energy and 35 mils lateral expansion. (Figs. 5.2-29 and 5.2-30, respectively). The highest temperature necessary to achieve 50 ft-lb absorbed energy is 118°F and the highest temperature necessary to achieve 35 mils lateral expansion is 96°F. A conservative RT_{NDT} for this material is 58°F ($T_{50\ ft\text{-}lbs}$ - 60).
←(DRN 00-1059, R11-A; 02-218, R11-A)

Because reactor vessel beltline materials are subject to neutron induced changes in mechanical properties, 10CFR50 Appendix G Section 3C requires that additional fracture toughness tests be performed. These materials were not tested in full accordance with 10CFR50, Appendix G.

Testing of weld and weld heat affected zone (HAZ) materials had not been required by the applicable code to which the materials were ordered; however, additional base metal testing was conducted with test specimens prepared from longitudinal (strong direction) material. Transverse (weak direction) tests on base metal, welds, and HAZ materials for the reactor vessel beltline have been made as part of the Waterford 3 reactor vessel material surveillance program, (described in Subsection 5.3.1.6).

5.2.3.3.2 Control of Welding

→(DRN 06-872, R15)

5.2.3.3.2.1 Avoidance of Cold Cracking

←(DRN 06-872, R15)

→(DRN 00-1059, R11-A)
Waterford 3 components comply with NRC Regulatory Guide 1.50, Control of Preheat Temperature for Welding of Low Alloy Steel, May 1973, except for Part C, Paragraphs 1.b and 2. The strict interpretation of Paragraph l.b would imply that the qualification plates are an infinite heat sink that would instantaneously dissipate the heat input from the welding process. The procedure qualification consists of starting the welding at the minimum preheat temperature. Welding is continued until the maximum interpass temperature is reached. At this time, the test plate is permitted to cool to the minimum preheat temperature and the welding is restarted. Preheat temperatures utilized for low alloy steel are in accordance with Appendix D of Section III of the ASME Code. The maximum interpass temperature utilized is 500°F. This position applies to the steam generators, reactor vessels, 42 in. and 30 in. RCS piping and pressurizer.
←(DRN 00-1059, R11-A)

The paragraph 2 requirement is considered an unnecessary extension of present NSSS vendor procedures, which continue to produce low alloy steel welds meeting ASME Code Sections III and IX requirements. The requirements of Regulatory Guide 1.50 are met by compliance with Paragraph 4. The soundness of all welds is verified by ASME Code acceptable examination procedures.

With regard to Regulatory Guide 1.43 (May 1973), major RCS components are fabricated with corrosion resistant cladding on internal surfaces exposed to reactor coolant. The major portion of the material protected by cladding from exposure to reactor coolant is SA-533, Grade B, Class 1 plate which, as discussed in the Regulatory Guide, is immune to underclad cracking. Cladding performed on SA-508, Class 2 forging material is performed using low-heat input welding processes controlled to minimize heating of the base metal. Low-heat-input welding processes are not known to induce underclad cracking.

→(EC-1020, R307)
The Replacement Reactor Vessel Closure Head (RRVCH) is fabricated from SA-508 Grade 3 Class 1 forging material. This material is considered to be resistant to underclad cracking.
←(EC-1020, R307)

5.2.3.3.2.2 Compliance with Regulatory Guide 1.34

Regulatory Guide 1.34 (December 28, 1972) addresses controls to be applied during welding using the electroslag process. The electroslag process has not been used in the fabrication of any RCPB components. Therefore, the recommendations of this guide are not applicable.

5.2.3.3.2.3 Compliance with Regulatory Guide 1.71

Waterford 3 does not comply with the specific requirements of Regulatory Guide 1.71 (December 1973). Performance qualifications, for personnel welding under conditions of limited accessibility, are conducted and maintained in accordance with the requirements of ASME Code Sections III and IX. A requalification is required when (1) any of the essential variables of Section IX are changed, or (2) when authorized personnel have reason to question the ability of the welder to satisfactorily perform to the applicable requirements. Production welding is monitored for compliance with the procedures parameters and welding qualification requirements are certified in accordance with Sections III and IX. Further assurance of acceptable welds of limited accessibility is afforded by the welding supervisor assigning only the most highly skilled personnel to these tasks. Finally, weld quality, regardless of accessibility, is verified by the performance of the required nondestructive examination.

5.2.3.3.2.4 Compliance with Regulatory Guide 1.66

All tubular products used for components of the RCPB (except the three components noted below) are nondestructively examined in accordance with the requirements of ASME B&PV Code, Section III, Division 1, 1974 Edition and Addenda through Summer 1974. In addition, these nondestructive examination requirements are consistent with the recommendations of Regulatory Guide 1.66. (October, 1973)

The three components not consistent with the recommendations of Regulatory Guide 1.66 were ultrasonically tested in accordance with the requirements of the following ASME B&PV Code Addenda for the 1971 Edition: reactor vessel instrument tubing, and heater sleeve tubing - Summer 1971; CEDM upper pressure housing Winter 1973. It is considered that performing the additional ultrasonic testing examination of these components will not provide additional meaningful information on material quality commensurate with safety.

5.2.3.4 <u>Fabrication and Processing of Austenitic Stainless Steel</u>

5.2.3.4.1 Avoidance of Stress Corrosion Cracking

5.2.3.4.1.1 Avoidance of Sensitization

5.2.3.4.1.1.1 NSSS Components

Waterford 3 is consistent with the recommendations of Regulatory Guide 1.44 as described in items a through d except for the criteria used to demonstrate freedom from sensitization. The ASTM A-393 Strauss Test was used in lieu of the ASTM A-262 Practice E, Modified Strauss Test to demonstrate freedom from sensitization in fabricated, unstabilized, stainless steel.

→(EC-2800, R307)
For replacement CEDMs, the ASTM A262 Practice E test was used.
←(EC-2800, R307)

a) Solution Heat Treatment Requirements

All raw austenitic-stainless steel material, both wrought and cast, in the fabrication of the major NSSS components in the RCPB, is supplied in the annealed condition as specified by the pertinent ASTM or ASME Code; viz., 1850-2050°F for 1/2 to one hour per in. of thickness and water quenched to below 700°F. The time at temperature is determined by the size and type of component. For example, reactor coolant pump casings which are cast from CF8M are usually subject to more than one solution anneal and, therefore, the time at temperature is limited to 1/2-hour per in. of thickness.

Solution heat treatment is not performed on completed or partially fabricated components. Rather, the extent of chromium carbide precipitation is controlled during all stages of fabrication as described below.

b) Material Inspection Program

Extensive testing on stainless steel mockups, fabricated using production techniques, has been conducted to determine the effect of various welding procedures on the susceptibility of unstabilized 300 series stainless steels to sensitization-induced intergranular corrosion. Only those procedures and/or practices demonstrated not to produce a sensitized structure are used in the fabrication of these RCPB components. The ASTM standard A-393 (Strauss test) is the criterion used to determine susceptibility to intergranular corrosion. This test has shown excellent correlation with a form of localized corrosion peculiar to sensitized stainless steels. As such, ASTM A393 is utilized as a go-no-go standard for acceptability.

→(EC-2800, R307)
For replacement CEDMs, the ASTM A262 Practice E test was used.
←(EC-2800, R307)

As a result of the above tests, a relationship was established between the carbon content of 304 stainless steel and weld heat input. This relationship is used to avoid weld heat affected zone sensitization, as described below.

c) Unstabilized Austenitic Stainless Steels

The unstabilized grades of austenitic stainless steel with carbon content of more than 0.03 percent used for components of the RCS are 304 and 316. These materials are furnished in the solution annealed condition. Exposure of completed or partially fabricated components to temperatures ranging from 800°F to 1500°F is prohibited wherever possible. Exceptions may arise where valves containing stellite seats which cannot be quenched are exposed to this temperature range during cooling from hard surfacing.

Duplex, austenitic stainless steels, containing more than five weight percent delta ferrite (weld metal, cast metal, weld deposit overlay), are not considered unstabilized since these alloys do not sensitize, that is, form a continuous network of chromium-iron carbides. Specifically, alloys in this category are:

→(DRN 06-911, R15)

CF8M Cast stainless steels	}Delta ferrite controlled to
CF8 Cast stainless steels	}5-25 v/o
308	}Singly and combined stainless
309	}steel weld filler metals.
312	}Delta ferrite controlled to
316	}5-18 v/o as deposited.

←(DRN 06-911, R15)

Delta ferrite of deposited weld metal or castings exposed to the temperature range of 1000°-1500°F was determined by either a magnetic measurement, chemical analysis in conjunction with the Schaeffler Diagram, or metallographic analysis.

In duplex, austenitic/ferritic alloys, chromium-iron carbides are precipitated preferentially at the ferrite/austenitic interfaces during exposure to temperatures ranging from 1000°-1500°F. This precipitate morphology precludes intergranular penetrations associated with sensitized 300 series stainless steels exposed to oxygenated or fluoride environments.

d) Avoidance of Sensitization

→(DRN 00-1059, R11-A)

Exposure of unstabilized austenitic 3XX stainless steels to temperatures ranging from 800°-1500°F will result in carbide precipitation. The degree of carbide precipitation, or sensitization, depends on the temperature, the time at the temperature, and also, the carbon content. Severe sensitization is defined as a continuous grain boundary chromium-iron carbide network. This condition induces susceptibility to intergranular corrosion in oxygenated aqueous environments, as well as those containing fluorides. Such a metallurgical structure will rapidly fail the Strauss test ASTM A-393. Discontinuous precipitates (i.e., an intermittent grain boundary carbide network) are not susceptible to intergranular corrosion in a PWR environment.

←(DRN 00-1059, R11-A)

→(EC-2800, R307)

For replacement CEDMs, the ASTM A262 Practice E test was used.

←(EC-2800, R307)

Weld heat affected zone sensitized austenitic stainless steels (which will fail the Strauss Test, ASTM, A393) are avoided by careful control of:

- Weld heat input to less than 60 kj/in.
- Interpass temperature to a maximum of 350°F

→(DRN 00-1059, R11-A; 02-218, R11-A)

Homogeneous or localized heat treatment in the temperature range 800-1500°F is prohibited for unstabilized austenitic stainless steel with a carbon content greater than 0.03 used in components of the RCPB. When stainless steel safe ends are required on component nozzles or piping, fabrication techniques and sequencing require that the stainless steel piece be welded to the component after final stress relief. This is accomplished by welding an Inconel overlay on the end of the nozzle. Following final stress relief of the

←(DRN 00-1059, R11-A; 02-218, R11-A)

→(DRN 02-218)
component, the stainless steel safe end is welded to the Inconel overlay, using Inconel weld filler metal.
←(DRN 02-218)

5.2.3.4.1.1.2 Components Other Than NSSS

a) Regulatory Guide 1.44

With respect to other Class 1 components, Waterford 3 is consistent with the recommendations of Regulatory Guide 1.44 as described in Subsection 6.1.1 for the ESF components.

5.2.3.4.1.2 Avoidance of Contaminants Causing Stress Corrosion Cracking

5.2.3.4.1.2.1 NSSS Components

Specific requirements for cleanliness and contaminating protection are included in the equipment specifications for components fabricated with austenitic stainless steel. The provisions described below indicate the type of procedures utilized for NSSS components to provide contamination control during fabrication, shipment and storage.

Contamination of austenitic stainless steels of the 300 type by compounds which can alter the physical or metallurgical structure and/or properties of the material was avoided during all stages of fabrication. Painting of 300 series stainless steels was prohibited. Grinding was accomplished with resin or rubber-bounded aluminum oxide or silicon carbide wheels which were not previously used on materials other than austenitic-ferrite alloys. Outside storage of partially fabricated components was avoided and, in most cases prohibited. Exceptions were made with certain structures provided they were dry, completely covered with a waterproof material, and kept above ground.

Internal surfaces of completed components, are cleaned to produce an item which was clean to the extent that grit, scale, corrosion products, grease oil wax, gum, adhered or embedded dust or extraneous materials were not visible to the unaided eye. Cleaning was effected by either solvents (acetone or isotropyl alcohol or inhibited water 30-200 ppm hydrazine or 0.5-0.75 weight percent trisodium phosphate). Water conformed to the following requirements:

Halides	
Chloride (ppm)	< 0.60
Fluoride (ppm)	< 0.40
Conductivity (pmhos/cm)	< 5.0
pH	6.0-8.0
Visual clarity	No turbidity, oil or sediment

Prior to shipment, RCPB components were packaged in such a manner that they were protected from the weather, dirt, wind, water spray, and any other extraneous environmental conditions encountered during shipment and subsequent site storage. The environment within the package and/or component was maintained clean and dry. In some instances, use of a desiccant breather system was utilized. The shipment package was employed for site storage and was not removed until the component was installed within the containment. Once in the containment, with the shipping package removed, the component was maintained clean and dry, either by covering with a polyethylene cover, or placing in a clean area.

To prevent halide-induced, intergranular corrosion which could occur in aqueous environment with significant quantities of dissolved oxygen, solutions were inhibited via additions of hydrazine. Results of tests such as those documented in Reference I have proven this inhibitor to be completely effective. Operational chemistry specifications restrict concentrations of halide and oxygen, both prerequisites of intergranular attacks. (Refer to Subsection 9.3.4).

5.2.3.4.1.2.2 Components Other Than NSSS

Specific requirements for cleanliness and contamination protection are included in the equipment specifications for components fabricated with austenitic stainless steel. The provisions described in Subsection 6.1.1 also apply to the Class 1 components during fabrication, shipment and storage.

5.2.3.4.1.3 Characteristics and Mechanical Properties of Cold-Worked Austenitic Stainless Steels for RCPB Components

Cold-worked austenitic stainless steel is not utilized for components of the RCPB.

5.2.3.4.2 Control of Welding

5.2.3.4.2.1 Avoidance of Hot Cracking

a) NSSS Components

1) Interim Position MTEB 5-1 on Regulatory Guide 1.31

In order to preclude microfissuring in austenitic stainless steel welds, Waterford 3 is consistent with the recommendations of the Interim Position (Branch Technical Position of the Interim Position (Branch Technical Position MTEB 5-1) on Regulatory Guide 1.31, Control of Stainless Steel Welding except for the difference noted below.

→(EC-2800, R307)

The replacement CEDMs conform to Reg. Guide 1.31 Rev. 3, which supersedes BTP MTEB 5-1. See Section 1.8.

←(EC-2800, R307)

(a) Major RCPB Components, Excluding Reactor Coolant Pumps

→(DRN 00-1059, R11-A; 02-88, R11-A; 06-911, R15)

The delta ferrite content of A-7 austenitic stainless steel filler metal, except for 16-8-2, in the fabrication of major components of the reactor coolant pressure boundary has been controlled to 5-15 vol percent. Delta ferrite content was predicted by magnetic measurement or chemical analysis in conjunction with the Schaeffler or McKay Diagram, performed on undiluted weld deposits. In the case of the filler metal used with a non-consumable electrode processes, the delta ferrite content may have been predicted by chemical analysis of the rod, wire or consumable insert in conjunction with the stainless steel constitution diagram.

←(DRN 00-1059, R11-A; 02-88, R11-A; 06-911, R15)

→(DRN 00-1059, R11-A; 02-88, R11-A)
←(DRN 00-1059, R11-A; 02-88, R11-A)

The ferrite requirements was met for each heat, lot, or heat/lot combination of weld filler material.

(b) Reactor Coolant Pumps

The quality and structural adequacy of welds in the reactor coolant pumps were assured by the use of controls on materials, procedures, and personnel. These controls were selected to be pertinent to the component functional safety level required and generally, were imposed through the appropriate ASME B&PV Code referenced in Table 5.2-1.

2) Regulatory Guide 1.34

Regulatory Guide 1.34 is discussed in Subsection 5.2.3.3.2.2 and Appendix 3A.

3) Regulatory Guide 1.71

Regulatory Guide 1.71 is discussed in Subsection 5.2.3.3.2.3 and Appendix 3A.

b) Components Other Than NSSS

1) Regulatory Guide 1.31 is discussed in Subsection 6.1.1.

2) Regulatory Guide 1.34 is discussed in Subsection 5.2.3.3.2.2.

3) Regulatory Guide 1.71 is discussed in Subsection 6.1.1.

5.2.3.4.3 Nondestructive Examination

Nondestructive examination of tubular products is discussed in Subsection 5.2.3.3.

SECTION 5.2.3: REFERENCES

1) Topical Report OCF-1, Nuclear Containment Insulation System, on file with U.S. Nuclear Regulatory Commission.

5.2.4 INSERVICE INSPECTION AND TESTING OF REACTOR COOLANT PRESSURE BOUNDARY

→(DRN 99-0821; 06-872, R15)

An inservice inspection (ISI) program is provided for the examination of the Reactor Coolant Pressure Boundary (RCPB) components and supports defined as Code Class 1. The program reflects the principles and intent embodied in the ASME Boiler and Pressure Vessel Code, Section XI. Specific Code Editions and addenda required by 10CFR50.55a are referenced in the Pre-Service Inspection (PSI) and ISI programs. The purpose of the inservice inspection program is to periodically monitor the systems or components requiring inservice inspection in order to identify and to repair those indications which do not meet acceptance standards.

←(DRN 99-0821; 06-872, R15)

5.2.4.1 System Boundary Subject to Inspection

→(DRN 99-0821; 00-1059, R11-A)

The reactor pressure vessel, pressurizer, primary side of the steam generator, and associated piping, pumps, valves, bolting and-component supports are subjected to inspection. Standard exemptions as applicable are listed in the inservice inspection program.

←(DRN 00-1059, R11-A)

5.2.4.2 Arrangement of Systems and Components to Provide Accessibility

The layout and arrangement of the plant provides adequate working space and access for inspection of specific areas of Code Class 1 components of the RCPB. The Code Class 1 components of RCPB subject to inspection are those components defined by ASME Section XI.

←(DRN 99-0821)

Listed below are the provisions for access for examination of the RCPB:

a) Reactor Vessel and Closure Head

1) From Inside the Vessel:

→(EC-5000082400, R301)

All internals of the reactor vessel (which is an open structure offering insignificant impediment to access) are removable making the entire inner surface of the vessel, as well as the weld zones of the internal load-carrying structure attachments available for the required surface and volumetric inspections. Provisions are made in the plant design to allow for the removal and storage of all vessel internals (except the flow skirt) during inservice inspection. Ultrasonic testing of all reactor vessel welds will be in accordance with 10 CFR 50.55a, with the exception of ASME Exam Category B-A, Item No. B1.30 and B1.40 welds. Examinations of ASME Exam Category B-A, Item No. B1.30 and B1.40 (flange) welds shall meet or exceed the requirements of Regulatory Guide 1.150, Revision 1.

←(EC-5000082400, R301)

2) Closure Head

→(DRN 99-0821)

The closure head as available for inspection whenever it is removed, and its removal makes available the vessel closure flange, the flange-to-shell weld, bolt holes and ligaments, flange studs and nuts.

←(DRN 99-0821)

b) Reactor Coolant Piping

Biological shielding around the reactor coolant piping in the area of the reactor vessel is designed to afford access to the circumferential and longitudinal welds, as well as the transition piece to nozzle welds. The volumetric examinations are performed using ultrasonic techniques.

All reactor coolant piping, as well as major components, excluding the reactor vessel, is provided with removable insulation in the areas of all welds and adjacent base metal requiring examination.

The primary coolant piping has access at each side of the welds to manually examine the welds.

c) Steam Generators

Sufficient space is provided within the stay cylinder to permit inspection of the welds. A 12 in. diameter access opening in the steam generator support skirt is provided. The insulation in this area is removable to the extent of the full size of the access opening.

The steam generators have removable insulation and access at welds requiring examination. Manways are provided for those inspections which must be made internally on the-steam generator.

d) Other Components

All other components, including portions of the steam generators, pressurizer and primary piping, are accessible for manual examination from the outside surface.

The pressurizer has sufficient clearance around the shell weld seam for manual ultrasonic examination of these welds. The insulation is removable at each weld and access is provided for ultrasonic and visual examinations in the area of the bottom head and its nozzle penetrations of the pressurizer. A manway is provided for those inspections which must be made internally on the pressurizer.

→ (DRN 99-0821)

The reactor coolant pumps require inside visual examination only when a pump is disassembled for maintenance, repair, or volumetric examination. Access is provided to the motor flywheels for ultrasonic examination.

← (DRN 99-0821)

General provisions are made for removable insulation, removable shielding, installation of handling machinery, adequate personnel and equipment access space and lay down space for all temporarily removed or serviced components. Storage space for the removable insulation panels is also provided. Working room for a man is provided adjacent to each weld in order to examine all piping system welds manually.

5.2.4.3 Examination Techniques and Procedures

→ (DRN 99-0821)

Examinations include liquid penetrant or magnetic particle techniques when surface examination is specified, ultrasonic or radiographic techniques when volumetric examination is specified, and visual inspection techniques will be used to determine surface condition of components and for evidence of leakage. Specific techniques, procedures, and equipment varies with the contractor chosen to perform the inservice inspection, and will be defined in inservice inspection program. Alternative examination methods, a combination of methods, or newly developed techniques may be substituted for the methods specified as allowed by ASME Section XI.

← (DRN 99-0821)

5.2.4.4 Inspection Intervals

→(DRN 99-0821)

The examination program for the 120 month inspection interval is defined in the ISI Program. The ISI Program for all Code Class 1 systems and components is in accordance with the ASME Section XI, edition and addenda as specified in 10CFR50.55a and as amended by alternatives authorized by the NRC. Subsequent 120 month inspection intervals throughout the service life of the facility will comply, where practical with those requirements in the editions of the Code and addenda in effect 12 months prior to the start of each inspection interval.

←(DRN 99-0821)

5.2.4.5 Categories and Requirements

→(DRN 99-0821)

The inservice inspection program category and examination requirements for the Reactor Coolant Pressure Boundary complies with Section XI. Requests for relief are listed in the inservice inspection plan.

←(DRN 99-0821)

5.2.4.6 Evaluation of Results

The evaluation of nondestructive examination results, acceptance standards and documentation will be in accordance with Section XI.

→(DRN 99-0821)

5.2.4.7 System Leakage Tests

Code Class 1 systems and components are subjected to a system leakage test prior to startup following each reactor refueling outage. Operational limitations during heatup, cool-down, and system pressure testing, are specified in the plant Technical Specifications.

←(DRN 99-0821)

5.2.5 DETECTION OF LEAKAGE THROUGH REACTOR COOLANT PRESSURE BOUNDARY

The reactor coolant pressure boundary (RCPB) Leakage Detection System is designed to detect and identify abnormal leakage within the limits given in the Technical Specifications. The Leakage Detection System is capable of reliably:

a) Detecting unidentified sources of abnormal leakage as low as 1.0 gpm.

b) Identifying particular sources of abnormal leakage as low as 1.0 gpm.

→(EC-5000082424, R301)

The RCPB Leakage Detection System is consistent with the recommendations of NRC Regulatory Guide 1.45 (May 1973) with exception to regulatory position C.5; only two of the four Leakage Detection methods will meet the sensitivity requirements of Regulatory Guide position C.5. Leakage Detection System is capable of performing the functions following seismic events that do not require plant shutdown. In addition, the airborne particulate radioactivity monitoring system is designed to remain functional when subjected to the safe shutdown earthquake (SSE).

←(EC-5000082424, R301)

→(EC-19087, R305)

The guidance of Regulatory Guide 1.45 (May 2008) was used for determining the acceptability of the leakage detection instruments and monitoring program for meeting a 0.25 gpm leakage detection capability for the surge line LBB analysis under WCAP-17187-P (Reference 3).

←(EC-19087, R305)

5.2.5.1 Leakage Detection Methods

The means provided for leak detection consists of instrumentation which can detect general leakage from the reactor coolant pressure boundary. Through changes in liquid level, flow rate or radioactivity level, specific sources of leakage can frequently be identified. The various methods of detecting leakage (unidentified and identified) are discussed in the following paragraphs.

5.2.5.1.1 Sump Level and Flow Monitoring

The collection of water in the reactor cavity containment sump indicates possible reactor coolant leakage. Reactor Building floor drains and containment fan cooling unit condensate drains are routed to the sump so that water does not accumulate in areas of the containment other than the sump.

→(DRN 00-1059, R11-A; 06-250, R14-B)

Equipment and floor drains are routed through a single eight in. diameter pipe to a measurement tank and from there to the sump. A triangular notch weir is machined on the outlet side of the measurement tank. The flow through the weir causes the level of the measurement tank to correspond to the flow of water into the tank. The measurement tank is fitted with a level transmitter. The measuring tank level is a function of the flow into the tank. The level transmitter sends 4-20 ma dc signal proportional to the tank level to the main control room for signal linearization, recording, input to the plant monitoring computer and annunciator. The alarm is set at one gpm leakage flow above normal as required by the Regulatory Guide 1.45. A second alarm is set at a higher flow rate to alert the Control Room Operator of rising leakage flow. The level transmitter is non-safety-related and capable of performing its function following seismic events up to a safe shutdown earthquake per Regulatory Guide 1.45.

←(DRN 00-1059, R11-A; 06-250, R14-B)

→(DRN 04-1221, R13-A; EC-19087, R305)

A second method of containment sump monitoring utilizes the containment sump level indication to formulate in-flow leakage rates. By maintaining level in the deep pit area of the containment sump, a change in sump level can be converted to an in-leakage flowrate. The containment sump level computer point is provided on the control room Plant Monitoring Computer (PMC) which displays data from the containment sump level transmitter (SP-ILT-6705B) to calculate the level change in the sump over a specified time period. The level change in the sump is converted to a volume change based on the deep portion of the sump pit. The change in sump volume over time is used to conservatively calculate the in-leakage flow rate. The leak rate calculation is based on 10 minutes of previous level data. The calculation is performed and displayed at the PMC scan rate of once every 1 second. Therefore, the calculated computer point is performed every second and it will display a leak rate that is obtained from 10 minutes of previous data. The PMC sump data could be delayed up to 10 minutes during a sump pump run to return sump level to its normal monitoring level or after a PMC restart. The sump level computer point on the PMC is non-seismic, however, transmitter SP-ILT-6705 B is safety-related, seismic qualified and environmentally qualified. This PMC sump level computer point meets the sensitivity requirements of 0.25 gpm unidentified leakage rate in WCAP-17187-P (Reference 3) to prevent potential surge line ruptures.

←(DRN 04-1221, R13-A; EC-19087, R305)

In order to assist the operator to detect the source of leakage, the four containment fan cooler pan drains are piped to the containment sump measuring tank inlet pipe. The presence of flow in each of the drain lines is detected by six flow switches which are monitored by the plant monitoring computer. The following are possible sources of flow in the fan coolers drain:

a) Normal condensation from the containment air.

b) Steam pipe rupture.

c) Component cooling water coil rupture inside of the fan cooler enclosure.

→(DRN 00-1059, R11-A)

All of the above will be detected by the sump measuring tank input flow transmitter.

←(DRN 00-1059, R11-A)

→(DRN 04-1221, R13-A; 03-2059, R14)

5.2.5.1.2 Containment Airborne Particulate Radioactivity Monitoring

The containment atmosphere radiation monitor is designed to provide a continuous indication in the main control room of the particulate, iodine and gaseous radioactivity levels inside the containment. Radioactivity in the containment atmosphere indicates the presence of fission products due to a Reactor Coolant System leak or leakage of a contaminated secondary fluid system. This system is described in Subsection 12.3.4.High radiation level and alert status alarms are provided in the main control room. Listings of time rate of change in noble gas concentration and time for 10 percent deviation from normal are shown in Table 5.2-10 which are based on a postulated step increase in direct leakage from 0.1 gpm to one gpm at 85 percent of the original thermal rating, 0.1 percent failed fuel, at the end of a 90 day purge cycle before airborne clean-up units are operational (a 10 percent deviation is considered to be a 10 percent change in portion of the analog indicator with the total space between the position at 0.1 gpm and end of scale representing the total scale). The response times indicated represent the worst case.

←(DRN 04-1221, R13-A; 03-2059, R14)

5.2.5.1.3 Primary (Pressurizer) Safety Valves

Leakage through the primary (pressurizer) safety valves is detected by a non safety grade acoustic monitoring system that provides valve position indication and an alarm in the control room. This system is described in Subsection 1.9.23. Backup methods of determining safety valve leakage are as follows:

a) Discharge Line Temperature - Each of the primary (pressurizer) safety valve discharge lines contain a temperature detector for monitoring valve leakage. The temperature indicator (TI 107/108) and alarm for each of these temperatures are provided in the main control room. The leakage at safety valves will produce a rapidly increasing temperature indication since the discharge piping has a relatively small volume.

b) Quench Tank Water Level - Since the safety valves discharge to the quench tank, steam leaking through the valves eventually condenses in the quench tank and causes increasing water level and temperature. Level indication (LI 116) and alarm and also temperature indication (TI-116) and alarm are provided in the main control room to detect rise in water level and temperature due to steam entry into the tank.

5.2.5.1.4 Safety Injection and Shutdown Cooling System Leakage During Operation

Leakage of reactor coolant through the safety injection tank check valves (SI 215, 225, 235 and 245) can be detected by:

a) Safety Injection Tank Water Level: Leakage of reactor coolant to the safety injection tank produces a rising water level in the tank. The level of water in each Safety Injection Tank is monitored by three level transmitters. The level monitoring instrumentation for each Safety Injection Tank, provided in the main control room, consists of three level indicators (LI 311, 312, 313), (LI 321, 322, 323), (LI 331, 332, 333), (LI 341, 342, 343) and two stage alarm to annunciate high and high-high water levels.

b) Safety Injection Tank Pressure - Since the-safety injection tank is a relatively small closed volume with a nitrogen cover gas, the rising water level due to reactor coolant inflow is accompanied by an increasing tank pressure. The pressure in each Safety Injection Tank is monitored by three pressure transmitters.

The pressure monitoring instrumentation for each Safety Injection Tank, provided in the main control room consists of three pressure indicators (PI 311, 312, 313), (PI 321, 322, 323), (PI 331, 332, 333) (PI 341, 342, 343) and two stage alarm to annunciate high and high high tank pressure.

Leakage from the RCPB to the SDCS is detected by measuring the flow from the shutdown cooling relief valves SI-486 and SI-487 (See Figure 6.3-1 Sheet 2 of 2) leakage past the RCPB valves SI-651, SI-652, SI-665 and SI-666 will pressurize the shutdown cooling lines and lift SI-486 or SI-487. The discharge from the shutdown cooling relief valves SI-486 and SI-487 is directed to the containment leak measuring tank. Flow from the containment leak measuring tank is recorded and alarmed in the main control room. Since RCPB leakage to the SDCS is released to the contaiment, additional leakage detection is provided by one or more of the indications listed in Subsection 5.2.5.2 and by an increased demand for RCS makeup water.

Leakage from the RCPB to the SIS is detected by the pressure transmitters located on the low pressure side of SIS line check valves SI-217, SI-227, SI-237, and SI-247 (see Figure 6.3-1 Sheet 2 of 2), and indication is provided in the main control room by PI-319, PI-329, PI-339, and PI-349. High pressure is alarmed in the main control room.

Leakage past hot leg injection check valves 1SI-V2507 or 1SI-2509 is detected by the pressure transmitters located on the low pressure side of these valves. Indication is provided in the main control room by PI-390 and PI-391. High pressure is alarmed in the main control room.

Leakage past valves SI-618, SI-628, SI-638 and SI-648 and SI-611, SI-621, SI-631 and SI-641 is detected by loss of water level in the SI tanks. Low water level in the SIT's is indicated and alarmed in the main control room.

Leakage past SIS line second check valves SI-113, SI-114, SI-123, SI-124, SI-133, SI-134, SI-143, and SI-144, and past SIS header isolation valves SI-615, SI-616, SI-617, SI-625, SI-626, SI-627, SI-635, SI-636, SI-637, SI-645, SI-646 and SI-647 is detected by HPSI and LPSI header pressure sensors. Pressure indication is provided in the main control room by PI-306, PI-307, PI-308, and PI-309. RCPB leakage to the HPSI and LPSI system will also increase the demand for PCS makeup water.

5.2.5.1.5 Heat Exchanger

Leakage of reactor coolant through the letdown heat exchanger and reactor coolant pump seal heat exchanger and thermal barrier can be detected by any combination of the following:

→(DRN 00-1059)

a) Component Cooling Water System radiation - Heat exchanger leaks will produce in-leakage of reactor coolant and fission products into Component Cooling Water System. Such in-leakage increases the normally low radiation level in the system and can be detected by the radiation detectors (Tags No RE-CC-7050A, RE-CC-7050B) in the recirculation lines from the component cooling water heat exchangers. These detectors are indicated and alarmed both locally and in the main control room. Recording is done in the main control room. All channels are seismically qualified.

←(DRN 00-1059)

Complete dispersion of only one gallon of primary coolant throughout the volume of approximately 69,000 gallons of the component cooling water system is sufficient to cause early detectable rapid change in detector reading provided there is no residual radioactivity already present in CCW fluid. In this case the limit on detection is the transport time around the Component Cooling Water System loop. The longest time a volume of coolant leakage would have to travel before reaching the detector is 3.5 minutes. The true detection time however is based both on component cooling water radiation being directly proportional to the product of percent failed fuel and leak rate, and the amount of residual radiation already in the system. For a change in leak rate from an existing 0.1 gpm to 1.0 gpm with 0.1 percent failed fuel, the elapsed time for a 10 percent change is approximately three hours.

b) Component cooling surge tank level - Leakage of reactor coolant increases the inventory in the component cooling system, causing an increase in the surge tank level. Level switch LS-CC-7010S provides a high level alarm in the main control room. Local indication of tank water level is provided by gage glasses LG-CC-7010A and B.

5.2.5.1.6 CVCS Leakage

Intersystem leakage between the RCPB and the CVCS is not monitored since the CVCS is in operation when the RCS is pressurized, and is thus processing fluid.

However, the CVCS can be used to identify any leakage from the RCS by observing makeup flowrates to the volume control tank for the purposes of identifying gross leakage over an extended period of plant operation. Leakage can also be identified through special testing in which leak rates are monitored by detecting level changes within the volume control tank; this sort of special testing is conducted in order to identify the particular source of the RCS leak. Basically, it would involve securing the makeup source to the volume control tank, securing and sampling of the RCS or CVCS, securing boration or dilution of the RCS, and recording the difference in the water inventory of the volume control tank over a set period of time.

An important means of detecting abnormal leakage from the RCS is through measurement of the net amount of makeup flow to the system. Since all normal sources of outflow from the system such as letdown flow and coolant pump controlled bleed off are collected and recycled back to the RCS by the Chemical and Volume Control System (CVCS) described in Subsection 9.3.4, the net inventory in the RCS and CVCS under normal operating conditions will be constant. Transient changes in letdown flow rate or RCS inventory can be accommodated by changes in the volume control tank level. The net makeup to the system under zero leakage steady state conditions should be essentially zero. The makeup flow rates from CVCS is continuously monitored and recorded. Analysis of the makeup flow record over a period of steady state operation can provide detection of abnormal leakage. Any increasing trend in the amount of makeup required indicates a leak which is increasing in rate. Suddenly occurring leaks are indicated by a step increase in the amount of makeup which does not decrease as would be the case for a purely transient condition.

The maximum capacity of the Reactor Coolant Makeup System is 132 gpm (three 44 gpm charging pumps) which gives a ratio of maximum allowable leakage to makeup of 1/132.

Numerous methods for identifying intersystem leaks for the CVCS are available. These methods are exemplified below:

a) Decrease in volume control tank level via LIC226; control room alarm and indication is provided for this measurement channel.

b) Increase in charging flow to maintain pressurizer level; charging flow is monitored by F1212 which provides control room indication; pressurizer level is monitored by LRCIIOX and LRC110Y in the main control room and alarm annunciation is also provided in the main control room.

c) Regenerative heat exchanger (RHX) and letdown heat exchanger (LHX) interfaces may show increase in temperature, pressure or activity; CVCS-related instruments for RHX leakage monitoring include TIC221 (control room alarm and indication), PI212 (control room alarm and indication); CVCS-related instruments for LHX leakage monitoring include T1C223 (control room indication), TIC224 (control room alarm and indication) and PIC201 (control room indication). Increase in activity within the CCW system (interface with the LHX) is detectable by monitor within that cooling system.

5.2.5.1.7 Reactor Coolant Pump Seals

Instrumentation is provided to detect abnormal seal leakage. The reactor coolant pumps are equipped with three stages of seals plus a vapor or back-up seal as described in section 5.4. During normal operation, the Reactor Coolant System operating pressure is decreased through the three seals to approximately CVCS volume control tank pressure. The vapor or backup seal prevents leakage to the containment atmosphere and allows sufficient pressure to be maintained to direct the controlled seal leakage to the volume control tank. The vapor or backup seal is designed to withstand full Reactor Coolant System pressure in the event of failure of any or all of the three primary seals.

The following conditions are postulated to exist prior to the unlikely event of a vapor (backup) seal failure:

a) The lower, middle and upper seal have failed;

b) The excess flow check valve has closed;

c) The reactor coolant pump has been stopped;

d) The pressure at the vapor seal is Reactor Coolant System pressure.

→(EC-6256, R302)

In the event of a failure, and an excess flow condition exists through the vapor seal, with resultant pressure decrease downstream of the middle seal because of seal differential pressure. The reactor coolant pump seal pressure gives this indication. The seal temperature indicator also shows an increase in temperature and increase in seal leak-off flow to the reactor drain tank or to the containment sump via the floor Drain System and/or increase in controlled bleed-off flow to the volume control tank is indicated. Abnormal seal leakage also is indicated by an increased temperature of the component cooling water from the reactor coolant pump seal. An alarm in the controlled bleed-off line is provided for high temperature.

←(EC-6256, R302)

5.2.5.1.8 Steam Generator Tube Leakage

→(DRN 01-3692, R12)

An increase in radioactivity indicated by the condenser vacuum pump exhaust radiation monitors, the steam generator blowdown radiation monitors, and the main steam line N-16 Sodium Iodide monitors will indicate reactor coolant leakage to the secondary side. Routine analysis of steam generator water samples would also indicate increasing leakage of reactor coolant.

←(DRN 01-3692, R12)

5.2.5.1.9 Reactor Vessel Head Closure Leakage

The space between the double 0-ring seal is monitored to detect an increase in pressure, which indicates a leak past the inner 0-ring. Alarm of this condition is available in the main control room.

5.2.5.1.10 Reactor Coolant Pump Flange Closure Leakage

→(DRN 02-317, R12)

The Reactor Coolant Pump case and pump cover / driver mount is sealed by an inner and outer gasket. Reactor Coolant Pump leak-off into the annulus between these two gaskets may be aligned to pressure switches, the Reactor Drain Tank, or isolated from the pressure switches or the Reactor Drain Tank.

←(DRN 02-317, R12)

→(EC-19087, R305)

5.2.5.1.11 Control Room Leakage Monitoring

Waterford has implemented RCS unidentified leakage monitoring and action levels in accordance with the guidance of WCAP-16465, (Reference 4). The PWR Owners Group concluded that leak rate measurements can reveal small leaks (< 0.1 gpm) when data is recorded for a sufficient period of time. WCAP-16465 established RCS unidentified leakage trending and action levels for three conditions during normal plant operation. This includes monitoring absolute unidentified leak rate (in gpm), deviation from the baseline mean (in gpm), and total integrated unidentified leakage (in gallons). The absolute unidentified leak rate action levels which a direct indication of RCS unidentified leakage are established at:

- One seven (7) day rolling average of daily unidentified RCS leak rates > 0.1 gpm.
- Two consecutive daily unidentified RCS leak rates > 0.15 gpm.
- One daily unidentified RCS leak rate > 0.3 gpm.

Waterford trends RCS normal unidentified leakage at levels below 0.1 gpm. The action level of 0.1 gpm is one tenth of the TS Limit for unidentified leakage which ensures that early detection of changes in RCS unidentified leakage will be identified and addressed prior to TS limiting conditions for operation are reached.

←(EC-19087, R305)

5.2.5.2 <u>Indication in Main Control Room</u>

The primary indications of reactor coolant leakage are:

a) High containment sump flow alarm

b) Very high containment -sump flow alarm

c) Containment airborne radioactivity monitor indication (particulate and iodine and gaseous)

d) High containment particulate radioactivity alarm

→(DRN 04-1221, R13-A)

e) Deleted

f) Deleted

←(DRN 04-1221, R13-A)

Other main control room instrumentation that indicates significant reactor coolant leakage includes:
→(DRN 00-1059, R11-A)

a) Temperature detectors downstream of primary (pressurizer) safety valves (M-107/108)
←(DRN 00-1059, R11-A)

b) Primary safety valves accoustic position monitors

c) Safety injection tank level indication (LI-311/321, LI-331/341)

d) High and high-high safety injection tank levels alarm

e) Safety injection tank pressure indication and high pressure alarm

f) CCW Radiation indication

g) CCW Surge Tank water level indication (LI-CC7010A, LI-CC7010B)

h) Steam generator radiation indication

i) Condenser vacuum pumps exhaust radiation indication

i) Safety injection check valve leakage pressure indication and alarm

k) Safety injection header high pressure

5.2.5.3 Limits for Reactor Coolant Leakage

The limits for both total and unidentified leakage are described in the Technical Specifications.

5.2.5.4 Unidentified Leakage
→(EC-19087, R305)

The anticipated normal total unidentified Reactor Coolant System leakage is <0.1 gpm as discussed in Section 5.2.5.1.11.
←(EC-19087, R305)

There is no practical analytical method available by which a leak rate can be correlated with crack size. Use of mathematical models to relate reactor coolant leakage to crack size requires assumptions regarding crack geometry and the number of leak sources. If it is assumed that the total leakage is from a single source, and that the crack can be treated, for example, as a square edged orifice, then the methods of references (1) and (2) would show that a through wall crack having an equivalent diameter of approximately 0.04 to 0.05 in. would result in a one gpm leak rate at operating pressure which is the maximum allowable leakage rate from unidentified sources.

For reactor coolant piping, the material defect acceptance criteria per NB-2532.1, Section III of the ASME Code, permits an indication of up to three in. It is thus conceivable that a crack up to three in. in length could exist beneath such a laminar condition and remain undetected.

By the methods of fracture mechanics it can be shown that a through wall crack three in. in length would be approximately 12 percent of the critical crack length for an axial crack and about eight percent of the critical crack length for a circumferential crack.

5.2.5.5 Maximum Allowable Total Leakage

The maximum allowable leakage rate from unidentified sources will be limited to one gpm as specified in the Technical Specifications.
→(EC-19087, R305)

The basis for the proposed one gpm leakage rate from unidentified sources in the reactor coolant system is that this rate can be readily detected and appropriate action taken prior to constituting a potential safety hazard.
←(EC-19087, R305)

The maximum allowable total leakage rate from an identified and evaluated leak will be limited to 10 gpm as specified in the Technical Specifications. This is well within the 44 gpm capacity of one charging pump. The 10 gpm leakage rate is based upon the ability of one charging pump to makeup reactor coolant leakage and still maintain a reasonable makeup margin (34 gpm)-

5.2.5.6 Differentiation Between Identified and Unidentified Leaks

RCS leakage is categorized as identified and unidentified leakage. Identified leakage is:

a) Leakage into closed systems such as pump seal, safety valve, and valve packing leaks that are captured and directed so that their flowrates are known.

b) Leakage into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of the unidentified leakage monitoring systems or not to be from flows in the RCPB.

→(EC-5000082424, R301)

All other leakage is unidentified leakage. Since identified leakage is known, its affect upon the various leakage detection systems also is known. An increase in leakage, resulting from unidentified leakage, is detected by the leakage detection systems. At least two of the systems are capable of responding to a one gpm leakage in one hour or less.

←(EC-5000082424, R301)

The containment air particulate and radioactive gas monitors provide the primary means of remotely identifying the source of leakage within the reactor building. If sump flow indicators detect leakage above normal without a corresponding increase in airborne activity level, the indicated source of leakage probably is a nonradioactive system.

In order to identify leaks from the RCPB to the secondary side of steam generator and to locate the general area of the leak, each steam generator has a sampling system. The sampling system is tapped off the blowdown line of each steam generator. Specimens from each steam generator are analyzed for radioactivity and chemistry to determine the integrity of the primary to secondary boundary within the steam generators.

Leakage from the RCS into the Component Cooling Water System is detectable by an increase in water level in the component cooling water surge tanks.

5.2.5.7 Testing and Inspection

Preoperational testing consists of calibrating the instruments, testing the automatic controls for activation at the proper set points and checking the operability and limits of alarm functions. Radiation detectors can be remotely checked against a standard source during normal operation.

Normal leakage rates will be identified at the early stages of plant operation by the makeup water data. The normal operating levels will be compared with the identified leakage and used to verify the sensitivity of the instrumentation.

Table 5.2-11 indicates the inservice inspection that will be performed on all valves in the HPSI, LPSI and RHR systems which form the pressure boundary for the RCS.

5.2.5.8 Leakage Checks During Shutdown

Leakage of reactor coolant is checked during shutdowns in the following manner:

a) Prior to reactor startup following each refueling outage, pressure retaining components of the reactor coolant pressure boundary will be visually examined for evidence of reactor coolant leakage while the system is under a test pressure of not less than the nominal system operating pressure at rated power.

→

These examinations, which need not require removal of insulation, will be performed by inspecting the exposed surfaces and joints of insulation, and the floor areas, or equipment directly underneath these components.

←

→

b) During the conduct of these examinations, particular attention will be given to the insulated areas of components constructed of ferritic steel to detect evidence of boric acid residues resulting from reactor coolant leakage which may have accumulated during the service period preceding the refueling outage.

←

→

c) These examinations will be performed in accordance with ASME Section XI.

←

→

d) The source of any reactor coolant leakage detected by these examinations will be located by the removal of insulation where necessary and the following corrective measures applied:

←

1) Normally expected leakage from component parts (e.g., valve stems) will be minimized by appropriate repair and maintenance procedures. Where such leakage may reach the surface of ferritic components of the reactor coolant pressure boundary, the leakage will be suitably channeled away from ferritic components.

2) Leakage from through wall flaws in the pressure retaining membrane of a component shall be eliminated, either by corrective repair or by component replacement.

→

e) If boric acid residues are detected by these examinations, insulation from ferritic steel components will be removed to the extent necessary for examination of the component surface wetted by reactor coolant leakage to detect evidence of corrosion and an evaluation of the effect of any corroded area upon the structural integrity of the component will be performed in accordance with Article IWA-5250 of ASME Section XI.

←

→

f) Repairs or replacements will be performed in accordance with Article IWA-4000 of ASME Section XI.

←

SECTION 5.2.5: REFERENCES

(1) Flow of Fluids, Technical Paper No. 410, Crane Co. 1957.

(2) The Discharge of Saturated Water Through Tubes, H.K. Fauske, Chemical Engineering Progress Symposium Series, Heat Transfer Cleveland, No. 59, Vol. 61.

→(EC-19087, R305)

(3) WCAP-17187-P, "Technical Justification for Eliminating Pressurizer Surge Line Rupture as the Structural Design Basis for Waterford Steam Electric Station, Unit 3, Using Leak-Before-Break Methodology Revision 0", February 2010.

(4) WCAP-16465, "Pressurized Water Reactor Owners Group Standard RCS Leakage Action Levels and Response Guidelines for Pressurized Water Reactors", Revision 0, September 2006.

←(EC-19087, R305)

TABLE 5.2-1 (Sheet 1 of 2) Revision 309 (06/16)

CODES AND ADDENDA APPLIED

TO THE REACTOR COOLANT PRESSURE BOUNDARY

→(EC-1020, R307, LBDCR 16-007, R309) Reactor vessel (except for the reactor vessel closure head), pressurizer ←(EC-1020, R307, LBDCR 16-007, R309)	1.	ASME Boiler and Pressure Vessel Code, Section III, Class 1, through Summer 1971 Addenda
	2.	ASME Boiler and Pressure Vessel Code, Section XI, Design Access and Pre-service Inspection, through Summer 1974 Addenda
→(EC-1020, R307) Reactor vessel closure head	1.	ASME Boiler and Pressure Vessel Code, Section III Class 1, 1998 Edition through Summer 2000 Addenda
←(EC-1020, R307)	2.	ASME Boiler and Pressure Vessel Code, Section XI, Rules for Inservice Inspection of Nuclear Power Plant Components
→(EC-8458, R307) Steam Generators (primary side)	1.	ASME Boiler and Pressure Vessel Code, Section III, Class 1 1998 Edition through 2000 Addenda
←(EC-8458, R307)	2.	ASME Boiler and Pressure Vessel Code, Section XI, Design Access and Pre-service Inspection, 2001 Edition through 2003 Addenda
Reactor coolant pump fly-wheels	1.	ASME Boiler and Pressure Vessel Code, Section III, Class 1, through Winter 1971 Addenda (Ultrasonic testing only)
	2.	NRC Safety Guide 14 (Reg Guide 1.14 - October 1971)
	3.	ASME Boiler and Pressure Vessel Code, Section XI, Design Access and Pre-service Inspection, through Summer 1974 Addenda
Reactor coolant pump casing	1.	ASME Boiler and Pressure Vessel Code, Section III, Class 1, through Winter 1971 Addenda
	2.	ASME Boiler and Pressure Vessel Code, Section XI, Design Access and Pre-service Inspection, through Summer 1974 Addenda
RCS Piping	1.	ASME Boiler and Pressure Vessel Code, Section III, Class 1, through Winter 1971 Addenda
	2.	ASME Boiler and Pressure Vessel Code, Section XI, Design Access and Pre-service Inspection, through Summer 1974 Addenda

→(DRN 99-0821)

*In-service inspection will be in accordance with the Waterford 3 Steam Electric Station Inservice Inspection Plan.

←(DRN 99-0821)

TABLE 5.2-1 (Sheet 2 of 2) Revision 309 (06/16)

CODES AND ADDENDA APPLIED

TO THE REACTOR COOLANT PRESSURE BOUNDARY

Valves (NSSS)	1.	ASME Boiler and Pressure Vessel Code, Section III, Class 1, through Winter 1971 Addenda and through Summer 1972 Addenda
	2.	Draft ASME Code for Pumps and Valves for Nuclear Power, Class I, through March 1970 Addenda
	3.	ASME Boiler and Pressure Vessel Code, Section XI, Design Access and Pre-service Inspection, through Summer 1974 Addenda
Valves (Non-NSSS)	1.	ASME Boiler and Pressure Vessel Code, Section III, Class 1, through Winter 1972 Addenda
	2.	ASME Boiler and Pressure Vessel Code, Section XI, Design Access and Pre-service Inspection, through Summer 1974 Addenda
→(LBDCR 15-021, R309)	3.	ASME Boiler and Pressure Vessel Code, Section III, Class 1, 1974 through Summer 1975 Addenda
←(LBDCR 15-021, R309)		
→(EC-2800, R307) Control element drive mechanisms	1.	ASME Boiler and Pressure Vessel Code, Section III, Nuclear Power Plant Components, Class 1, 1998 Edition and 2000 Addenda
	2.	ASME Boiler and Pressure Vessel Code, Section XI, Design Access and Pre-service Inspection, 2001 Edition through 2003Addenda
←(EC-2800, R307)		
Bolting (studs and nuts)	1.	ASME Boiler and Pressure Vessel Code, Section III, Nuclear Power Plant Components, Class 1, Summer 1971 Addenda
	2.	ASME Boiler and Pressure Vessel Code, Section XI, Design Access and Pre-service Inspection, through Summer 1974 Addenda

TABLE 5.2-2 (Sheet 1 of 3) Revision 307 (07/13)

APPLICABLE CODE CASES

COMPONENT	CODE CASE	SUBJECT
Reactor Vessel	1141-1	Foreign Produced Steel.
	1332-5	Requirements for Steel Forgings, Section III and VIII, Division 2.
	1344-2	Nickel-Chromium. Age-Hardenable Alloys, (Alloy X-750) Section iii.
→(EC-1020, R307)	2142-2	F-Number Grouping for Ni-Cr-Fe Filler Metals Section IX
←(EC-1020, R307)	1401-1	Welding, Repairs to Cladding of Section III Components After Final Post Weld Heat Treatment.
	1492	Post Weld Heat Treatment Section I, III and VIII, Division I and 2.
→(EC-1020, R307)	N-698	Design Stress Intensities and Yield Strength Values for UNS N06690 with a Minimum Specified Yield Strength of 35 ksi (240 MPa)
←(EC-1020, R307)		
	1557	Steel Products Refined by Secondary Remelting.
Steam Generators	1332-4, 5	Requirements for Steel Forgings, Section III and VIII, Division 2.
	1459-1	Welding Repairs to Base Metal of Section III Components After Final Post Weld Heat Treatment
→(DRN 00-1631) Pressurizer	474-2	Design Stress Intensities and Yield Strength Values for UNS N06690
	1361-1	Socket Welds, Section III.
	2142-1	F-Number Grouping for Ni-Cr-Fe, Classification UNS N06052 Filler Metal
	2143-1	F-Number Grouping for Ni-Cr-Fe, Classification UNS W86152 Welding Electrode
←(DRN 00-1631)		

TABLE 5.2-2 (Sheet 2 of 3) Revision 307 (07/13)

APPLICABLE CODE CASES

COMPONENT	CODE CASE	SUBJECT
Reactor Coolant Pump (casing)	1604	Hydrostatic Testing of Class 1 Pumps.
→ (DRN 00-1631) Piping (Main RCS loops) ←(DRN 00-1631)	474-2	Design Stress Intensities and Yield Strength Values for UNS N06690
	1332-6	Requirements for Steel Forgings, Section III and V Division 2
	1401-1	Welding Repairs to Cladding of Section III Components After Final Post Weld Heat Treat-ment.
	1459	Welding Repairs to Base Metal of Section III Components After Final Post Weld Heat Treat-ment.
→(DRN 00-1631)	2142-1	Number Grouping for Ni-Cr-Fe, classification UNS N06052 Filler Metal
←(DRN 00-1631) →(EC-2800, R307) CEDM	N-4-12	Special Type 403 Modified Forgings or Bars, Class 1 and CS, Section III, Division 1.
	2142-2	F-Number Grouping for Ni-Cr-Fe, Filler Metals, Section IX
←(EC-2800, R307) →(EC-14300, R303) Valves	N-282	Nameplates for Valves, Section III, Division 1, Class 1, 2 and 3 Construction.
	N-24 (1516-2)	Welding of Seats or Minor Internal Permanent Attachments in Valves for Section III Applications
←(EC-14300, R303) Piping and Supports	N-242-1	Material Certification, Section III, Division 1, Class 1, 2, 3, MC and CS Construction.

→(DRN 06-552, R15)

TABLE 5.2-2 (Sheet 3 of 3) Revision 303 (06/09)

APPLICABLE CODE CASES

COMPONENT	CODE CASE	SUBJECT
	N-316	Alternate Rules for Fillet Weld Dimensions for Socket Welded Fittings, Section III, Division I, Class 1, 2 and 3.
	N-122	Procedure for Evaluation of the Design of Rectangular Cross Section Attachments on Class 1 Piping, Section III, Division 1
	N-318	Procedure for Evaluation of the Design of Rectangular Cross Section Attachments on Class 2 or 3 Piping, Section III, Division 1
	N-391	Procedure for Evaluation of the Design of Hollow Circular Cross Section Welded Attachments on Class 1 Piping, Section III, Division 1
	N-392	Procedure for Evaluation of the Design of Hollow Circular Cross Section Welded Attachments on Classes 2 and 3 Piping, Section III, Division 1

←(DRN 06-552, R15)

TABLE 5.2-3 (Sheet 1 of 4) Revision 307 (07/13)

REACTOR COOLANT SYSTEM MATERIALS

Component	Material Specification
Reactor vessel	
Shell	SA-533 Grade B, Class I Steel
(DRN 05-1400, R14-A; EC-1020, R307)	
Forgings (except for closure head)	SA-508 Class I or II
Closure Head	SA-508, Grade 3, Class I
Cladding[a]	Weld deposited austenitic stainless steel with greater than 5% delta ferrite or NiCrFe alloy
(DRN 05-1400, R14-A)	
Reactor vessel head[a] CEDM Nozzles	SB-166
Instrument nozzles[a]	SB-166 and SA-182, F-304
(EC-1020, R307)	
Control element drive mechanism housings	
(DRN 05-1400, R14-A; EC-2800, R307)	
Lower [a]	Type 403 Modified Stainless Steel per Code Case N-4-12 Condition 2, with end fittings to SB-166 and upper end fittings to SA-182, F348.
Upper [a]	SA-213 Type 316 stainless steel with end fittings of SA-479 Type 316, vent valve seal of ASTM A276 Type 440C stainless steel seat
(DRN 05-1400, R14-A; EC-2800, R307)	
Closure head bolts	SA-540 B24
Pressurizer	SA-533 Grade B Class I
(DRN 05-1400, R14-A)	
Shell Cladding[a]	Weld deposited austenitic stainless steel with greater than 5 percent delta ferrite or NiCrFe alloy
(DRN 05-1400, R14-A)	
(DRN 00-1631)	
Shell [a]	A gap exists between the original Inconel 600 and replacement Inconel 690 materials on the repaired instrument nozzles and heater sleeves.
(DRN 00-1631)	
Forged nozzles	SA-508 Class II
Instrument nozzles [a]	SB-166
(DRN 00-1631; 05-892, R14)	
Surge and safety valve nozzle safe ends	SA-351, Grade CF8M
Heater sleeves[b]	SB-167 / SB-166
(DRN 05-1400, R14-A)	
Heater sleeve plug / cap	SB-167 / SA479 TP304
(DRN 00-1631; 05-892, R14; 05-1400, R14-A)	

(DRN 05-892, R14; 06-911, R15)

(a) Materials exposed to reactor coolant

(b) Half-sleeve repair. A remnant of the original sleeve is left in-place

(DRN 05-892, R14; 06-911, R15)

REACTOR COOLANT SYSTEM MATERIALS

→(DRN 06-911, R15)
←(DRN 06-911, R15)

→(DRN 05-1400, R14-A)

Component	Material Specification
Studs and nuts	SA-540 Grade B24 and SA-193 Grade B7
→(EC-8458, R307)	
Steam generator	
Primary head	SA-508 Grade 3 Class 2 (forging)
Primary nozzles and safe ends	SA 508 Grade 3 Class 2
←(DRN 05-1400, R14-A)	
Primary head cladding[(a)]	Weld deposited Stainless Steel with less than 0.10% Cobalt
→(DRN 05-1400, R14-A)	
Tubesheet	SA-508 Grade 3 Class 2
Structural Divider Plate	SG-168 Alloy UNS N06690
Tube Support Plates	Type 405 Ferritic Stainless Steel
Tubesheet cladding [(a)]	Weld deposited Alloy 690 with less than 0.10% Cobalt
←(DRN 05-1400, R14-A)	
Tubes [(a)]	SB-163 Alloy 690 TT
→(DRN 05-1400, R14-A)	
Secondary shell and head	SA 508 Grade 3 Class 2
Secondary nozzles	SA 508 Grade 3 Class 2
←(DRN 05-1400, R14-A)	
Secondary instrument nozzles	SA-508 Grade 1A
Studs and nuts	SA-193 Grade B7 or SA-194 Grade 7
Hydranuts	SA-540 Grade B23 Class 3
→(EC-8458, R307)	
Reactor coolant pumps	
Casing [(a)]	SA-351 GR CF8M
Pump Cover (Lower Flange of Driver Mount)	SA-105
Cladding [(a)]	Austenitic Steel Wire Electrodes Conforming to Requirements of ASME/AWS SFA/A-5.4 and SFA/A-5.9 Type 308 or 309.
→(DRN 05-1400, R14-A)	
Bolts	SA 540 Gr B23 Class 4 SA-564, Type 630, H-1100 (For seal cartridge and seal heat exchanger)

←(DRN 05-1400, R14-A)

TABLE 5.2-3 (Sheet 3 of 4) Revision 14-A (03/06)

REACTOR COOLANT SYSTEM MATERIALS

Component	Material Specification
Nuts	SA 194 Gr 7 SA 564, Type 630, H-1100 (For seal cartridge and seal heat exchanger)
→(DRN 05-1400, R14-A)	
Heat Exchanger Flange	SA 240 Tp 304 Annealed or SA-182 F304
Reactor Coolant Piping	
Piping (30" and 42")	SA-516 Grade 70* (SA-264 Clad Plate)*
←(DRN 05-1400, R14-A)	
Cladding [(a)]	SA-240 - 304L
Surge Line (12") [(a)]	SA-351 - CF8M
Piping [(a)]	
Pressurizer spray	SA-376, TP-304
Shutdown Cooling Return	SA-376, TP-304
Reactor coolant drain	SA-376, TP-316 or TP-304
→(DRN 05-1400, R14-A)	
Charging line	SA-376, TP-304
←(DRN 05-1400, R14-A)	
Safety injection	SA-376, TP-304
Letdown line	SA-376, TP-316 or TP-304
Shutdown cooling bypass	SA-358, TP-304
Piping nozzles and safe ends [(a)]	
→(DRN 05-1400, R14-A)	
Piping safe ends (30")	SA-351 – Grade CF8M
←(DRN 05-1400, R14-A)	
Surge nozzle forging	SA-105 Grade II
→(DRN 05-1400, R14-A)	
Surge nozzle safe end	SA-351 – Grade CF8M
←(DRN 05-1400, R14-A)	
Shutdown cooling outlet nozzle forgings	SA-105-Grade II
→(DRN 05-1400, R14-A)	
Shutdown cooling outlet nozzle safe ends	SA-351 – Grade CF8M

*Filler metal used for Field Welds P1OW1 and P1OW2 have been rated with a strength level of 65 ksi per CE Analytical Evaluation Report CENC-1460.

←(DRN 05-1400, R14-A)

TABLE 5.2-3 (Sheet 4 of 4) Revision 15 (03/07)

REACTOR COOLANT SYSTEM MATERIALS

Component	Material Specification
Safety injection nozzle forgings	SA-182 - F1
→(DRN 05-1400, R14-A)	
Safety injection nozzle safe ends	SA-351 – Grade CF8M
←(DRN 05-1400, R14-A)	
Charging inlet nozzle forging	SA-182 - F1
Charging inlet nozzle safe end	SA-182 - F316
Spray nozzle forgings	SA-105 Grade II
Spray nozzle safe ends	SA-182 - F316
Letdown and drain or drain nozzle forgings	SA-105 Grade II
Letdown and drain or drain nozzle safe ends	SA-182 - F316
Sampling or pressure measurement nozzles	SB-166
Sampling or pressure measurement nozzle safe ends	SA-182 - F316
→(DRN 05-1400, R14-A)	
RTD nozzles	SB-166 and SA-182 F316
Sampling nozzle (surge line)	SA-182-F316
Valves [(a)]	
Body	SA-182 F316, SA-479 Type 316 and SA-351 Grade CF8M
Bonnet	SA-105 Grade II, SA-351 Grade CF8M, SA-479 Type 316, SA-240 Type 316 and SA-182 F316
Disc or Poppet	SA-637 Grade 688, SA-240 Type 316, SA-479 Type 316, SA-182 F316, SA-351 Grade CF8M, SA-351 Grade CF3 and SA-564 Grade 630
←(DRN 05-1400, R14-A)	

→(DRN 03-1707, R13; 06-720, R15)

←(DRN 03-1707, R13; 06-720, R15)

TABLE 5.2-4 (Sheet 1 of 2) Revision 307 (07/13)

WELD MATERIALS FOR REACTOR COOLANT PRESSURE BOUNDARY COMPONENTS

→(DRN 05-1400, R14-A)

Material Specification	Base Material	Weld Material
1. SA-533 Grade B Class 1	SA-533 Grade B Class 1	a. SFA 5.5, (b), E-8018, C3 b. MIL-E-18193, B-4
2. SA-508 Class 2	SA-533 Grade B Class 1	a. SFA 5.5, E-8018, C3 b. MIL-E-18193, B-4
3. SA-508 Class 1	SA-508 Class 2	SFA 5.5, E-8018, C3
4. SA-516 Grade 70	SA-516 Grade 70	SFA 5.1, E-7018 (c)
5. SA-182 F1	SA-516 Grade 70	SFA, 5.1, E-7018
6. SA-105 Grade II	SA-351 CF8M	SFA 5.11, ENiCrFe-3
7. SA-182 F1	SA-351 CF8M	SFA 5.11, ENiCrFe-3
8. SA-105 Grade II	SA-182 F316	SFA 5.11, ENiCrFe-3
9. SB-166	SA-182 F316	Root SFA 5.14, ERNiCr-3 Remaining SFA 5.11, ENiCrFe-3
10. SB-167	SA-182 F304	Root SFA 5.14, ERNiCr-3 Remaining SFA 5.11, ENiCrFe-3
11. SA-516 Grade 70	SA-351 CF8M	SFA 5.11, ENiCrFe-3
12. SA-182 F1	SA-182 F316	SFA 5.11, ENiCrFe-3
13. SB-166	SA-533 Grade B Class 1	SFA 5.11, ENiCrFe-3
→(EC-2800, R307)		
14. SA-182 Code Case N-4-12	SB-166	SFA 5.14, ERNiCrFe-7A

←(DRN 05-1400, R14-A; EC-2800, R307)

b) Special weld wire with low residual elements of copper and phosphorus is specified for the beltline region.

c) Filler metal used for Field Welds PIOWI and PIOW2 have been rated with a strength level of 65 ksi per CE Analytical Report CENC-1460.

TABLE 5.2-4 (Sheet 2 of 2) Revision 307 (07/13)

WELD MATERIALS FOR REACTOR COOLANT PRESSURE BOUNDARY COMPONENTS

→(DRN 05-1400, R14-A)

Material Specification	Base Material	Weld Material
→(EC-1020, R307)		
15. SA-516 Grade 70	SA-508 Class 2	a. SFA 5.1, E-7018 b. MIL-E-18193, B-4
←(EC-1020, R307)		
16. Austenitic stainless steel cladding		SFA 5.9, ER-308 SFA 5.9, ER-309 SFA 5.9, ER-312
17. Inconel	Inconel	SFA 5.11, ENiCrFe-3 SFA 5.14, ERNiCr-3
18. SA-182 F-316	SA-508 Class 2	SFA 5.11, ENiCrFe-3
19. SA-351 CF8M	SA-508 Class 2	SFA 5.11, ENiCrFe-3
←(DRN 05-1400, R14-A)		
→(EC-1830, R302)		
20. Inconel ←(EC-1830, R302)	Varies(d)	SFA 5.14, ERNiCrFe-7A
→(EC-2800, R307)		
21. SA-182 Code Case N-4-12	SA-182 F348	SFA 5.14, ERNiCrFe-7A
22. SA-479	SA-213 F316	SFA 5.9, ER316L
←(EC-2800, R307)		
→(EC-1020, R307)		
23. SA-508 Grade B Class 1	SB-166 UNS N06690	SFA 5.14 ERNiCrFe-7A
←(EC-1020, R307)		

→(DRN 00-331, R11)
When welding SB-166 N06690 or SB-167, SB-06690 base materials, ERNiCrFe-7 and EniCrFe-7 weld materials may be substituted for ERNiCr-3 and EniCrFe-3.
←(DRN 00-331, R11)

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b) Special weld wire with low residual elements of copper and phosphorus is specified for the beltline region.

→(DRN 05-1400, R14-A)
c) Filter metal used for Field Welds P1OW1 and P1OW2 have been rated with a strength level of 65 ksi per CE Analytical Report CENC-1460.
←(DRN 05-1400, R14-A)
→(EC-1830, R302)
d) Filler metal used for structural weld overlays of Reactor Coolant System dissimilar metal welds.
←(EC-1830, R302)

TABLE 5.2-5

CHEMICAL ANALYSES OF PLATE MATERIAL IN WATERFORD 3
REACTOR VESSEL BELTLINE

	INTERMEDIATE SHELL PLATE			LOWER SHELL PLATE		
Heat No.	NR-56488-1	NR-56512-1	NR-56484-1	NR-57326-1	NR-57286-1	NR-57359-1
Code No.	M-1003-1	M-1003-2	M-1003-3	M-1004-1	M-1004-2	M-1004-3
Element	(W/O)					
si	0.22	0.24	0.20	0.21	0.23	0.22
s	0.010	0.007	0.009	0.008	0.005	0.007
p	0.004	0.006	0.007	0.006	0.005	0.007
Mn	1.34	1.36	1.41	1.41	1.38	1.42
c	0.20	0.19	0.19	0.19	0.23	0.22
Cr	0.05	0.04	0.05	0.03	0.01	0.02
Ni	0.71	0.67	0.70	0.62	0.58	0.62
Mo	0.57	0.59	0.58	0.56	0.57	0.56
B	<.001	<.001	<.001	<.001	<.001	<.001
Cb	<.01	<.01	<.01	<.01	<.01	<.01
Ti	<.01	<.01	<.01	<.01	<.01	<.01
Co	0.033	0.029	0.033	0.010	0.009	0.009
Cu	0.02	0.02	0.02	0.03	0.03	0.03
Al	0.012	0.014	0.013	0.013	0.016	0.012
N 2	01.011	0.009	0.010	0.009	0.009	0.010
v	0.002	0.003	0.002	0.002	0.002	0.002
w	<.01	<.01	<.01	<.01	<.01	<.01
AS	0.065	0.061	0.072	0.020	0.018	0.016
Sn	0.001	0.001	0.001	0.002	0.002	0.003
Zr	<.001	<.001	<.001	<.001	<.001	<.001
Sb	0.0026	0.0011	0.0015	0.0018	0.0015	0.020
Pb	<.001	<.001	<.001	<.001	<.001	<.001

TABLE 5.2-6 (Sheet 1 of 2) **Revision 307 (07/13)**

WATERFORD UNIT 3 REACTOR VESSEL FRACTURE TOUGHNESS DATA

Piece Number	Drawing Number	Code Number	Material	Vessel Location	Drop Weight NDTT(°F)	RT_{NDT} (A) (°F)	Charpy 30 ft-lb. Fix Temp.(°F) Long.	Charpy 50 ft-lb. Fix Temp (°F) Long.	35 Mils Lateral Expansion Temp. (°F) Long	Charpy Upper Shelf Energy (ft-lb) Long.
→(EC-1020, R307)										
N/A	N149327	N/A	SA508 Gr. 3,Cl.1	Closure Head Forging	-44	-44(B)	-94	-75	-75	282(c)
←(EC-1020, R307)										
126-101	741701 6103	M-1001-1	SA508 CL-II	Vessel Flange	20	20	-20	-5	2	154
131-102A	741701 6103	M-1013-1	SA508 CL-I	Safe End	10	10	-35	0	-15	148
131-102D	741701 6103	M-1013-2	SA508 CL-I	Safe End	10	10	-35	0	-15	148
131-102C	741701 6103	M-1013-3	SA508 CL-I	Safe End	10	10	-27	0	-32	149
131-102B	741701 6103	M-1013-4	SA508 CL-I	Safe End	10	10	-27	0	-30	149
131-101A	741701 6103	M-1012-1	SA508 CL-I	Safe End	10	10	0	25	-5	146
131-1O1B	741701 6103	M-1012-2	SA508 CL-I	Safe End	10	10	0	25	-5	146
128-301	741701 6103	M-1011-1	SA508 CL-II	Outlet Nozzle	-20	-20	-37	0	5	99
128-101	741701 6103	M-1010-1	SA508 CL-II	Inlet Nozzle	20	20	-37	10	0	135
128-101	741701 6103	M-1010-2	SA508 CL-II	Inlet Nozzle	20	20	-50	-35	-40	140
128-101	741701 6103	M-1010-3	SA508 CL-II	Inlet Nozzle	10	10	-70	-47	-42	133
128-101	741701 6103	M-1010-4	SA508 CL-II	Inlet Nozzle	30	30	-40	-20	-30	140
128-301	741701 6103	M-1011-2	SA508 CL-II	Outlet Nozzle	0	0	-30	-10	-12	188
124-102	741701 6103	M-1003-1	SA533-B CL-I	Intermediate Shell Plate	-30	-30	-30	-10	-10	144
124-102	741701 6103	M-10032	SA533-B CL-I	Intermediate Shell Plate	-50	-50	-55	-12	15	149
124-102	741701 6103	M-1003-3	SA533-B CL-I	Intermediate Shell Plate	-50	-42	-22	-12	10	138
122-102	741701 6103	M-1002-1	SA533-B CL-I	Upper Shell Plate	-40	-8	13	32	23	151
122-102	741701 6103	M-1002-2	SA533-B CL-I	Upper Shell Plate	-20	-20	-20	12	15	128
122-102	741701 6103	M-1002-3	SA533-B CL-I	Upper Shell Plate	-40	-40	-20	0	0	153
154-102	741701 6103	M-1007-1	SA533-B CL-1	Bottom Head Torus	-80	-80	-72	-62	-60	174

TABLE 5.2-6 (Sheet 2 of 2) Revision 307 (07/13)

WATERFORD UNIT 3 REACTOR VESSEL FRACTURE TOUGHNESS DATA

Piece Number	Drawing Number	Code Number	Material	Location Vessel	Drop Weight NDTT(°F)	RT_{NDT} (A) (°F)	Charpy 30 ft-lb Fix Temp. (°F) Long.	Charpy 50 ft-lb Fix Temp. (°F) Long.	35 Mils Lateral Expansion Temp. (°F) Long.	Charpy Upper Shelf Energy (ft-lb) Long.
152-101	741701 6103	M-1008-1	SA533-B CL-I	Bottom Head Dome	-40	-40	-35	-10	-15	141
➔(EC-1020, R307)										
➔(EC-1020, R307)										
142-101	741701 6103	M-1004-1	SA533-B CL-I	Lower Shell Plate	-50	-15	10	25	20	163
142-101	741701 6103	M-1004-2	SA533-B CL-1	Lower Shell Plate	-20	22	37	62	55	144
142-101	741701 6103	M-1004-3	SA533-B CL-I	Lower Shell Plate	-50	-10	12	30	25	145
➔(EC-1020, R307)										
➔(EC-1020, R307)										

(A)-MTEB Position 5-2 "Fracture Toughness Requirements," Paragraph 1.1(3)(b).

➔(EC-1020, R307)
(B) – RT_{NDT} per NB-2300 of Section III of the ASME B&PV Code, 1998 Edition through 2000 Addenda.

(C) – Charpy Upper Shelf Energy (Transverse) = 263 ft-lb.
➔(EC-1020, R307)

TABLE 5.2-7 (Sheet 1 of 2)

WATERFORD UNIT 3 PIPING MATERIALS FRACTURE TOUGHNESS DATA

Piece Number	Drawing Number	Code Number	Material	Location	Drop Weight NDTT(F)	RT_{NDT} °(F)	Test Temp °(F)	Charpy Energy (ft-lbs) 1 2 3 Avg	Lat- Exp- (Mils) 1 2 3 Avg.
722-108	74470-761-001	M2804-3	SA516GR70	Straight Seg.	NA	30[B]	+10	40 40 40 40	32 33 33 32.7
722-108	74470-761-001	M2804-5	SA516GR70	Straight Seg.		30[B]	+10	40 40 40 40	32 33 33 32.7
722-108	74470-761-001	M2804-6	SA516GR70	Straight Seg.		30[B]	+10	40 40 40 40	32 33 33 32.7
722-108	74470-761-001	M2804-8	SA516GR70	Straight Seg.		30[B]	+10	40 40 40 40	32 33 33 32.7
742-108	74470-761-001	M2808-1	SA516GR70	Elbow Seg.		58[D]	+10	22 20 28 23	19 20 26 21.7
742-108	74470-761-001	M2808-2	SA516GR70	Elbow Seg.		30[B]	+10	46 40 28 38	34 30 40 34.7
742-108	74470-761-001	M2808-3	SA516GR70	Elbow Seg.		58[D]	+10	24 26 21 23.7	26 17 20 21
742-104	74470-761-001	M2806-1	SA516GR70	Elbow Seg.		30[B].	+10	31 52 35 39.3	32 34 49 38.3
722-108	74470-761-001	M2804-1	SA516GR70	Straight Seg.		30[B]	+10	40 40 40 40	32 33 33 32.7
722-102	74470-761-001	M2801-3	SA516GR70	Straight Seg.		30[B]	+10	37 33 44 38	43 36 35 38
722-102	74470-761-001	M2801-4	SA516GR70	Straight Seg.		30[B]	+0	37 33 44 38	43 36 35 38
722-102	74470-761-001	M1406-1	SA516GR70	Straight Seg.		10[A]	+10	62 52 105 73	46 55 82 61
722-102	74470-761-001	M2801-1	SA516GR70	Straight Seg.		58[A]	+10	24 22 21 22.3	20 20 20 20
722-102	74470-761-001	M2801-2	SA516GR70	Straight Seg.		58[D]	+10	24 22 21 22.3	20 20 20 20
742-102	74470-742-001	M2805-1	SA516GR70	Elbow Seg.		58[D]	10	30 23 32 28.3	26 31 30 29
742-106	74470-742-001	M2807-1	SA516GR70	Elbow Seg.		58[D]	+10	25 28 28 27	21 22 22 21.7
742-108	74470-761-001	M2808-4	SA516GR70	Elbow Seg.		58[D]	+10	25 26 20 23.7	26 21 27 24.7
722-108	74470-761-001	M2804-2	SA516GR70	Straight Seg.		58[D]	+10	32 28 24 28	32 22 28 27.3
722-108	74470-761-'001	M2804-4	SA516GR70	Straight Seg.		58[D]	10	32 28 24 28	32 22 28 27.3
722-108	74470-761-001	M2804-7	SA516GR70	Straight Seg.		58[D]	+10	32 28 24 28	32 22 28 27,3
722-108	74470-761-001	M2809-1	SA516GR70	Elbow Seg.		58[D]	+10	25 25 31 27	23 22 28 24.3
722-104	74470-761-002	M2802-2	SA516GR70	Straight Seg.		58[D]	+10	38 30 16 28	38 18 30 28.7

A MTEB Position 5.2, "Fracture Toughness Requirements" Paragraph 1.1 (4) Minimum of 3 tests at a single temp. > 45 ft-lbs.
B-MTEB Position 5.2, "Fracture Toughness Requirements" Paragraph 1.1 Minimum of 3 tests at a single temp. > 30 45<ft-lbs.
C-MTEB Position 5.2, "Fracture Toughness Requirements" Paragraph 1.1 (3)(b).
D-Subsection 5.2.3.3.1

TABLE 5.2-7 (Sheet 2 of 2)

WATERFORD UNIT 3 PIPING MATERIALS FRACTURE TOUGHNESS DATA

Piece Number	Drawing Number	Number	Material	Location	Drop Weight NDTT(°F)	RT_{NDT} (°F)	Test Temp (°F)	Charpy Energy (ft-lbs) 1 2 3 Avg.	Lat. Exp.(Mils) 1 2 3 Avg.
722-106	74470-761-002	M2803-2	SA516GR70	Straight Seg.	NA	+58[D]	+10	38 30 16 28	38 18 30 28.7
722-104	74470-761-002	M2802-3	SA516GR70	Straight Seg.		+30[B]	+10	52 56 43 50.3	50 48 42 46.7
722-106	74470-761-002	M2803-3	SA516GR70	Straight Seg.		+30[B]	+10	52 56 43 50.3	50 48 42 46.7
722-104	74470-761-002	M2802-4	SA516GR70	Straight Seg.		+58[D]	+10	26 33 30 29.7	31 34 33 32.7
722-106	74470-761-002	M2803-4	SA516GR70	Straight Seg.		+58[D]	+10	26 33 30 29.7	31 34 33 32.7
722-104	74470-761-002	M2802-1	SA516GR70	Straight Seg.		+30[B]	+10	47 41 47 45	44 44 38 42
722-106	74470-761-002	M2803-1	SA516GR70	Straight Seg.		+30[B]	+10	47 41 47 45	44 44 38 42
728-301	D-728-003-06	M2810-2	SA182F1	Safety Injection Nozzle Forg.		20[A]	-20	74 70 90 78	54 51 66 57
728-301	D-728-003-06	M2810-3	SA182F1	Safety Injection Nozzle Forg.		20[A]	-20	74 70 90 78	54 51 66 57
728-301	D-728-003-06	M2810-4	SA182F1	Safety Injection Nozzle Forg.		0[B]	-20	50 38 67 51.7	40 33 53 42
728-201	C-728-002-00	M2813-1	SA105-2	Shutdown Cooling Outlet Nozzle		+30[B]	+10	36 55 39 43.3	40 56 41 45.7
728-201	C-728-002-00	M2813-2	SA105-2	Shutdown Cooling Outlet Nozzle		+30[B]	+10	35 57 35 42.3	41 56 40 45.7
728-501	D-728-005-00	M2811-1	SA182F1	Charging Inlet Nozzle Forg.		-20[A]	-20	70 83 75 76	53 59 56 56
728-501	D-728-005-00	M2811-2	SA182F1	Charging Inlet Nozzle Forg.		-20[A]	-20	70 83 75 76	53 59 56 56
728-301	D-728-003-00	M2810-1	SA182F1	Safety Injection Nozzle Forg.		-20[A]	-20	55 67 74 65.3	43 53 54 50
728-401	C-728-004-00	M2814-1	SA105-2	Spray Noz. Forg-		+30[B]	+10	227 43 42 101.	89 41 39 56.3
728-401	C-728-004-00	M2814-2	SA105-2	Spray Noz.Forg.		+3O[B]	+10	227 43 42 101.	89 41 39 56.3
738-601	C-728-006-01	M2815-1	SA105-2	Letdown and Drain Nozzle Forg.		+30[B]	+10	227 43 42 101.	89 41 39 56.3
728-601	C-728-006-01	M2815-2	SA105-2	Letdown and Drain Nozzle Forg.		+30[B]	+10	227 43 42 101.	89 41 39 56.3
728-601	C-728-006-01	M2815-3	SA105-2	Letdown and Drain Nozzle Forg.		+30[B]	+10	227 43 42 101.	89 41 39 56.3
728-601	C-728-006-01	M2815-4	SA105-2	Letdown and Drain Nozzle Forg.		+30[B]	+10	227 43 42 101.	89 41 39 56.3
728-701	C-728-007-01	M2816-1	SA105-2	Drain Nozzle Forg.		+30[B]	+10	227 43 42 101.	89 41 39 56.3

A-MTEB Position 5.2 "Fracture Toughness Requirements" Paragraph 1.1 (4), Greater Than 45 ft-lbs.
B-MTEB Position 5.2 "Fracture Toughness Requirements" Paragraph 1.1 Less Than 45 ft-lbs, More Than 30 ft-lbs.
C-MTEB Position 5.2 "Fracture Toughness Requirements" Paragraph 1.1 (3)(b)
D-Subsection 5.2.3.3.1

TABLE 5.2-8

WATERFORD 3 PRESSURIZER MATERIALS FRACTURE TOUGHNESS DATA

Piece Number	Drawing Number	Code Number	Material	Location	Drop Weight NDTT (°F)	RT_{NDT} (°F)	Test Temp (°F)	Charpy Energy (ft-lbs.) at 0 Position 1	2	3	Avg.	Charpy Energy (ft-lbs.) at 180 Position 1	2	3	Avg.	Lat. Exp. (mils at O Position 1	2	3	Avg	Lat. Exp. (Mils) at 180 Position 1	2	3	Avg.
658-101	E661-002-03	M2601-1	SA508CLII	Surge Nozzle	-50	-30[B]	+10	110	104	101	105	106	126	99	110	82	71	74	75.7	73	84	73	76.7
608-101	E661-002-03	M2602-1	SA508CLII	Spray Nozzle	-50	0[B]	-20	28	34	35	32.3	51	30	25	35.3	29	31	33	31	37	29	25	30.3
608-201	E661-002-03	M2603-1	SA508CLII	Safety Valve Noz	-50	-30[C]	+10	135	120	97	117	88	119	140	115	89	78	72	79.7	69	82	94	81.7
608-201	E661-002-03	M2603-2	SA508CLII	Safety Valve Noz	-50	-30[C]	+10	135	120	97	117	88	119	140	115	89	78	72	79.7	69	82	94	81.7
656-101	E661-002-03	M2611-1	SA508CLII	Support Forging	-50	-30[C]	+10	130	131	121	127	118	97	100	105	96	84	88	89.3	84	76	79	79.7

Piece Number	Drawing Number	Code Number	Material	Location	Drop Weight NDTT (°F)	RT_{NDT} (°F)	Test Temp (°F)	Charpy Energy Absorbed (ft-lbs) 1	2	3	Avg.	Lateral Expansion (Mils) 1	2	3	Avg.
642-101	E661-002-03	M2606-2	SA533BCLI	Shell Plate(Lower)	-50	-30[C]	+10	78	79	73	76.7	53	53	47	51
236-200	E661-002-03	M2610-1	SA533BCLI	Top Head	+30	+30[B]	+10	42	36	50	42.7	46	46	36	42.7
236-200	E661-002-03	M2610-2	SA533BCLI	Bottom Head	+30	+30[B]	+10	44	35	34	37.7	42	34	83	36.3
673-102	E661-002-03	M2637-8	SA516GR70	Supp. Ring Flange	+30	+30[B]	+10	39	34	56	43	38	44	33	38.3
673-104	E661-002-03	M2638-1	SA5166R70	Supp. Ring Segment	-50	-30[C]	+10	77	85	81	81	65	65	60	63.3
676-102	E661-002-03	C3529-1	SA516GR70	Manway Cover	-50	-30[C]	+10	51	65	51	55.6	73	68	64	68.3
622-102	E661-002-03	M2605-1	SA533BCLI	Upper Shell Plate	+10	+30[B]	+10	55	43	44	47.3	36	27	26	29.9
622-102	E661-002-03	M2605-2	SA533BCLI	Upper Shell Plate	-50	-30[B]	+10	58	59	59	58.7	41	40	39	40
642-102	E661-002-03	M2606-1	SA533BCLI	Lower Shell Plate	+30	+30[B]	+10	40	43	48	43.7	26	27	32	28.3

A-MTEB Position 5.2, "Fracture Toughness Requirements" Paragraph 1.1 (4), Minimum of 3 tests at a single temperature >45 ft-lbs.
B-MTEB Position 5.2, "Fracture Toughness Requirements" Paragraph 1.1 (4), Minimum of 3 tests at a single temperature >30 <45 ft lbs.
C-MTEB Powition 5.2, "Fracture Toughness Requirements" Paragraph 1.1 (3)b

TABLE 5.2-9 Revision 307 (07/13)

WATERFORD UNIT 3 STEAM GENERATOR MATERIALS FRACTURE TOUGHNESS DATA

→(EC-8458, R307)

Material Spec.	Material ID	Material Type	Heat Number	NDT (F)	NDT (C)	TCv50 ft lbs(F)	TCv50 ft lbs(C)	TCv35 mils (F)	TCv35 mils (C)	RT_{NDT} (F)	RT_{NDT} (C)	YS (ksi)	YS (MPa)	TS (ksi)	TS (MPa)	Elongation (%)	Reduction In Area (%)
SA-508 Grade 2 Class 2	F1	Forging	4584	0	-18	145	63	160	71	100	38	88.3	608.8	105.5	727.4	21.7	62.4
SA-508 Grade 2 Class 2	F2	Forging	5387	20	-7	115	46	90	32	55	13	83.2	573.7	99.3	684.7	20.5	57.0
SA-508 Grade 2 Class 2	F3	Forging	5389	50	10	130	54	105	41	70	21	89.1	614.3	105.9	730.2	20.4	59.2
SA-508 Grade 2 Class 2	FHAZ1	HAZ - SAW	4585/4109	-30	-34	15	-9	20	-7	-30	-34	93.0	641.2	111.6	769.5	25.7	49.1
SA-508 Grade 2 Class 2	FHAZ2	HAZ-SMAW	4585/3993, 4004 & 4009	-10	-23	50	10	40	4	-10	-23	93.9	647.4	105.2	725.4	21.8	38.9
SA-508 Grade 2 Class 2	FHAZ3	HAZ - SAW	5387/4109	-20	-29	15	-9	35	2	-20	-29	97.2	670.2	112.2	773.6	33.6	47.5
SA-508 Grade 2 Class 2	FHAZ4	HAZ - SAW	5389/4109	-10	-23	-5	-21	20	-7	-10	-23	101.9	702.6	117.8	812.2	35.0	55.0
SA-533 Type A, Class 2	P1	Plate	2864	30	-1	20	-7	20	-7	30	-1	82.3	567.5	102	703.3	22.8	---
SA-533 Type A, Class 2	P2	Plate	2899	-20	-29	-25	-32	-30	-34	-20	-29	79.6	548.8	101.4	699.2	26.1	---
SA-533 Type A, Class 2	P3	Plate	3272	10	-12	15	-9	15	-9	10	-12	74	510.2	94.5	651.6	25.6	---
SA-533 Type A, Class 2	P4	Plate	3312	-20	-29	-15	-26	-25	-32	-20	-29	69.4	478.5	89.5	617.1	26.4	---
SA-533 Type A, Class 2	SAW1	Weld – SAW	4336/4098	-80	-62	25	-4	15	-9	-35	-37	89.9	619.9	105.8	729.5	23.6	---
SA-533 Type A, Class 2	SAW2	Weld – SAW	4335/4098	-50	-46	40	4	40	4	-20	-29	90.9	626.8	106.5	734.3	56.2	---
SA-533 Type A, Class 2	SAW3	Weld – SAW	3742/3881	-60	-51	35	2	25	-4	-25	-32	81.6	562.6	98.9	681.9	56.8	---
SA-533 Type A, Class 2	PHAZ1	HAZ - SAW	4335/4098	-30	-34	0	-18	0	-18	-30	-34	97.4	671.6	114.2	787.4	48.8	---
SA-533 Type A, Class 2	PHAZ2	HAZ – SAW	4336/4113	-90	-68	-20	-29	-20	-29	-80	-62	93.9	647.4	113.8	784.7	49.4	---
SA-533 Type A, Class 2	PHAZ3	HAZ – SAW	3742/4113	-60	-51	0	-18	15	-9	-45	-43	84.4	581.9	100.5	692.9	56.4	---
SA-533 Type A, Class 2	GTAW1	Weld – GTAW	D4603/32J7	-50	-46	60	16	55	13	0	-18	103.9	716.4	120.6	831.5	48.2	59.8
SA-533 Type A, Class 2	GTAW2	Weld – GTAW	B1035/A070	-90	-68	-25	-32	-30	-34	-85	-65	109.2	752.9	123.2	849.5	58.8	68.2
SA-533 Type A, Class 2	GTAW3	Weld – GTAW	B1035/B481	-100	-73	-5	-21	-10	-23	-65	-54	101.4	699.2	116.5	803.3	55.7	68.4
SA-508 Grade 3, Class 2	F4	Forging	7341	-10	-23	165	74	67	19	105	41	81.1	559.2	99.7	687.4	22.4	58.6
SA-508 Grade 3, Class 2	F5	Forging	7431	80	27	175	79	92	33	115	46	92.5	637.8	111.1	766.0	18.3	53.8
SA-508 Grade 3, Class 2	FHAZ5	HAZ – SAW	7341 HAZ	-20	-29	80	27	85	29	25	-4	99.3	684.7	119.1	821.2	38.8	52.9
SA-508 Grade 3, Class 2	FHAZ6	HAZ – SAW	7431 HAZ	-60	-51	30	-1	40	4	-20	-29	101.3	698.5	121.3	836.4	39.9	58.0
SA-508 Grade 3, Class 2	F6	Forging	---	---	---	---	-37 to -43	---	-49 to -52	---	-25	---	510 to 570	---	650 to 700	---	---

←(EC-8458, R307)

REACTOR COOLANT LEAK DETECTION SENSITIVITY

Leakage Source	Detection Instrumentation	Instrument Range	Normal Reading	Average Rate of Change for 1.0 gpm leak	Time for Scale to Move 10% from Normal Reading for 1.0 gpm leak
1. Direct	Sump input flow measurement system	0-20 gpm***** (Nonlinear)	0	1.55 min first 25 percent change	Approximately 1 min.
→(DRN 04-1221, R13-A)	Sump level flow measurement system	gpm	NA	10 min	NA
←(DRN 04-1221, R13-A)					
	Containment Radiation	10-10^6 cpm	35,000 cpm*	94 cpm	1.0 hrs. approximately
2. Safety Valves	Valve Position Monitors	0-100%	Flow-No Flow	NA	NA
	Discharge Line Temperature	0-300 °F	Operating temp 120 °F	NA	NA
	Quench Tank Water Level	0-100%	NA	NA	NA
→(DRN 06-885, R15) 3. S.I. Tank Check Valves	S.I. Tank Water Level	0-100%	NA	NA	NA
	S.I. Tank Pressure	350-750 psig	600 psig	NA	NA
←(DRN 06-885, R15)					
4. Heat Exchangers	CCW Radiation	10-10^6 cpm	200,000 cpm**	300 cpm	2.0 hrs. approximately
	CCW Surge Tank Water Level	0-100%	72.80%	1 gpm leak takes 3-1/2 hrs for level to rise from normal 72.80% to high 98.70%	1 1/2 hrs.
5. Steam Generator Tubing	Blowdown Line Radiation	10-10^6 cpm	20,000 cpm***	300 cpm	2.0 hrs. approximately
	Condenser vacuum pumps exhaust Radiation	10-10^{6}cpm	70-100 cpm****	NA	NA
6. Reactor Vessel Closure Head	O-Ring Space Pressure	0-3000 psig	0	NA	NA

TABLE 5.2-10 (Sheet 2 of 2)

REACTOR COOLANT LEAK DETECTION SENSITIVITY

Leakage Source	Detection Instrumentation	Instrument Range	Normal Reading	Average Rate of Change for 1.0 gpm leak	Time for Scale to Move 10% from Normal Reading for 1.0 gpm leak
7. Reactor Coolant Pump Closure Cover	Flange Gasket leak of pressure	0-3000 psig	0	NA	NA

* Based on 0.1 gpm and 0.1% failed fuel, noble gas monitor faster sensibility can be achieved with the particulate monitor for lower leakages or lower percentages of failed fuel. However particulate monitor will be surveyed at this level. Step advance on filter paper will allow determination of leakage increase by taking readings of identical duration.

** Assumes leak from one of the potentially very radioactive components (i.e., 0.1% failed fuel content). If leak is from lower activity component, leakage change can be detected more rapidly.

*** Variable as a function of blowdown rate, assumed rate < 8500 lbm/hr.

**** Based on Xe 133 which represents bulk of activity released.

***** Linearization performed within plant analog control system (PAC) by a function generator before signal is used as input to computer and flow recorder.

TABLE 5.2-11 (Sheet 1 of 5) Revision 305 (11/11)

ISI FOR VALVES WHICH FORM THE PRESSURE BOUNDARY OF THE RCS

VALVE NUMBER	CLASS	VALVE CATEGORIES_ PER ASME CODE SECTION XI, IWV A B C D	SIZE (INCHES)	VALVE TYPE (1)	ACTUATOR TYPE (2)	FAILURE POSITION (3)	TEST REQUIREMENTS (4)	CLARIFICATION (5)	TESTING_ ALTERNATIVES	LEAK RATE TEST VALUE (GPM)	POSITION (6)
→(EC-935, R302)											
1501 B (SI-665) SI-405B	1	X	14	GA	PP	FC	CS, LT	3	-	5	LC
←(EC-935, R302) →(DRN 06-897, R15) →(EC-14765, R305)											
SI-4052B	1	X	3/4	GL	S	FC	CS, LT	3	-	5	LC
←(EC-14765, R305)											
1502 B (SI-666) SI-401B	1	X	14	GA	M	-	CS, LT	3	-	5	LC
→(EC-935, R302)											
1503 A (SI-651) SI-405A	1	X	14	GA	PP	-	CS, LT	3	-	5	LC
←(DRN 06-897, R15; EC-935, R302) →(EC-14765, R305)											
SI-4052A	1	X	3/4	GL	S	FC	CS, LT	3	-	5	LC
←(EC-14765, R305)											
1504 A (SI-652) SI-401A	1	X	14	GA	M	-	CS, LT	3	-	5	LC
1509 RL1A (SI-217) SI-335A	1	X X	12	CH	-	-	CS, LT	1	-	1	C
1510 TK1A (SI-215) SI-329A	1	X X	12	CH	-	-	CS, LT	1	-	1	C
1511 RL1B (SI-227) SI-335B	1	X X	12	CH	-	-	CS, LT	1	-	1	C

TABLE 5.2-11 (Sheet 2 of 5) Revision 305 (11/11)

ISI FOR VALVES WHICH FORM THE PRESSURE BOUNDARY OF THE RCS

VALVE NUMBER	CLASS	VALVE CATEGORIES PER ASME CODE SECTION XI, IWV A	B	C	D	SIZE (INCHES)	VALVE TYPE (1)	ACTUATOR TYPE (2)	FAILURE POSITION (3)	TEST REQUIREMENTS (4)	CLARIFICATION (5)	TESTING ALTERNATIVES	LEAK RATE TEST VALUE (GPM)	POSITION (6)
1512 TK1B (SI-225) SI-329B	1	X		X		12	CH	-	-	CS, LT	1	-	1	C
1513 RL2A (SI-237) SI-336a	1	X		X		12	CH	-	-	CS, LT	1	-	1	C
1514 TK2A (SI-235) SI-330A	1	X		X		12	CH	-	-	CS, LT	1	-	1	C
1515 RL2B (SI-247) SI-336B	1	X		X		12	CH	-	-	CS, LT	1	-	1	C
→(DRN 06-897, R15)														
1516 TK2B (SI-245) SI-330B	1	X		X		12	CH	-	-	CS, LT	2	-	1	C
←(DRN 06-897, R15)														
1517 RL1A (SI-114) SI-143B	1	X		X		8	CH	-	-	CS, LT	1	-	1	C
1518 RL1B (SI-124) SI-142B	1	X		X		8	CH	-	-	CS, LT	1	-	1	C

TABLE 5.2-11 (Sheet 3 of 5) Revision 305 (11/11)

ISI FOR VALVES WHICH FORM THE PRESSURE BOUNDARY OF THE RCS

VALVE NUMBER	CLASS	VALVE CATEGORIES (PER ASME CODE SECTION XI, IWV A B C D	SIZE (INCHES)	VALVE TYPE (1)	ACTUATOR TYPE (2)	FAILURE POSITION (3)	TEST REQUIREMENTS (4)	CLARIFICATION (5)	TESTING ALTERNATIVES	LEAK RATE TEST VALUE (GPM)	NORMAL POSITION (6)
1519 RL2A (SI-134) SI-143A	1	X X	8	CH	-	-	CS, LT	1	-	1	C
1520 RL2B (SI-144) SI-142A	1	X X	8	CH	-	-	CS, LT	1	-	1	C
1522 RL1A (SI-113) SI-241	1	X X	3	CH	-	-	CS, LT	1	-	1	C
1523 RL1B (SI-123) SI-242	1	X X	3	CH	-	-	CS, LT	1	-	1	C
1524 RL2A (SI-133) SI-243	1	X X	3	CH	-	-	CS, LT	1	-	1	C
1525 RL2B (SI-143) SI-244	1	X X	3	CH	-	-	CS, LT	1	-	1	C
2506 SI-510A	1	X X	3	CH	-	-	LT	1	-	1	C
2507 SI-512A	1	X X	3	CH	-	-	LT	1	-	1	C
2508 SI-510B	1	X X	3	CH	-	-	LT, CS	1	-	1	C
2509 SI-512B	1	X X	3	CH	-	-	LT, CS	1	-	1	C

TABLE 5.2-11 (Sheet 4 of 5) Revision 305 (11/11)

ISI FOR VALVES WHICH FORM THE PRESSURE BOUNDARY OF THE RCS

Notes:

1) Valve Type

GA - Gate
CH - Check
GL - Globe

2) Actuator Type

HP - Hydraulic, Pneumatic
M - Motor
DP - Diaphragm, Pneumatic
→(EC-935, R302)
PP – Piston, Pneumatic
←(EC-935, R302)
→(EC-14765, R305)
S – Solenoid
←(EC-14765, R305)

3) Failure Position

FC - Fail Closed

4) Test Requirements

Q - Exercise valve (full stroke) for operability every three months

LT - Valves are leak tested per Section XI, Subsection IWV

MT - Stroke time measurements are taken and compared to the stroke time function every three months

CV - Exercise check valves to the position required to fulfill their function every three months.

SRV - Safety and relief valves are tested per Section XI, Subsection IWV

DT - TEST Category D valves per Section XI, Subsection IWV

CS - Exercise valve for operability every cold shutdown

RR - Exercise valve for operability every reactor refueling

5) Clarification
→(DRN 06-897, R15)

1. Exercising and leak testing will be performed once per refuel outage, prior to unit start-up of reduced pressure.

←(DRN 06-897, R15)

TABLE 5.2-11 (Sheet 5 of 5) Revision 15 (03/07)

ISI FOR VALVES WHICH FORM THE PRESSURE
BOUNDARY OF THE RCS

→(DRN 06-897, R15)

2. Exercising and leak testing will be performed once per refuel outage prior to unit start-up. Leak rate will be measured by rise in water level in the Safety Injection Tank.

3. Exercising and leak testing will be performed once per refuel outage prior to unit start-up at full pressure.

←(DRN 06-897, R15)

4. Leak testing performed at reduced pressure.

General: Provisions have been made for part stroke exercising of Class 1 check valves in the SIS and RHR Systems during normal operation. However, it is considered that probable system upset during part stroke exercising may do more damage than benefit of partial exercising. Thus, these valves will be full stroke exercised during refueling . This follows the requirements of ASME Section XI, Subsection IWV.

6) Normal Position

LC - Locked Closed
C - Closed